Appendices:

1.0 INTRODUCTION TO THE GINNA STATION EXTENDED POWER UPRATE LICENSING REPORT

General Overview of the Ginna Extended Power Uprate (EPU) Licensing Report

The Ginna Nuclear Power Plant, LLC (Ginna) EPU Licensing Report (LR) is a technical summary of the results of the analyses and evaluations performed to demonstrate that the proposed increase in plant power can be safely achieved and that the increase will not be inimical to the common defense and security or to the health and safety of the public. The LR is an attachment to the license amendment request submitted requesting permission to increase plant power level. The LR provides the details that support the requested license and technical specification changes and works in concert with the other attachments to the amendment request to provide reviewers with a comprehensive evaluation of the effects of the proposed EPU.

The fundamental outcome of the Ginna evaluations have been formatted and documented in accordance with the template and criteria provided in RS-001, "Review Standard for Extended Power Uprates," revision 0. The role of the LR is to document the technical basis for the evaluation of the effects of the proposed changes necessary to implement EPU with a sufficient level of detail to permit the NRC staff to reach an informed determination regarding the consistency, quality, and completeness of the evaluation with respect to the areas that are within the NRC's scope of review. It is recognized that the NRC will review the report in its entirety and will perform independent evaluations, calculations and audits as deemed necessary and appropriate to reach its own conclusion concerning the effects of the proposed EPU and continued safe operation of the Ginna Station. To that end, careful attention has been paid to ensure that the technical evaluations presented in the LR include, when appropriate, discussion of the effects of EPU on plant operating limits, functional performance requirements and design margins as well as describing the methods Ginna used in reaching the conclusions documented in the report consistent with the guidance of RS-001. To enhance the efficiency of the NRC review, Ginna has included identification of any differences between the information in the review standard and the design bases of the Ginna Station.

Ginna used RS-001 to the extent possible and added information to the licensing report that is specific to a good understanding of the effects of EPU on Ginna Station, as appropriate. Accordingly, reviewers of the LR must understand the parameters that influenced its format and content. Factors considered salient to the understanding of the LR are described below.

Summary of Plant Modifications and Schedule

The discussion below provides a summary level description of how Ginna Station will achieve the power uprate. The description follows the energy flow from its source (the fuel assembly) to the eventual delivery of electrical energy to the grid. Significant changes in plant parameters are discussed, as well as a brief description of the basis for the plant modifications that are required. Table 1.0 provides a listing of all the required plant modifications for reference. As indicated in the table, the majority of these

modifications are planned for implementation in the 2006 refueling outage. Four of the modifications were completed in the 2005 refueling outage. In addition to the plant modifications, many setpoints and control system settings will require adjustment. The setpoint and setting changes are not discussed here, but are discussed in detail in LR section 2.4.1. The setpoint changes are also planned for the 2006 refueling outage. Power escalation to the new uprate power level is planned immediately after the 2006 refueling outage.

Kewaunee has successfully achieved an uprate to 1772 MWt, 3 MWt less than the proposed EPU for Ginna. Kewaunee and Ginna are both two-loop Westinghouse nuclear steam supply system (NSSS) plants. A comparison of the significant NSSS design parameters for both the uprated Ginna and the current Kewaunee is provided in Table 1.0-1. Kewaunee has both larger reactor coolant pumps and a larger pressurizer as is evidenced by the larger volumes shown in Table 1.0-1. These differences have been accounted for in the Ginna EPU safety analysis with acceptable results.

Fuel Assembly

The fuel assembly used for the uprate will be similar to that used now at Point Beach and Kewaunee. The fuel assembly design is the 14X14 422V+ design which is the updated Westinghouse generic design for their 14X14 plants. The fuel rod and fuel pellet design for the Ginna EPU will be the same as that used at Point Beach and Kewaunee. The Ginna fuel assembly skeleton design will differ slightly from the Point Beach and Kewaunee design (e.g. nine grids versus seven grids) for compatibility with the current Ginna Optimized Fuel Assembly (OFA). As compared to the current OFA assembly now in use at Ginna, the new assembly utilizes a shorter top nozzle which allows for approximately 3" longer fuel rods. The added rod length along with an increase in the fuel pellet diameter (by 0.021") and increased enrichment will allow loading additional U-235 to support the increase in energy requirements at the uprate power level. Fuel handling equipment has been modified to accommodate both the new short top nozzle and the current tall top nozzle. Since the new assembly top nozzle is shorter, and the rod stop is on the bottom of the top nozzle, the control rods in the new assemblies will sit about 3" higher when fully inserted. The offset control rods will cause a different rod position indication signal for 'rods on bottom' that will be addressed by a modification to the rod position indication software, or by adjusting the position indication coils upward using spacers beneath the coil stacks. Ginna will implement a full region of the new assemblies for the first time in the fall 2006 refueling outage.

Reactor Core

Core Design:
The uprated core will operate at a core thermal power of 1775 MWt as compared to the current core thermal power of 1520 MWt. This represents an increase of 16.8% in core thermal power. The uprate core design will be very similar to the core design now in use at Ginna. The core will utilize a low leakage loading pattern. Fuel enrichment will be increased slightly on average, as will burnable poison loading, to meet the uprate energy requirements. Core power peaking limits will remain essentially the same. Core cycle length (EFPD) will remain essentially the same as for current designs. The core power density will increase to support the uprate power increase resulting in a proportionally higher coolant temperature increase across the core. The higher core outlet

temperature will increase the boiling fraction in the upper regions of the core. Ginna will address the potential operational challenges that could arise if Axial Offset Anomaly (AOA) were to occur by implementing the Relaxed Axial Offset Control (RAOC) power distribution control methodology. The RAOC methodology allows the axial flux difference control bands to be widened by removing conservatism inherent in the Constant Axial Offset Control (CAOC) methodology currently used at Ginna. Kewaunee has not experienced AOA while operating at the same core thermal power using a similar fuel assembly design. In order to gain margin for the rod withdrawal accident, the power mismatch gains to the rod control system are being changed to reduce the rod withdrawal speed in response to a nuclear power vs. turbine power mismatch.

Core Decay Heat:
Core decay heat will increase proportional to the uprate core power increase. The higher decay heat results in the need for minor modifications to systems that are used to remove decay heat. The required flow from the Standby Auxiliary Feedwater (SAFW) pump will increase from 225 gpm in a feed line break scenario to 235 gpm. To achieve this flow the SAFW pump discharge valve internals will be replaced. The required minimum volume in the Condensate Storage Tank (CST) will be increased to 24350 gallons from 22500 gallons to provide the inventory needed to remove the integrated decay heat for at least two hours after a trip. This increase in volume will require a minor modification to raise the CST overfill line. To meet Appendix R requirements to achieve safe shutdown in 72 hours with no RHR available, Appendix R procedures will require steaming through the Atmospheric Relief Valves (ARV) on both steam generators. Current procedures require steaming via only the 'A' steam generator ARV. This procedure change will also require fire hardening of the 'B' steam generator level indication to assure this indication is available. The required flow rate from the SAFW pump in this same Appendix R loss of RHR scenario will increase from 225 gpm to 250 gpm to provide water-solid steam generator heat transfer. This change will require additional spool pieces to be manufactured to support repair efforts directed by the Appendix R cool down procedures. To further enhance operator response in Appendix R fire scenarios and reduce the core damage frequency, local control will be provided for the Turbine Driven AFW pump discharge valve and the 'A' charging pump control power disconnect switch will be relocated. Finally, again to reduce core damage frequency, back-up air tanks will be installed for the charging pumps to assure their availability on loss of air.

Core Accident Source Term:
The core accident source term will increase approximately proportional to the uprate power level increase. The higher source term may require modification of the containment fan cooler motor and/or cabling in order to maintain equipment qualification.

Reactor Coolant System

The reactor coolant system operation will change very little for uprate. The best estimate reactor coolant flow will ultimately increase a small amount (less than 0.5%) with a full core of the new fuel assemblies as the new fuel assembly has less flow resistance. The system operating pressure will not change. Other than the new fuel assembly, there are no physical modifications planned to the reactor coolant system or reactor vessel internals. In order to provide the necessary steam pressure at the uprate power level, the full power best estimate average coolant temperature will

increase 12°F to approximately 573°F. The system was operated very close to this average coolant temperature prior to replacing steam generators in 1996. T_{avg} was reduced since 1996 after steam generator replacement to reduce turbine inlet steam pressure and mitigate Primary Water Stress Corrosion Cracking of Alloy 600 components. However, since Ginna replaced the reactor vessel head in 2003, and modified the remaining hot side pressure boundary so as to eliminate components with Alloy 600 material, elevated hot side temperature concerns have been reduced. As previously mentioned, the coolant temperature increase across the core will increase approximately proportional to the increase in uprate power. With the higher T_{avg} and higher core temperature difference, the reactor vessel outlet temperature (T_{hot}) will increase 17°F to 606°F. The reactor vessel inlet temperature (T_{cold}) will also increase about 7°F to 539°F. The reactor coolant system zero-power T_{avg} will not change from the current value of 547°F. The larger change in T_{avg} from full power to zero power will result in a greater shrink in pressurizer level following a reactor trip. This in turn will require an increase in the full power pressurizer program level to 57% (from 50%) and a decrease in the zero power pressurizer program level to 20% (from 35%). As previously stated the pressurizer pressure will not change. As a result of the expected incremental increase in hot leg temperature streaming a 3.5 second time delay filter will be installed on the T_{hot} signal that feeds control and protection circuits to dampen oscillations of the signal for hot leg temperature.

Steam Generator and Main Steam

The Ginna steam generators were replaced in 1996 with a Babcock & Wilcox, Canada steam generator model that is oversized for the pre-uprate power level. The best estimate steam generator steam pressure for the uprate will increase to 798 psia (from 770 psia) as a result of the increase in RCS T_{avg}. The mass flow rate of steam will increase proportional to the power increase to deliver the energy to the main turbine. Steam velocity in the main steam piping will increase due to the steam mass flow rate increase. The higher velocity will increase the forcing function for piping vibration. An extensive vibration monitoring program will be implemented during the power escalation process to verify vibration levels remain acceptable. Analysis of the dynamic forces associated with closure of the main turbine stop valve will result in minor modifications to main steam system pipe supports. The turbine steam inlet pressure will be 730 psia, slightly higher than the current inlet pressure of 720 psia. Since the condenser steam dump valves will not be modified, the capacity of the steam dump system relative to full power steam flow will be less after uprate. To maintain the plant load rejection capability the steam dump control system settings will be adjusted to be more responsive to changes in T_{avg} relative to T_{ref}. Steam generator blowdown flows will be increased slightly on average to remove the additional contaminants delivered to the steam generator by the higher return feedwater flow.

Main Turbine

The high pressure turbine will be replaced in order to pass the additional volumetric steam flow. The new high pressure turbine is a full arc admission design that will operate with the turbine control valves wide open at full power. The turbine control valves will also be replaced to reduce the pressure drop through the valves and replace the current linkage system with in-line direct actuators. The new valves will deliver more energy to the turbine and result in approximately 3.5 additional MWE. The low pressure turbines will not be modified as they have adequate flow margin to pass the higher

volumetric flow rate. The steam pressure to the moisture separator reheater (MSR) will increase in order to pass additional steam through the low pressure turbine. The higher MSR pressure and increased steam flow will require an increase in the MSR relief valve setpoints and new valves to increase the relieving capacity of the system. Five rupture discs will be used in addition to one existing valve to maximize relieving capacity at the required back pressure. The main turbine over speed trip setpoint will be reduced to compensate for the increase in steam flow at full power. Increased high pressure turbine gland leakage will require modification to gland seal spillover.

Main Condenser and Circulating Water

The higher main steam flow will result in a reduction of 1.5" Hg in the main condenser vacuum. This will affect plant efficiency such that, even though the new high pressure turbine is more efficient than the old turbine, the heat rate will increase approximately 0.5%. The temperature of the circulating water discharged to Lake Ontario will increase about 3°F as additional heat is rejected. A request has been submitted to modify the State Pollution Discharge Elimination System (SPDES) permit to increase both the maximum discharge temperature limit (to 106°F from 102°F) and the winter-time condenser delta-T limit (from 28°F to 35°F). The circulating water system will not be modified, so the circulating water flow rate will remain unchanged. The increase in heat rejected to Lake Ontario will also increase the lake surface area that will be affected (increased by 3°F or more) by the plant discharge. This will also be addressed in the SPDES permit modification with the State of New York.

Condensate and Feedwater

The condensate and feedwater flow rates will increase proportional to the uprate power increase. Higher capacity condensate booster and main feedwater pumps will be installed to deliver the needed flow to the steam generators. The condensate pumps will not be modified. Even with the higher capacity condensate booster pumps, the additional head loss in the condensate and feedwater piping will result in less NPSH margin for the main feedwater pumps. The setpoints associated with NPSH margin will be changed and a time delay will be installed on the automatic feedwater heater bypass function to reduce the likelihood of reactivity transients. The feedwater regulating valve internals will be replaced to pass the additional feedwater flow. The feedwater temperature to the steam generator will increase to 432°F from 425°F. A fast-acting automatic actuator will be added to a manual feedwater isolation valve in the intermediate building to provide more rapid isolation of feedwater in a steam line break scenario. This new actuator will also reduce the required core shutdown margin requirement in the COLR since this modification limits the cooldown in the worst overcooling scenario. Analysis of the dynamic forces associated with closure of the feedwater regulating valve will result in minor modifications to feedwater system pipe supports. Main feedwater pipe vibration will also be monitored closely as higher feedwater flow velocity will increase the forcing function.

Extraction Steam and Heater Drains

There will be a slight increase in the temperatures, pressures and flows in the extraction steam piping and in the various heater drains. This will require various minor modifications to assure each drain flow path is capable of passing the additional flow. Monitoring of the various drain and level control systems will be an important part of the power escalation procedure. It is likely that these systems will require tuning to assure stable control. The erosion-corrosion program will be updated for the new conditions in these systems at uprate. Certain lines will require increased monitoring but all lines will be within the capability of the program to be monitored safely.

Main Generator

The main generator electrical output will increase by approximately 86 MWe. The machine will be re-rated from 608.4 MVA to 667 MVA with allowable power factor of 0.92 (lagging) and 0.975 (leading). The Ginna main generator is very similar to the Connecticut Yankee (CY) main generator and the CY generator was rated for the same conditions that the Ginna generator will see after uprate. Additional monitoring instrumentation was installed during the 2005 refueling outage to monitor stator winding partial discharge activity, stator winding end turn vibration, and rotor winding shorts. This installation will allow station personnel to acquire base line performance data prior to increasing output power for the uprate. The indications from this instrumentation post-2005 outage show that the winding performance is acceptable. However, given the large increase in current and the age of the windings, the winding performance will be closely monitored after the uprate and plans will be in place to rewind the stator should that need arise. The additional current in the stator windings will create additional heat that will require an increase in the cooling capacity of the hydrogen cooling system. The condensate cooler that supplies cold water to the hydrogen coolers will be replaced to provide additional cooling in the summer months. A dynamic transient analysis of a close-in fault on the main generator identified the need to modify the generator to exciter coupling key way in order to withstand the higher stress associated with the fault. This modification was completed during the 2005 refueling outage. Generator protective relays and voltage regulator settings will be adjusted for operation at the uprate power conditions.

Iso-Phase Bus Ducts/Generator Step-up Transformer/Oilstatic Grid Feeder Cables

To transfer the power from the main generator to the grid the design capacity of the Iso-Phase Bus Duct system will be increased. The bus duct cooling fan capacity will be increased to provide the additional cooling. The Generator Step-Up (GSU) transformer capacity has also been increased to meet uprate power requirements. During the 2005 refueling outage the GSU high voltage bushings were replaced and a fifth cooler was added to assure the transformer cooler reliability requirements were at least as robust for uprate as prior to uprate. At uprate power conditions, forced oil cooling capability of the oilstatic cables that feed the power underground to the grid will be activated during summer months to provide additional cooling for these cables. The original design of this system incorporated a forced flow capability, but this capability has not previously been utilized. Forced circulation of the oil alone to transfer the heat to the environment is adequate to provide the necessary cooling without addition of a cooling water system.

Additional monitoring instrumentation will be installed to provide continuous monitoring of temperatures in the system. The differential current relay that protects the oilstatic cables will also be replaced to increase its current rating.

Switchyard Station 13A

A grid stability study was performed to verify that while operating at the uprate power, Ginna Station will not impact the reliability of the grid. Station non-safety power is normally supplied by Transformer 11 which is fed directly from the main generator. During abnormal scenarios individual switchyard circuit breakers (1G13A72 and 9X13A72), their associated current transformers and disconnect switches will have an individual long term current rating that may limit plant power output when one of the switchyard circuit breakers is out for maintenance. This configuration will be managed administratively as it is very unlikely that critical equipment will be removed from service while power is being supplied to the grid.

Table1.0
Ginna Power Uprate Planned Modifications and Schedule for Implementation

#	Modification	Planned Implementation
1	Modified Fuel Assembly and Core	2006 RFO
2	Condensate Booster Pump and Motor Upgrade	2005 RFO & 2006 RFO
3	Main Step Up Transformer Bushing and Cooler Replacement	2005 RFO **COMPLETE**
4	MSR Relief Valve Modification	2006 RFO
5	Install 10" Heater Drain Tank Emg. Drn. Valve & Add 12" Disengagement Chamber for 2nd Pass Reheater	2006 RFO
6	Generator Instrumentation	2005 RFO **COMPLETE**
7	Fuel Handling Equipment Modification to Accommodate New Fuel Assembly	2005 RFO **COMPLETE**
8	Replace 1A/B FWH Normal Vent Orifices & Add 1" Vent to Reheater 4th Pass Drain Tank	2006 RFO
9	Upgrade HP Turbine and TCVs	2006 RFO
10	Modify Exciter Coupling Keyway	2005 RFO **COMPLETE**
11	Install Actuator for Feed Isolation Valve	2006 RFO
12	Replace Generator Condensate Cooler	2006 RFO
13	Main Feed Pump Impeller/Motor	2006 RFO
14	Feed Regulating Valve Trim Change	2006 RFO
15	Iso-Phase Bus Duct Fan and/or Motor Upgrade	2006 RFO
16	Oilstatic Cable Monitoring Instrumentation	On-Line 2006
17	Generator Protection and Voltage Regulator Setting Changes	2006 RFO
18	Instrumentation Replacement/Scaling/Thot Filter (See LR section 2.4.1)	2006 RFO
19	Standby AFW MOV Trim Change	2006 RFO
20	Raise CST Overfill Line	2006 RFO
21	Replace Various Snubbers and Rods on Main Steam and Feedwater Supports	2006 RFO
22	Replace Oilstatic Cable Differential Current Protection Relay	2006 RFO
23	Provide SG Water Solid Cooldown Spool Pieces	On-Line 2006
24	Modify/Shield Containment Fan Cooler Motor and Cable (potential)	2006 RFO
25	MRPI Coil Stack Spacer/Software Change to Address New Fuel Assembly Height	2006 RFO
26	Provide Local TD AFW Pump Discharge MOV Control for Appendix R Scenarios	2006 RFO
27	Relocate Charging Pump Control Power Disconnect for Appendix R Scenarios	2006 RFO
28	Install Air Tanks for Backup Air Supply to Charging	2006 RFO

#	Modification	Planned Implementation
	Pumps	
29	Fire Harden B SG Level Indication	2006 RFO
30	Modify Turbine Gland Sealing Steam Spillover	2006 RFO

Table 1.0-1
COMPARISON OF GINNA AND KEWAUNEE NSSS DESIGN PARAMETERS

Parameter	GINNA	KEWAUNEE
Total Core Power	1775 mwt	1772 mwt
System Pressure	2250 psia	2250 psia
Minimum Reactor Flow	85,200 gpm/loop	89,000 gpm/loop
Coolant Volume with Pressurizer	6084 ft^3	6435 ft^3
Pressurizer Volume	800 ft^3	1000 ft^3
Maximum Inlet Temperature	540.2°F	539.2°F
Maximum Average Temperature	576.0°F	573.0°F
Maximum Outlet Temperature	611.8°F	606.8°F

Current Licensing Basis

The Ginna Station Current Licensing Basis (CLB) is presented in the Updated Final Safety Analysis Report (UFSAR). The CLB includes the application of various general design criteria (GDC) originally developed by industry (Atomic Industrial Forum) in the early 1960s to provide some consistent guidance to evaluate the design and performance of commercial nuclear power plants which were beginning to proliferate as an emerging industry. The fundamental precept in the development of these GDC was to identify plant performance criteria which, if satisfied by specific design of structures, systems, and components, would provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public with respect to the radiological consequences of normal plant operation and unlikely postulated accidents. The GDC are intended to provide general performance objectives against which the facility design may be judged. Absent significant commercial nuclear plant operating experience at the time, the safety philosophy at this early stage of industry development was to incorporate appropriate engineered safeguards with conservative safety margins into plant design and into the analyses of the probability and consequences of postulated accidents.

As the commercial nuclear power industry began to mature, both the industry and the regulatory agency gained the benefit of increased, although still limited, operating experience and advances in technology. This resulted, over time, in an increase in regulatory guidance and requirements for plant design and operation. Substantial additional regulatory guidance and requirements were promulgated subsequent to the September 1969 issuance of the Ginna Operating License. The additional guidance and requirements often took the form of expanded details or additional alternative approaches based on advances in technology and/or the NRC staff understanding of plant operations and processes for demonstrating regulatory compliance.

In 1971, the Atomic Energy Commission issued Appendix A to 10CFR50, "General Design Criteria," (36FR03255, 2/20/71). These proposed GDC had been published for interim use in 1967. The GDC are intended to establish minimum requirements for the principal design standards. The Appendix A GDC were in many instances similar to the former AIF GDC, but they were expanded to include additional design considerations (both AIF and NRC GDC are discussed in the Ginna UFSAR). Ginna historical documents indicate that the AEC GDC issued for interim use in July 1967 were applied to the Ginna design prior to issuance of the Operating License. Details concerning Ginna's design relative to the GDC are found in UFSAR section 3.1, which addresses conformance to both the AIF GDC (section 3.1.1) and the NRC GDC (section 3.1.2)

Through the 1970s and into the 1980s there was a significant number of new regulatory requirements and guidance documents issued. By the late 1970s, the documented scope of review for plant licensing had expanded substantially from the era when Ginna, and several other older operating plants, was licensed. Federal regulations were expanded, Safety Guides (later renamed Regulatory Guides) were originated and continuously issued, Branch Technical Positions and Standard Review Plans (NUREG-75/087 dated 12/75) were developed, and various NRC generic communications were being continuously issued.

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to assess the design of older operating plants in light of the more current licensing criteria and to

identify any substandard performance of safety systems based on plant operating experience. As presented in SECY-76-545 dated 11/12/76, based at least in part on concerns raised by the Congressional Joint Committee on Atomic Energy and the Advisory Committee for Reactor Safeguards, the NRC recognized that it needed a more thoroughly documented evaluation of the disparity between technical positions on safety issues that existed in the mid- to late-1970s and those that existed when a particular plant was licensed. The objective of the SEP was to effectively and convincingly confirm the level of safety provided in operating plants and to maintain the documented acceptability of that level of safety. The NRC acknowledged that, in general, the detailed acceptance criteria prescribed in the SRPs (NUREG-75/087) did not represent new criteria; rather, they were current methods of review that, in most cases, were not previously published in any regulatory document. Therefore, the underlying concern that initiated SEP was not one of safety, but rather comparison to current regulatory criteria and its documentation. The SEP included Ginna. It was acknowledged that variation in conformance to with the newer licensing criteria existed, but the safety of the Ginna design was confirmed by the NRC SEP in NUREG-0821 completed in August 1983. In NUREG-0821, the NRC additionally emphasized that all plants, regardless of licensing date, have been reviewed against a substantial number of major safety issues that have evolved since the operating license was issued. Modifications resulting from SEP were based on value-impact assessments and the principles of 10CFR50.109.

Each section of the Ginna Station Extended Power Uprate (EPU) Licensing Report (LR) that details a structure, system or component (SSC) or selected analyses contains a brief description of Ginna's Current Licensing Basis (CLB) with respect to the SSC or analysis under evaluation. The basis for providing a CLB discussion is twofold: Foremost, Ginna's licensing basis is comprehensive in that it incorporates the results of the NRCs SEP, as well as other post construction regulatory initiatives such as TMI Lessons Learned. Secondly, Ginna has recently received a renewed Operating License. Because Ginna is still several years away from entering its extended period of operation, the results of the NRCs license renewal evaluations are not incorporated into the current UFSAR. However, EPU LR section address, as applicable, anticipated impact of EPU on license renewal evaluations.

Recent licensing reviews by the NRC rely on the SRP as the base document for defining the areas of review and the acceptance criteria for the structure, system, component, or analysis of interest. SRPs were first compiled and issued as NUREG-75/087 in 1975. The numerical structure of individual sections closely conformed to the numerical structure of the Standard Format and Content Guide for Safety Analysis Reports (SARs) in effect at the time in order to provide a logical correlation between SRP sections and their corresponding sections in the SAR. SRPs are today compiled in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and are numerically structured consistent with RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The acceptance criteria are intended confirm the safety basis for compliance with applicable Title 10 NRC regulations. Typically, the GDC acceptance criteria involve demonstrating compliance with specific regulations, the regulatory positions of Regulatory Guides, and/or Branch Technical Positions (which are an adjunct to many GDC). It is important to note that the GDC are not regulatory requirements but rather define technical approaches to plant design and analysis which, if implemented by the applicant, would be found acceptable by the NRC for safety compliance purposes. Alternate approaches may be proposed by the applicant for NRC acceptance review.

In December 2003, the NRC issued its Review Standard for Extended Power Uprate (EPU), RS-001. This Licensing Report (LR) submitted to support NRC approval of the proposed EPU for the Ginna Station is intended to conform, to the maximum practical extent, to the guidance of RS-001. The regulatory review criteria portion of the RS-001 section details specific NRC review and acceptance criteria. The review standard acknowledges that there can, and will, be differences between the review standard and the design basis of a particular facility. The review standard contains provisions to ensure these differences do not impede the NRC staff's review. Consistent with the review standard, and because of the evolution of Ginna's licensing basis, additional details concerning Ginna's CLB are provided at a level sufficient to supply the NRC reviewers with a clear and concise summary of the licensing basis with respect to the SSC design or analyses under review.

Treatment of Issues Related to the Renewed Operating License

By letter dated July 30, 2002, the Ginna Nuclear Power Plant, LLC (Ginna) submitted to the NRC an application requesting the Nuclear Regulatory Commission (NRC) to renew the Ginna Operating License for up to 20 additional years. The NRC reviewed the application in accordance with 10CFR54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," utilizing the guidance of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants." The NRC completed its review and approved the Ginna license renewal application as documented in NUREG-1786, "Safety Evaluation Report Related to the License Renewal of R.E. Ginna Nuclear Power Plant," in May 2004.

In the Safety Evaluation Report (SER), structures, systems and components (SSCs) subject to aging management review are discussed in sections 2.3 through 2.5. For those identified SSCs, the specific applicable aging management programs are discussed in sections 3.1 through 3.6.

The requirements for renewal of nuclear power plant operating licenses are contained in 10CFR54 which identifies plant SSCs that are within the scope of the rule (10CFR54.21), as well as requirements for performing aging management reviews of those SSCs. Additionally, the rule requires an evaluation of time-limited aging analyses (TLAA) to account for the effects of aging on the intended functions of SSCs that are not subject to replacement based on a qualified life or specified time period. The TLAA are intended to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The operating conditions associated with the proposed extended power uprate (EPU) may increase certain operating parameters such as pressure, temperature, flow, and radiation compared to current operating conditions. In addition, the EPU introduces the possibility that components not currently within the scope of the rule (either currently installed in the plant or added as the result of EPU) may, as the result of EPU, meet the scoping inclusionary criteria detailed in the rule.

As discussed in each section of this LR which addresses specific SSCs, an evaluation of the impact of EPU on the extended period of operation of the plant was performed. The purpose of this evaluation was to identify which, if any, SSCs warranted additional aging management action. These may include SSCs subject to new aging effects because of

changes in the operating environment resulting from EPU or the addition of, or modification to, components relied upon to satisfy EPU operating conditions.

SSCs relied upon for achieving the license renewal scoping objectives are evaluated within the structure or system that contains them. A limited number of plant components which were not included within the initial scope of License Renewal were evaluated within the context of the operating conditions associated with the proposed EPU. Any additional aging management actions associated with the extended period of plant operation are captured as internal commitments which signify the need to modify the scope of license renewal aging management programs.

The potential impact of the proposed EPU on license renewal TLAA was also evaluated. Specifically, the evaluation considered any new aging effects or increases in degradation rates potentially created by the new EPU operating parameters. In addition to the discussion contained in the individual LR section, the impact of EPU on license renewal TLAA is further discussed in LR section 2.14.

Effect of EPU on Plant Programs

In addition to the aging management programs defined in the license renewal SER, numerous ongoing initiatives and processes used at the facility can be characterized as programs. In order to ensure these programs remain effective after EPU a review was performed to identify which, if any, programs could be impacted by the changes associated with EPU. The review considered two basic impacts:

- Does EPU increase the scope of an existing program? (i.e., should additional SSCs be included within the program boundary?) and
- Does EPU introduce new factors that should be accounted for within the parameters the program is trying to monitor or maintain? (i.e., does the technical basis for the program need to be modified to account for new or different plant or system conditions? Should a new program be created to monitor for new effects that are a result of EPU?)

In order to answer these questions program owners were tasked with evaluating the effects of EPU on their programs. The owners reviewed approximately 72 plant initiatives that are considered programs. Additionally, the review also considered the effects of EPU on the license renewal programs and the programs specifically called out in RS-001. The results of these reviews showed that some programs did indeed require an expansion of scope (e.g. flow accelerated corrosion, fire protection, etc.) but no new programs were required and the existing programs adequately bound the parameters established as a result of EPU.

Changes required to the programs are identified in, and tracked by, the Ginna internal commitment tracking process.

Sections within the Licensing Report in addition to those specified in RS-001

In order to provide a complete description of the analysis performed, the licensing report takes advantage of the provision in RS-001 to add additional sections (additional review areas). The following sections are in addition to the standard template:

1.0	Introduction to the Ginna Station Extended Power Uprate Licensing Report
1.1	Nuclear Steam Supply System Parameters
2.2.6	NSSS Design Transients
2.2.7	Bottom Mounted Instrumentation and Guide Tubes
2.4.2	Plant Operability
2.4.3	Pressurizer Component Sizing
2.5.8.1	Circulating Water System
2.7.7	Other Ventilation Systems (Containment)
2.8.5.0	Non LOCA Analysis Introduction
2.8.7.1	Auxiliary Systems Pumps, Heat Exchangers, Valves and Tanks
2.8.7.2	Natural Circulation Cooldown
2.8.7.3	Loss of Residual Heat Removal at Midloop
2.14	The Effects of EPU on the Renewed Licensing and License Renewal Programs
Appendix A	Safety Evaluation Report Compliance
Appendix B	Additional Codes and Methods
Appendix C	Associated Technical Review Guidance
Appendix D	Acronyms in addition to those in RS-001

Use of Industry Operating Experience

Both the regulators and the nuclear industry peer groups strongly advocate incorporating operating experience and lessons learned as basic input in design, maintenance, operating and licensing activities. The analysis and evaluations performed for the Ginna EPU took full advantage of past EPU experiences by:

- Review of previous power uprate applicant NRC Requests for Additional Information (RAI). All PWR RAIs issued over the past several years were reviewed and, where appropriate, the plant analysis or evaluations relating to the subject were reviewed against the expressed concern and documented to provide reviewer confidence that the issue was appropriately examined. (BWR RAIs were also reviewed when the RAI was related to issues other then those unique to BWRs.)
- NRC Staff interaction. During the development of the engineering analysis and Licensing Report public meetings were held with the Staff. At these meetings the NRC Staff ensured that Ginna was aware of their growing body of lessons learned. When necessary, the Staff also held pubic meetings to provide direct access to various branch technical review experts so they could directly communicate the experience they had gained from previous work.
- Review of Institute of Nuclear Power Operations communications relating to power uprates.

- Review of internal operating experience. During the analysis and evaluation activities careful attention was paid to ensure that system and component operating history was considered. System engineers were interviewed to ensure all pertinent information was available for inclusion in the EPU evaluations. Component maintenance and trouble report histories where reviewed when appropriate to ensure that the changes made as a result of EPU would be factored into the installed capacity of the equipment, with an assessment of the remaining margin.
- Peer reviews. Peer reviews were held both at a project management level and at the functional working level (evaluation and analysis results review). Additionally, a quality assurance performance assessment was performed. This assessment evaluated the EPU development process and verified that quality performance standards were met.
- Executive Oversight Committee. A high level committee was formed to oversee EPU project plans and progress. The committee was comprised of site, Constellation corporate and vendor senior management to assure appropriate resources were brought to bear where necessary. Members of the committee also included industry experts with prior uprate experience and members of the Nuclear Safety Review Board (NSRB) to bring focus to the application of industry operating experience.

EPU Relation To Other Concurrent Licensing Activity

Ongoing Current Licensing Basis Issues

During preparation of the EPU License Amendment Request (LAR) certain licensing issues that are not directly related were actively being evaluated, and their resolutions were not yet decided. These other issues are not now reflected in the Ginna current licensing basis, however they might, as appropriate, later create new aspects of Ginna's licensing basis. Consequently, other than this mention, the EPU LAR is silent on them. Nonetheless, the effect of EPU on each of the issues is being evaluated and will be a consideration in their final resolution.

The active licensing issues are:

Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance"

This issue involves the potential blockage of containment sump screens. Containment sumps are required to perform post-accident by collecting emergency cooling water that has been sprayed/injected into containment from the RCS accumulators and the refueling water storage tank, and by providing the source for continued recirculation of the water once the initial source has been emptied. Potential blockage is due to either 1) foreign material that might be present in containment prior to the accident and be washed to the sump screens by the accident blowdown or by the initial spray/injection in response to the accident, or 2) debris generated as the result of the accident, such as loosened insulation or peeling paint.

Conditions that will change with implementation of EPU will be evaluated to determine if they will either mitigate or exacerbate postulated sump blockage by

modifying the amount of debris available within containment or changing the flow toward the sump, or modifying the sump design configuration.

Fire Protection Circuit Analysis

Like all operating U.S. nuclear power plants, Ginna has a fire protection Safe Shutdown Report which demonstrates that at least one train of redundant safe shutdown equipment remains available following a fire to successfully achieve and maintain safe shutdown conditions

NRC Regulatory Issue Summary RIS 04-003, Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspections, Revision 1, December 29, 2004, addresses the history and current status of this topic. In support of industry efforts, and NRC's resumption of inspections, Ginna is actively reviewing its existing safe shutdown analysis based on guidance.

Certain plant modifications needed for implementation of EPU will involve installation of new circuits, or changing out cables in existing circuits to ones having different specifications (without changing their routing). These new wires/cables have been reviewed for their impact under the EPU project, and, as with every plant modification, will again be addressed independently for impact on the Safe Shutdown Report during the preparation of the specific modification packages.

Hemyc Fire Wrap

The efficacy of Hemyc (brand name) electrical cable raceway fire blanket has recently come into question due to its apparent inability to afford the expected protection during very conservative bounding performance tests. The NRC has issued Information Notice 2005-07 to inform industry of the issues related to this performance and to remind plants of the appropriate methods for establishing compensatory actions, but there are no specific actions required of plants at this time.

Nonetheless, Ginna is investigating its use of Hemyc and is pursuing evaluation of its specific applications and configurations. During this evaluation, the protection of electrical cables that function during events that are evaluated for EPU might coincidently be included. Resolution of the Hemyc fire wrap issue will adequately address all of its applications, including those which affect wiring and equipment evaluated for EPU.

Reliability of Offsite Power Sources

The rolling blackout that occurred in the Northeastern United States and Southeastern Canada on August 14, 2003 has fostered questions about the reliability of offsite AC power sources to nuclear power plants throughout the country. The availability of offsite power is assumed for accident analyses, except those specifically for the loss of offsite power and station blackout. As pointed out in the NRC's Regulatory Issue Summary 2004-05, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power, "although Nuclear Power Plants

are designed to cope with a LOOP [loss of offsite power] event through the use of onsite power supplies, LOOP events are considered to be precursors to SBO [station blackout]. An increase in the frequency or duration of LOOP events increases the risk of core damage."

In the Regulatory Issue Summary the NRC notes the value of licensees fully understanding their LOOP coping strategies, and the impact that plant maintenance activities might have on offsite power availability. By draft Generic Letter 2005-XX (7590-01 (P)) the NRC has proposed to request licensees to submit information regarding loss of offsite power issues.

Ginna EPU might be affected by grid reliability improvements and a better understanding of the plant's interface with the grid only in that the power availability assumptions in accident analyses that were evaluated or performed for the EPU are confirmed. Ginna will continue to participate in and follow industry evaluation of this topic.

Concurrent Ginna License Amendment Requests

At the time of submittal of this License Amendment Request for Extended Power Uprate, there were other active Ginna requests for license amendments before the NRC that had not yet been decided. Some are directly related to EPU, i.e., required for EPU implementation; others are not.

<u>EPU Related LARs</u>

A. Adoption of Relaxed Axial Offset Control
   Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC) dated April 29, 2005

B. Revised Loss of Coolant Accident (LOCA) Analyses-Changes to Accumulator, Refueling Water Storage (RWST) and Administrative Control Technical Specifications
   Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC) dated April 29, 2005

C. Main Feedwater Isolation Valves
   Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC) dated April 29, 2005

D. Fracture Mechanics Analysis per GDC-4
   Letter from J.A. Widay (Ginna) to R.L. Clark (NRC) dated September 30, 2004

E. Application of 10CFR50.90 Process
   Letter from T.A. Marlow (Ginna) to D.M. Skay (NRC) dated May 28, 2005

LARs Not Related To EPU

A. Revision to Diesel Fuel Oil Requirements
   Letter from Mary G. Korsnick (Ginna) to Robert L. Clark (NRC) dated December 20, 2004

B. One-time Integrated Leak Rate Test (ILRT) Interval Extension
   Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC) dated March 10, 2005

Treatment of Proprietary Information referenced within the Licensing Report

Two versions of the LR have been prepared: proprietary and non-proprietary. The non-proprietary version is for placement within the public document room. The proprietary version is for use by the NRC reviewers. The affidavit describing the nature of the information is provided in the EPU license amendment request. The affidavit references WCAP-16461-P, Ginna Station Extended Power Uprate Supplemental Information. This WCAP was the document through which Westinghouse transmitted the information to Ginna. The WCAP itself is not included in the LR. Instead, the information contained within the WCAP is included directly within the appropriate version of the LR. Every effort was made to minimize the amount of information withheld. Bracketed [ ]a,c information designates data that is Westinghouse Proprietary.

Onsite EPU Inspections and Plant Internal Commitment Tracking

In addition to the regulatory commitments detailed in attachment 9 of this license amendment request internal commitments related to EPU are tracked within the plants Commitment Action Tracking System (CATs). Typically, these CATs provide a finer level of detail then the regulatory commitments and are related to specific activities that must be accomplished to achieve EPU. EPU related CATs are flagged against the engineering modification tracking number(s) used for EPU. This process will allow for onsite inspectors to rapidly ascertain the thoroughness, status and impact of the going forward internal commitments related to EPU.

Objectives of Reliability and Risk Improvement

Constellation Generation Group (CGG) established the additional objective for the Ginna EPU to maintain or improve overall plant reliability and risk. This objective will be achieved through the planned plant modifications shown in Table 1.0, above. Examples of modifications planned to improve plant reliability include the condensate booster pump and motor upgrade and main step-up transformer cooler upgrade. Whereas the feasibility evaluation showed that adequate NPSH was provided, without modification, with three of three installed condensate booster pumps operating, the decision was made to replace all three condensate booster pumps and motors to increase their capacity. The higher capacity will support plant operation at the full EPU power with two of three condensate booster pumps operating as in the pre-EPU configuration. Similarly, whereas four of four installed main step-up transformer coolers were adequate without modification, a fifth cooler was installed in order to provide at least one installed spare and maintain an equivalent level of reliability.

Examples of modifications planned to improve overall plant risk include installing air tanks to provide a backup air supply to the charging pumps, relocating the charging pump control power disconnect, and installing local controls for the turbine driven auxiliary feedwater pump discharge motor operated valve. These modifications are not required for EPU implementation. LR section 2.13; Risk Evaluation, documents the risk evaluation of these modifications, as well as procedure enhancements, and concludes that the risk increase associated with EPU will be offset through the implementation of these modifications and the procedure enhancements. CGG is committed to the safe and reliable operation of Ginna Station. The rigorous analysis associated with EPU provides an opportunity use risk reduction insights throughout the engineering and modification process.

Additional Information Provided

To aid the reviewers, in addition to the information contained within this EPU submittal, an additional compact disk is provided. The disk contains the station Updated Final Safety Analysis Report, Plant Technical Specifications, and copies of the submittals previously docketed for which this submittal relies upon for completeness. The additional review information disk contains already publicly available information, is for information only, and is not being docketed as part of this EPU submittal.

1.1 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

The nuclear steam supply system (NSSS) design parameters are the fundamental parameters used as input in all of the NSSS analyses. The current Ginna NSSS design parameters are summarized in Tables 4.4.1 and 5.4.2 of the *Ginna Updated Final Safety Analysis* (UFSAR). The NSSS design parameters provide the primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that serve as the basis for all of the NSSS analyses and evaluations. As a result of the EPU, the Ginna Station NSSS design parameters have been revised as shown in Table 1-1. Table 1-1 provides information for the current design bases conditions as well as for various cases representing operation following the EPU. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the EPU.

1.2 Input Parameters, Assumptions, and Acceptance Criteria

The NSSS design parameters, also referred to as the Performance Capability Working Group (PCWG) parameters, provide the reactor coolant system (RCS) and secondary system conditions (thermal power, temperatures, pressures, and flow) that are used as the basis for the design transients, systems, structures, components, accidents, and fuel analyses and evaluations.

The code used to determine the NSSS design parameters was SGPER (Steam Generator PERformance). There is no explicit NRC approval for the code since it is used to facilitate calculations that could be performed by hand. That is, the code and method used to calculate these values have been successfully used to license all previous similar programs for Westinghouse plants. They use basic thermal-hydraulic calculations, along with first principles of engineering, to generate the temperatures, pressures, and flows shown in Table 1-1.

The major input parameters and assumptions used in the calculation of the four cases of PCWG parameters established for the EPU Program are summarized by the following:

- The parameters are applicable to the existing replacement steam generators.

- With one exception, an uprated NSSS power level of 1817 MWt (1811 MWt core power + 6 MWt RCS net heat input) was assumed for the NSSS analyses. The 1811 MWt core thermal power is a conservatively high value that anticipates achieving up to a 2% measurement uncertainty recapture uprate at a later date. The one exception is the thermal-hydraulic design analysis for the fuel as discussed in LR section 2.8.3. In this analysis a nominal core power of 1775 MWt with a power measurement uncertainty of 2% was used. Upon approval of this license amendment request Ginna Nuclear Power Plant, LLC (Ginna) will operate the Ginna Station at a core thermal power of 1775 MWt (NSSS power level of 1781 MWt, including 6 MWt RCS net heat input) with a power measurement uncertainty of 2%. Note that an uprated NSSS power level of 1781 MWt

(1775 MWt core power + 6 MWt RCS net heat input) was assumed for the BOP analyses.

- A feedwater temperature (T_{feed}) range of 390° to 435°F was selected for the analyses.
- The design core bypass flow was assumed to be 6.5%; this accounts for thimble plugs removed (TPR).
- The current thermal design flow of 85,100 gpm/loop was maintained for the analyses.
- A full-power normal operating T_{avg} range of 564.6° to 576°F was assumed for the analyses.
- Steam generator tube plugging (SGTP) levels of 0% and 10% were assumed.
- The current assumed steam generator fouling factor of 0.00015 hr-ft^2-°F/BTU was maintained.
- A maximum steam generator moisture carryover of 0.10% was utilized.

Acceptance Criteria

The acceptance criteria for determining the NSSS design parameters were that the results of the EPU analyses and evaluations continue to comply with all industry and regulatory requirements applicable to Ginna Station, and that they provide adequate flexibility and margin during plant operation.

1.3 Description of Analyses and Evaluation

Table 1-1 provides the NSSS design parameter cases that were evaluated and serve as the basis for the EPU.

- The current design bases NSSS parameters are provided in the first column.
- EPU Cases 1 and 2 of Table 1-1 are based on a T_{avg} of 564.6°F. Case 2 yielded the minimum secondary side steam generator pressure and temperature since it assumed 10% SGTP. Note that all primary side temperatures were identical for these two cases. This is a result of defining the thermal design flow based on 10% SGTP and conservatively applying it to all cases.
- EPU Cases 3 and 4 of Table 1-1 are based on the T_{avg} of 576.0°F. Case 3 yields the highest secondary side steam pressure performance conditions since it assumed 0% SGTP. Note that all primary side temperatures were identical for these two cases. Note

that for Case 3, for instances where an absolute upper limit steam generator outlet pressure is controlling, steam generator outlet temperature, pressure, and flow are increased above the values in Table 1-1 (see table footnote (b)). The higher values for steam generator outlet temperature, pressure, and flow result from assuming a fouling factor of zero.

Best-estimate station calorimetric measurement based performance predictions were also calculated for the EPU. These calorimetric measurement based calculations were performed to estimate the actual expected steam conditions at the steam generator outlet as opposed to the design conditions shown in Table 1-1.

The calorimetric measurement based calculations used Ginna Station plant measured calorimetric data from cycles 28, 29, 30, and 31 to determine NSSS performance. This measured NSSS performance, particularly for the steam generator, was then used to estimate the steam generator outlet and turbine throttle conditions at EPU conditions.

To achieve the desired turbine throttle pressure of 730 psia, it was determined that an approximate T_{avg} of 572°F – 573°F was required at the uprated NSSS power of 1781 MWt.

A simplified primary heat balance diagram is provided in Figure 1.1-1. This heat balance diagram illustrates the design parameters for Case 3 from Table 1-1.

1.4 NSSS Parameters Conclusions

The resulting PCWG parameters (Table 1-1) were used by Westinghouse as the basis for all the analytical efforts. Westinghouse performed the analyses and evaluations based on the parameter sets that were most limiting, so that the analyses would support operation over the entire range of conditions specified.

Table 1-1 NSSS PCWG Parameters for Ginna Station Uprate Program					
Thermal Design Parameters	**Current[c]**	**EPU**			
		Case 1	**Case 2**	**Case 3**	**Case 4**
NSSS Power					
MWt	1526	1817	1817	1817	1817
10^6 Btu/hr	5,207	6,200	6,200	6,200	6,200
Reactor Power MWt	1520	1811	1811	1811	1811
10^6 Btu/hr	5,186	6,179	6,179	6,179	6,179
Thermal Design Flow, loop gpm	85,100	85,100	85,100	85,100	85,100
Reactor 10^6lb/hr	64.6	65.8	65.8	64.8	64.8
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.5[a]	6.5[a]	6.5[a]	6.5[a]	6.5[a]
Reactor Coolant Temperature, °F					
Core Outlet	607.8	605.5	605.5	616.2	616.2
Vessel Outlet	603.9	601.0	601.0	611.8	611.8
Core Average	576.9	568.8	568.8	580.3	580.3
Vessel Average	573.5	564.6	564.6	576.0	576.0
Vessel/Core Inlet	543.1	528.3	528.3	540.2	540.2
Steam Generator Outlet	541.3	528.0	528.0	539.9	539.9
Steam Generator					
Steam Outlet Temperature, °F	520.9	506.5	503.1	518.8[b]	515.4
Steam Outlet Pressure, psia	819	722	700	804[b]	781
Steam Outlet Flow, 10^6 lb/hr total	6.52	7.42/7.88	7.41/7.87	7.44/7.9[b]	7.43/7.89
Feed Temperature, °F	425	390/435	390/435	390/435	390/435
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10	0.10
Design FF, hr. sq. ft. °F/Btu	0.00015	0.00015	0.00015	0.00015	0.00015
Tube Plugging Level (%)	15	0	10	0	10
Zero Load Temperature, °F	547	547	547	547	547

Table 1-1 (cont.) NSSS PCWG Parameters for Ginna Station Uprate Program	
Hydraulic Design Parameters	
Pump Design Point, Flow (gpm)/Head (ft.)	90,000/252
Mechanical Design Flow, loop gpm	101,200
Minimum Measured Flow, loop gpm	88,650

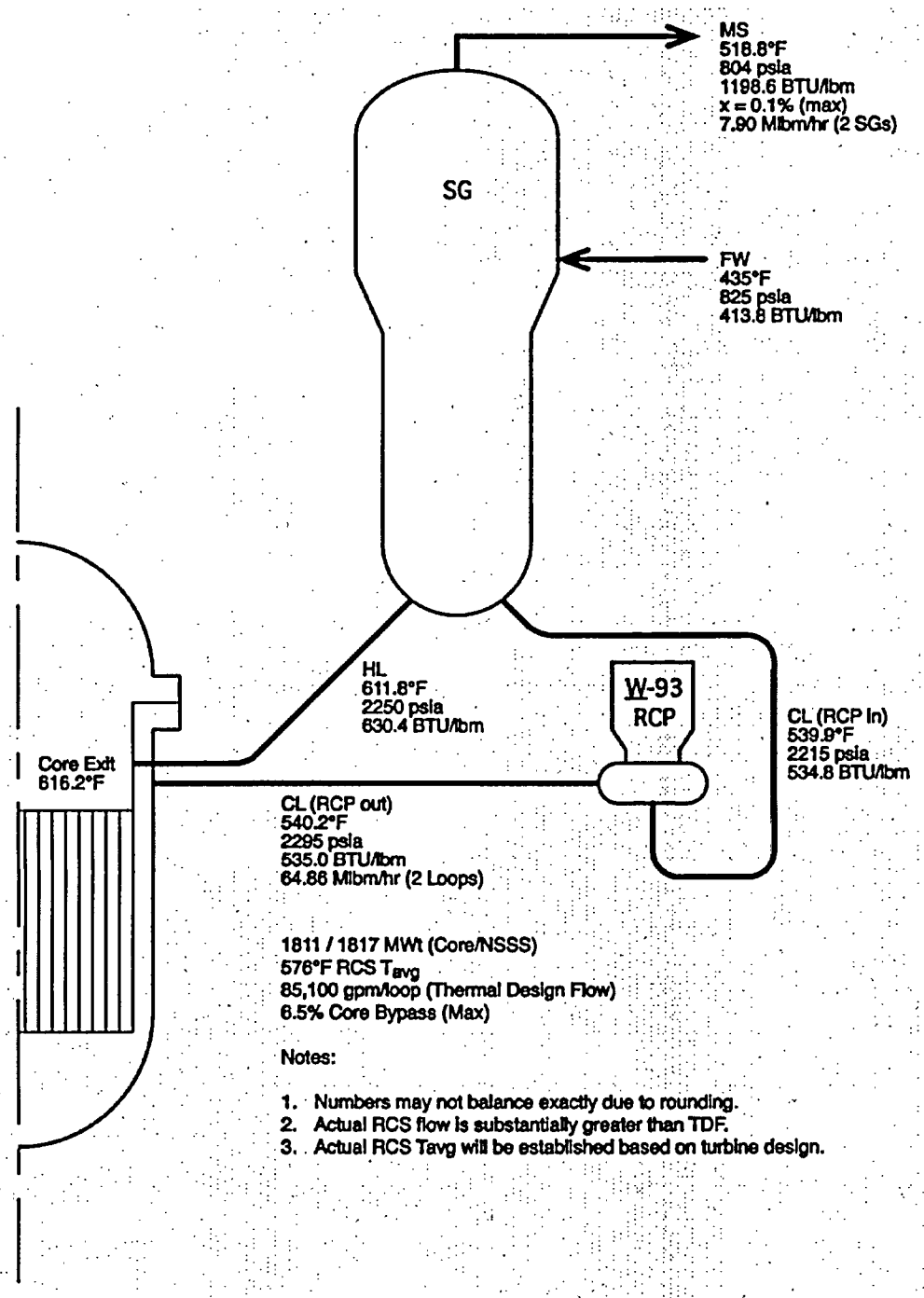
Notes:

a. Core bypass flow includes 2.0% due to thimble plugs removed (TPR).

b. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 855 psia, steam temperature of 525.9°F, and steam flow of 7.92×10^6 lb/hr total should be assumed. This envelopes the possibility that the steam generator could perform better than expected.

c. Current parameters obtained from Tables 4.4.1 and 5.4.2 of UFSAR, or from the most recent NSSS PCWG parameters for fuel upgrade.

Figure 1.1-1
Simplified Primary Heat Balance Diagram



2.0 Evaluation

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

2.1.1.1 Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. Ginna Nuclear Power Plant, LLC's (Ginna) review primarily focused on the effects of the proposed extended power uprate (EPU) on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on:

- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture;

- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized;

- 10CFR50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and,

- 10CFR50.60, which requires compliance with the requirements of 10CFR50, Appendix H.

Specific review criteria are contained in the Standard Review Plan (SRP), Section 5.3.1, and other guidance provided in Matrix 1 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or

closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station reactor vessel material surveillance program relative to conformance to:

- GDC-14, is described in Ginna UFSAR, Section 3.1.2.2.5, "GDC-14, Reactor Coolant Pressure Boundary."

- GDC-31, is described in Ginna UFSAR, Section 3.1.2.4.2, "GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary."

In addition to the evaluations described the UFSAR, the Ginna Station's reactor vessel material surveillance program was evaluated for plant license renewal, which is documented in Section 3.1.2.3.4, "Reactor Vessel Surveillance Program," of NUREG-1786, the Safety Evaluation Report (SER) Related to the License Renewal of R. E. Ginna Nuclear Power Plant, May 2004.

The surveillance program withdrawal schedule is in compliance with ASTM E185 and 10CFR50 Appendix H.

2.1.1.2 Technical Evaluation

2.1.1.2.1 Introduction

The surveillance program is described in Ginna UFSAR Sections 5.1.3.9, "Reactor Coolant Pressure Boundary Surveillance", 5.3.1.3, "Testing and Surveillance" and 5.3.3.2, "Material Surveillance Program." In addition, the License Renewal SER, NUREG-1786, discusses it in Sections 3.1.2.3.4, "Reactor Vessel Surveillance Program," and 4.2, "Reactor Vessel Neutron Embrittlement." The commitments made under the license renewal SER are contained in Commitment 38 in Appendix A to NUREG-1786. The withdrawal schedule itself is now contained in the Ginna "Pressure and Temperature Limits Report" (PTLR), and not in the UFSAR.

Reactor vessel integrity is impacted by any change in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the EPU have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the Ginna surveillance capsule withdrawal schedule contained in the Ginna PTLR.

2.1.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

EPU Fluence Projections

The calculated (projected) fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. These fluence projections for various effective full-power years (EFPYs) are presented in Table 2.1.1-1. Updated surveillance capsule fluence values are also provided in Table 2.1.2-2. Note that capsule fluence values listed in Table 2.1.2-2 are not impacted by the EPU because these capsules have already been removed from the vessel and thus will never be subjected to the higher power. The capsules listed with a fluence of "N/A" will be exposed during operation at EPU conditions, thus new projected removal dates are provided in Table 2.1.1-5.

Typically, vessel fluence values are used to evaluate end-of-license renewal (EOLR) transition temperature shift (EOLR ΔRT_{NDT}) for development of surveillance capsule withdrawal schedules. The calculated fluence projections used in the EPU evaluation complied with Regulatory Guide 1.190. As these calculations were performed on a plant-by-plant basis, there was no generic topical report for the approved method. The methodology used was that of Regulatory Guide 1.190.

Chemistry Factor Values

The chemistry factors (CFs), along with the fluence factor (FF), are used to determine the ΔRT_{NDT}. The CFs used in this evaluation are presented in Table 2.1.1-3 and are documented in WCAP-15885 (Reference 1). Note that WCAP-15885 also contains the copper and nickel weight percent values that were used to calculate the chemistry factor values.

Inlet Temperature

As presented in LR section 1.1, Nuclear Steam Supply System Parameters, the reactor vessel inlet temperature (T_{COLD}) will change with the proposed EPU operating conditions. T_{COLD} changes from 543.1°F to 540.2°F.

Acceptance Criteria

The acceptance criteria for performing material surveillance of the reactor vessel, and for generating a withdrawal schedule are in 10CFR50, Appendix H and ASTM E185-82. A satisfactory number of surveillance capsules remain in the Ginna reactor vessel so that further analysis, such as for life extension, can be completed as necessary.

The acceptance criteria for the T_{COLD} are provided in Regulatory Guide 1.99, Revision 2, which is the basis for 10CFR50.61, and state that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater

embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Thus, the T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of Regulatory Guide 1.99, Revision 2 to remain valid.

2.1.1.2.3 Description of Analyses and Evaluations

The reactor vessel surveillance capsule removal schedule evaluation for the proposed Ginna EPU includes a review of the T_{COLD} to verify that it complies with Regulatory Guide 1.99, Revision 2, and a review of the vessel fluence projections to determine if changes are required as a result of potential changes due to the EPU. This evaluation is consistent with the recommended practices of ASTM E185-82 and meets the requirements of 10CFR50, Appendix H.

A surveillance capsule withdrawal schedule was developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. ASTM E185-82 defines the recommended number of surveillance capsules, and the recommended withdrawal schedule, based on the predicted transition temperature shifts (ΔRT_{NDT}) of the vessel material. The surveillance capsule withdrawal schedule is in terms of EFPYs of plant operation with a design life of 54 EFPYs, as is the case for Ginna. Other factors considered in establishing the surveillance capsule withdrawal schedule were the maximum fluence values at the vessel surface and at the 1/4-thickness (1/4T) location.

The first surveillance capsule is usually scheduled to be withdrawn early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds the expected scatter by a sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of the reactor vessel pressure temperature (PT) operational limits. The withdrawal schedule for the maximum number of surveillance capsules to be withdrawn was adjusted by the lead factor so that:

- The neutron fluence exposure of the second surveillance capsule withdrawn is midway between that of the first and third capsules.
- The exposure of the third surveillance capsule withdrawn does not exceed the peak end-of-life 1/4T fluence.
- The exposure of the fourth surveillance capsule withdrawn does not exceed the peak end-of-life reactor vessel fluence.
- The exposure of the fifth surveillance capsule withdrawn does not exceed twice the peak end-of-life reactor vessel fluence.

Per ASTM E185-82, the four steps used for the development of a surveillance capsule withdrawal schedule are as follows:

- Estimate the peak vessel inside surface fluence at end-of-life and the corresponding transition temperature shift (ΔRT_{NDT}). This identifies the number of capsules required. Per Regulatory Guide 1.99, Revision 2, ΔRT_{NDT} is equal to the chemistry factor times the fluence factor. In the case of determining the number of capsules to be withdrawn, the peak vessel surface fluence is used to determine fluence factor.

- Obtain the lead factor for each surveillance capsule relative to the peak beltline fluence.

- Calculate the EFPYs for the capsule to reach the peak vessel end-of-life fluence at the inside surface and 1/4T locations. These are used to establish the withdrawal schedule for all but the first surveillance capsule.

- Schedule the surveillance capsule withdrawals at the nearest vessel refueling date.

A surveillance capsule withdrawal schedule was developed for the Ginna reactor vessel and documented in the PTLR.

2.1.1.2.4 Reactor Vessel Material Surveillance Program Results

Reactor vessel fluence projections and updated capsule fluence values were generated for the proposed EPU following the guidance of Regulatory Guide 1.190 and are presented in Tables 2.1.1-1 and 2.1.1-2. By comparison, the vessel fluence projections were higher than the vessel fluence projections documented in WCAP-15885 (Reference 1), but lower than the vessel fluence projections documented in WCAP-14684 (Reference 2; incorporated into the Ginna PTLR), which is the fluence analysis basis for the current withdrawal schedule in the Ginna PTLR.

The current surveillance capsule withdrawal schedule for Ginna complies with ASTM E185-82. Per ASTM E185-82, the withdrawal of a capsule is scheduled for the vessel refueling outage nearest to the calculated EFPYs established for the particular surveillance capsule withdrawal.

The removal of capsules from the Ginna reactor vessel to date has met the intent of ASTM E185-82. However, since the revised fluence projections for the proposed EPU are lower than the fluence projections used to develop the current withdrawal schedule, the calculated ΔRT_{NDT} at 54 EFPYs is lower than in the current analysis. The updated calculation of ΔRT_{NDT} is documented in Table 2.1.1-4 and it shows that the maximum ΔRT_{NDT} using the uprated fluence projections for Ginna at 54 EFPYs are greater than 200°F. Per ASTM E185-82, these ΔRT_{NDT} values require five capsules be withdrawn. This quantity is unchanged from the current withdrawal schedule. Thus, the only changes to the current withdrawal schedule are to the updated capsule fluence values, lead factors, and the notes referring to the timing of the future

withdrawals as required for license renewal purposes. The updated withdrawal schedule is documented in Table 2.1.1-5.

As presented in LR section 1.1, Nuclear Steam Supply System Parameters, T_{COLD} is maintained above 525°F and below 590°F. Therefore, the equations and results remain valid without adjustments for temperature affects.

2.1.1.2.5 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

As described in section 2.1.1.2.4 above, the fluence projections for EPU are lower than the vessel fluence projections documented in WCAP-14684 (Reference 2; incorporated into the Ginna PTLR), which is the fluence analysis basis for the current withdrawal schedule in the Ginna PTLR and described in the License Renewal SER, NUREG-1786. The only changes required to the surveillance program are to the updated capsule fluence values, lead factors, and the notes referring to the timing of the future withdrawals as required for license renewal purposes. The updated withdrawal schedule is documented in Table 2.1.1-5 and will supersede the table in the License Renewal SER, NUREG-1786.

2.1.1.2.6 Reactor Vessel Material Surveillance Program Conclusions

An NRC-approved withdrawal schedule exists in the Ginna PTLR that meets the intent of ASTM E185-82 and 10CFR50 Appendix H. Having this withdrawal schedule satisfies 10CFR50.60, GDC-14, GDC-31, and the SRP (see Section 5.3.1).

Ginna has evaluated the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that it has adequately addressed changes in neutron fluence and their effects on the schedule. Ginna further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10CFR50, Appendix H, and 10CFR50.60, and will provide information to ensure continued compliance with GDC-14 and GDC-31 in this respect following implementation of the proposed EPU. Therefore, Ginna finds the EPU acceptable with respect to the reactor vessel material surveillance program.

Finally, since the Reactor Vessel inlet temperature (T_{cold}) is being maintained between 525° and 590°F, the equations and results for predicting ΔRT_{NDT} remain valid without any adjustments.

2.1.1.2.7 Reactor Vessel Material Surveillance Program References

1. WCAP-15885, R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation, July 2002.

2. WCAP-14684, R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation, June 1996.

3. Ginna Design Analysis DA-ME-2003-024, Evaluation of Reactor Vessel Beltline Welds' RT_{NDT} for Pressurized Thermal Shock during Period of Extended Operation, June 10, 2003.

Table 2.1.1-1 Calculated Maximum Neutron Exposure of the Reactor Pressure Vessel				
EFPYs	Azimuthal Location			
	0°	15°	30°	45°
Peak Vessel Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC* 31	2.88E+19	1.82E+19	1.32E+19	1.18E+19
29.0 – EOC 32	2.99E+19	1.89E+19	1.37E+19	1.23E+19
30.4 – EOC 33	3.15E+19	1.99E+19	1.44E+19	1.29E+19
31.8 – EOC 34	3.31E+19	2.08E+19	1.50E+19	1.36E+19
33.2 – EOC 35	3.45E+19	2.18E+19	1.57E+19	1.42E+19
37.0	3.83 E+19	2.42E+19	1.76E+19	1.59E+19
52.0	5.22E+19	3.33E+19	2.43E+19	2.21E+19
54.0	5.42E+19	3.46E+19	2.53E+19	2.30E+19
Intermediate-to-Nozzle-Shell-Girth Weld and Nozzle Shell Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC 31	1.13E+18	7.15E+17	5.17E+17	4.61E+17
29.0 – EOC 32	1.17E+18	7.41E+17	5.36E+17	4.79E+17
30.4 – EOC 33	1.25E+18	7.86E+17	5.67E+17	5.09E+17
31.8 – EOC 34	1.33E+18	8.31E+17	5.99E+17	5.39E+17
33.2 – EOC 35	1.41E+18	8.87E+17	6.40E+17	5.77E+17
37.0	1.64 E+18	1.03E+18	7.50E+17	6.78E+17
52.0	2.47E+18	1.57E+18	1.15E+18	1.05E+18
54.0	2.58E+18	1.65E+18	1.21E+18	1.10E+18

Note:
The future projections account for the EPU from 1520 to 1811 MWt beginning from cycle 33.
* EOC – end-of-cycle

Table 2.1.1-2
Calculated Integrated Neutron Exposure of the Ginna Surveillance Capsules Removed to Date

Capsule & Withdrawal Time	Fluence[a]	Capsule Lead Factor[a,b]
V (@ 1.4 EFPYs)	5.87×10^{18} n/cm^2, (E > 1.0 MeV)	2.96
R (@ 2.6 EFPYs)	1.02×10^{19} n/cm^2, (E > 1.0 MeV)	2.97
T (@ 6.9 EFPYs)	1.69×10^{19} n/cm^2, (E > 1.0 MeV)	1.82
S (@ 17.0 EFPYs)	3.64×10^{19} n/cm^2, (E > 1.0 MeV)	1.79
P (Standby)	N/A[c]	1.91
N (Standby)	N/A[c]	1.81

Notes:

a. These capsule fluence and lead factor values are also documented in WCAP-15885 (Reference 1).

b. Lead factor for capsules remaining in the reactor are based on cycle-specific exposure calculations through fuel cycle 29.

c. Fluence values are not determined until after the capsule is removed from the reactor vessel.

Table 2.1.1-3
Summary of the Ginna Beltline Material CF* Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	CF	
	Position 1.1[a]	Position 2.1[a]
Nozzle Shell Forging 123P118	44.0°F	---
Intermediate Shell Forging 125S255	44.0°F	16.6°F
Lower Shell Forging 125P666	31.0°F	28.3°F
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	167.6°F[b]	180.8°F[c]
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	170.4°F	161.9°F
Ginna Surveillance Weld (Heat #61782)	158.9°F	---

Notes:
a. All CFs are taken from WCAP-15885 (Reference 1).
b. The best-estimate copper and nickel for this heat is different than that used in Ginna Report DA-ME-2003-024 (Reference 3), which used a generic/worst case copper and nickel. This uprate evaluation and WCAP-15885 used a weld-heat-specific copper and nickel best-estimate average, which is also used by Turkey Point.
c. This was determined using Turkey Point Surveillance data and adjusted for Ginna (i.e., based of temperature and chemistry differences).

*CF – chemistry factor

Table 2.1.1-4
ΔRT_{NDT} Values for all Ginna Beltline Materials @ 54 EFPYs

Material	CF	Fluence @ 54 EFPYs[a]	FF[b]*	ΔRT_{NDT}[c]
Nozzle Shell Forging 123P118	44.0°F	0.258×10^{19}	0.632	27.8
Intermediate Shell Forging 125S255	44.0°F	5.42×10^{19}	1.418	62.4
Using Surveillance Capsule Data	16.6°F	5.42×10^{19}	1.418	23.5
Lower Shell Forging 125P666	31.0°F	5.42×10^{19}	1.418	44.0
Using Surveillance Capsule Data	28.3°F	5.42×10^{19}	1.418	40.1
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	167.6°F	0.258×10^{19}	0.632	105.9
Using Surveillance Capsule Data	180.8°F	0.258×10^{19}	0.632	114.3
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	170.4°F	5.42×10^{19}	1.418	241.6
Using Surveillance Capsule Data	161.9°F	5.42×10^{19}	1.418	229.6

Notes:
a. Fluence is at the clad/base metal interface ($\times 10^{19}$ n.cm^2, E > 1.0 MeV).
b. $FF = f^{(0.28 - 0.1 \log f)}$, where f is the clad/base metal interface fluence.
c. Per Regulatory Guide 1.99 Rev. 2 , ΔRT_{NDT} = CF * FF (°F).

*FF – fluence factor

Table 2.1.1-5
Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor[a]	Withdrawal EFPYs[a,b]	Fluence (n/cm^2)[a]
V	77°	2.96	1.4	5.87×10^{18} [c]
R	257°	2.97	2.6	1.02×10^{19} [c]
T	67°	1.82	6.9	1.69×10^{19} [c]
S	57°	1.79	17.0	3.64×10^{19} [c]
N	237°	1.81	29.2[d]	[d]
P	247°	1.91	Standby [e]	[e]

Notes:

a. Updated in EPU dosimetry analysis.
b. EFPYs from plant startup.
c. Plant-specific evaluation.
d. Capsule N will be removed at shortly after receiving a fast neutron fluence equivalent to operation to 2029 (~54 EFPY or 60 year license). The fluence on Capsule N will be between 1 and 2 times the peak end of life fluence
e. Capsule P will be removed shortly following receiving a fast neutron fluence equivalent to operation to 2049. The specific withdrawal EFPY and fluence will be determined following the analysis of Capsule N.

2.1.2 Pressure-Temperature Limits and Upper Shelf Energy

2.1.2.1 Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the reactor coolant pressure boundary during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. Ginna Nuclear Power Plant, LLC's (Ginna) review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full-power years (EFPYs) specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. NRC's acceptance criteria for P-T limits are based on:

- GDC-14, insofar as it requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure

- GDC-31, insofar as it requires that the reactor coolant pressure boundary be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized

- 10CFR50, Appendix G, which specifies fracture toughness requirements for ferritic components of the reactor coolant pressure boundary

- 10CFR50.60, which requires compliance with the requirements of 10CFR50, Appendix G

Specific review criteria are contained in the SRP, Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Ginna Current Licensing Basis

As noted in the *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station reactor vessel material surveillance program relative to conformance to:

- GDC-14, is described in Ginna UFSAR, Section 3.1.2.2.5, "GDC-14, Reactor Coolant Pressure Boundary."

- GDC-31, is described in Ginna UFSAR, Section 3.1.2.4.2, "GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary." As described in this UFSAR section, operating limits during startup and shutdown of the reactor coolant system (RCS) were evaluated using Appendix G of the ASME Code, Section III, and the review of heatup and cooldown curves showed that the pressure and temperature limits in the *Ginna Pressure and Temperature Limits Report* (PTLR) were very conservative. Also, UFSAR, Section 5.3.2 describes how controlling the pressure and temperature limits is a way of ensuring vessel integrity throughout the life of the reactor vessel.

In addition to the evaluations described in the UFSAR, the Ginna Station's pressure and temperature limit curves and upper shelf energy (USE) were evaluated for plant license renewal and documented in Section 4.2, "Reactor Vessel Neutron Embrittlement," of NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R. E. Ginna Nuclear Power Plant, May 2004. Section 4.2.1, "Reactor Vessel Upper-Shelf Energy," provides a description of the reactor vessel upper shelf energy licensing basis, which refers to Framatone Report BAW-2425, Revision 1 (Reference 1). The Ginna pressure and temperature limits are also noted in the SER, Section 4.2.3.

2.1.2.2 Technical Evaluation

2.1.2.2.1 Introduction

The pressure-temperature (P-T) limits are described in Ginna UFSAR Sections 5.1.3.9, Reactor Coolant Pressure Boundary Surveillance" and 5.3.2, "Pressure-Temperature Limits." UFSAR Section 5.1.3.9 also states that the P-T Limits are in accordance with the "Pressure Temperature Limits Report" (PTLR), which were developed using Regulatory Guide 1.99, Revision 2, and Appendix G of Section III of the ASME Boiler and Pressure Vessel Code and are also discussed in the Ginna Technical Specifications, Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits", and Technical Specification Bases Section B 3.4.1, "RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits." Upper shelf energy is discussed in Sections 4.2.1, "Reactor Vessel Upper-Shelf Energy," of NUREG-1786, the License renewal SER.

Reactor vessel integrity is potentially impacted by any change in plant parameters that affect neutron fluence levels or P-T transients. The changes in neutron fluence resulting from the

EPU were evaluated to determine the impact on reactor vessel integrity. The assessment presented herein, in Section 2.1.2, focuses on the Ginna P-T limits and the end-of-license renewal (EOLR) projected values of upper shelf energy .

2.1.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

EPU Fluence Projections

The calculated (projected) fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. These fluence projections for various EFPYs are presented in Table 2.1.2-1. Updated surveillance capsule fluence values are also provided in Table 2.1.2-2. Note that capsule fluence values listed in Table 2.1.2-2 are not impacted by the EPU because these capsules are already removed from the vessel and thus will never be subjected to the higher power. The capsules listed with a fluence of "N/A" will be impacted, thus new projected removal dates are provided in Table 2.1.1-5.

Typically, fluence values are used to calculate the EOLR transition temperature shift (EOLR ΔRT_{NDT}) for development of the P-T limits. The calculated fluence projections used in the EPU evaluation complied with Regulatory Guide 1.190. As these calculations are performed on a plant-by-plant basis, there is no generic topical for their application, however, the methodology used conforms with Regulatory Guide 1.190.

Inlet Temperature

As presented in Section 1.1, Nuclear Steam Supply System Parameters, the reactor vessel inlet temperature (T_{COLD}) will change with the proposed EPU. T_{COLD} changes from 543.1°F to 540.2°F.

Chemistry Factor Values

The chemistry values or weight percent copper were used along with fluence to determine the percent decrease in USE at EOLR. The weight percent copper used in this evaluation is presented in Table 2.1.2-3 for all the beltline materials. It is also documented in WCAP-15885 (Reference 2).

USE

The initial USE values for each beltline material were used as a baseline for determining the EOLR USE. The initial USE values are presented in Table 2.1.2-4.

P-T Limits

The P-T limit curves are presently contained in the PTLR, and are documented in WCAP-14684 (Reference 3). Since these P-T limits will expire prior to the end of licensed period, new P-T

limits will be implemented. In Fact, 52 EFPY P-T limit curves were developed prior to this proposed EPU program assessment. These curves are documented in WCAP-15885 (Reference 2).

Acceptance Criteria

For P-T limit curves, the acceptance criteria are that Ginna have NRC-approved P-T limits developed in accordance with 10CFR50, Appendix G, and that the applicable EFPYs of those P-T limit curves after implementation of the proposed EPU be greater than the current EFPYs of the Ginna reactor vessel.

For USE at EPU conditions, the EOLR USE values for all reactor beltline materials must meet the requirements of 10CFR50 Appendix G, which states the USE must be maintained above 50 ft-lbs, otherwise an equivalent margins analysis (EMA) must be performed to demonstrate that the vessel has adequate margin of safety.

For the vessel inlet temperature, the acceptance criteria are from Regulatory Guide 1.99, Revision 2, which is the basis for 10CFR50.61, and they state that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Thus, the T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of 10CFR50.61 to remain valid. Prior to the EPU T_{COLD} is 543.1°F: it will be 540.2 after EPU.

2.1.2.2.3 Description of Analyses and Evaluations

Applicability of P-T Limits Curves

If the post-EPU reactor vessel fluence projections had exceeded those of the analysis of record, then a new applicability date of the current P-T limit curves would have been calculated using the EPU fluence projections. In addition, a new applicability date would have been calculated for the P-T limit curves in WCAP-15885 (Reference 2), which are planned for implementation prior to or after the EPU has been initiated. The current analysis of record for the P-T limit curves are documented in WCAP-14684 (Reference 3) and contained in the Ginna PTLR. They are applicable to 28 EFPYs. The current PTLR curves will be updated to account for EPU out to 32 EFPYs based on WCAP-14684. The P-T limits in WCAP-15885 are currently applicable to 52 EFPYs.

The calculation was carried out by comparing the fluence projections used in the current calculations of adjusted reference temperature for both the PTLR P-T limit curves and the P-T limit curves in WCAP-15885 (Reference 2), to the EPU vessel fluence projections. If the fluence were lower under EPU conditions, then, conservatively, there would be no change to the applicability date. If the fluence were higher under EPU conditions, then a new date would

have been calculated. This would have been done by determining the peak surface fluence values used in the analysis of record and then determining where they fell, in terms of the EFPY of the Ginna reactor vessel, under the proposed EPU conditions. This calculation is a simple interpolation.

USE

The evaluation to assess the impact of the EPU on the USE was performed in two steps. First, new EOLR USE values were calculated for all reactor vessel beltline materials using the results of the EPU neutron fluence evaluation (Table 2.1.2-1) and Figure 2 of Regulatory Guide 1.99, Revision 2. If any of the materials had produced an EOLR USE below 50 ft-lbs, then an EMA would have been required.

Prior to this EPU evaluation, it was already known that the intermediate-to-lower-shell-girth weld was below 50 ft-lbs at EOLR, so an EMA had already been performed (Reference 1). Thus, the second step of the EPU evaluation for USE was to review the existing EMA for applicability under the EPU conditions. If any other materials dropped below 50 ft-lbs, a new EMA evaluation would be performed, or the existing evaluation would be used to bound the additional low USE material.

2.1.2.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the P-T limits and USE of all the beltline materials. The aging evaluations approved by the NRC in NUREG-1786 for the beltline materials with respect to USE and P-T Limits remain valid for EPU conditions, as discussed below.

2.1.2.2.5 P-T Limits and USE Results

Applicability of Heatup and Cooldown P-T Limit Curves

A review was completed on the current 28 EFPYs P-T limit curve applicability date for Ginna. This review indicated that the revised fluence projections associated with the proposed EPU did not exceed the fluence projections used in developing the current adjusted reference temperature values for Ginna at 28 EFPYs. Thus, the P-T limits contained in the Ginna PTLR remain valid up to 28 EFPYs. Note that these curves will be updated to the 32 EFPY curves from WCAP-14684. There is no reduction in EFPY for these P-T limits.

In addition to the 28 and 32 EFPYs P-T limit curves, a review was completed for Ginna on the 52 EFPYs P-T limit curve applicability date from WCAP-15885 (Reference 2). This review shows that the revised fluence projections associated with the EPU exceed the fluence projections used in developing the adjusted reference temperature values at 52 EFPYs. Thus,

a new applicability date was calculated that shows the 52 EFPYs P-T limit curves from WCAP-15885 to be acceptable to 47.3 EFPYs. Note that no new Adjusted Reference Temperatures (ART) values are required to be calculated; the impact is determined only in terms of EFPY.

USE

Based on the EPU fluence projections, all beltline materials are expected to have a USE greater than 50 ft-lb through EOLR (54 EFPYs), as required by 10CFR50, Appendix G, except the intermediate-to-lower-shell-girth weld (38 lbs/ft) and the intermediate-to-nozzle-shell-girth weld (49.4 lbs/ft). The EOLR (54 EFPYs) USE values, as presented in Table 2.1.2-4, were predicted using the EOLR 1/4-thickness (1/4T) fluence projection.

Of the two limiting beltline materials, the intermediate-to-lower-shell-girth weld was already predicted to drop below 50 lbs/ft prior to this evaluation and consideration of the EPU. Thus, an EMA (Reference 1) showing sufficient margin had already been performed. The difference now was that the fluence used in that EMA was slightly lower than the EPU fluence at the same EFPY. With that in mind, the calculation in BAW-2425, Revision 1, (Reference 1) was redone using the EPU fluence projection.

From BAW-2425, Revision 1 (Reference 1), the J(mean) and J(lower bound) were 853 and 596 lbs/in, respectively, using a 54 EFPYs fluence value of 5.01×10^{19} n/cm^2. The 54 EFPYs fluence value associated with the EPU has been predicted to be 5.42×10^{19} n/cm^2. This changed the J(mean) and J(lower bound) to 849 and 593 lbs/in, respectively. There is sufficient margin when compared to a J-integral limit of 103 lbs/in.

As for the intermediate-nozzle-shell-girth weld, it too fell below the 50 lbs/ft screening criteria by 0.6 ft-lbs. Since the condition of the Ginna vessel (i.e., J-Applied) had already been presented in BAW-2425 (Reference 1), the same calculational methodology performed for the intermediate-to-lower-shell-girth weld could be applied to the intermediate-to-nozzle-shell-girth weld.

As shown in Tables 2.1.2-1 and 2.1.2-3, both the fluence and copper content for the intermediate-to-nozzle-shell girth weld are lower than that used in BAW-2425 (Reference 1) for the intermediate-to-lower-shell-girth weld. Since all other parameters and equations remained unchanged, it is concluded that the corresponding J(mean) and J(lower bound) are higher than that calculated for the intermediate-to-lower-shell-girth weld under EPU conditions. Thus the margin is greater for the intermediate- to-nozzle-shell-girth weld.

Inlet Temperature

The T_{COLD} will be maintained above 525°F and below 590°F (see LR section 1.1, Nuclear Steam Supply System Parameters).

2.1.2.2.6 P-T Limits and USE Conclusions

The fluence projections under the EPU condition, while considering actual Ginna power distributions incorporated to date, exceed the fluence projection used in the most recent USE evaluation for the limiting material (i.e., the intermediate-to-lower-shell-girth weld), and the fluence used in the adjusted reference temperature calculation for 52 EFPYs, but do not exceed the fluence projections used for the adjusted reference temperature calculation for 28 and 32 EFPYs (As documented in WCAP-14684). The effect of the higher fluence projections is minimal for both the EOLR 52 EFPYs P-T curves (reduced to 47.3 EFPYs) and the EOLR USE. In addition, the EPU did not alter any license renewal/aging commitments with respect to the P-T limits or USE. Lastly, since the inlet temperature is to be maintained between 525° and 590°F, the equations for calculating the adjusted reference temperatures and predicting USE decrease remain valid without any adjustments.

Based on the results presented in Section 2.1.2.2.5, "P-T Limits and USE Results,", for operation at the proposed EPU conditions, the requirements of 10CFR50, Appendix G are met, satisfying the requirements for 10CFR50.60, which meets the Ginna Station current licensing basis with respect to the NRC SRP, GDC-14, and GDC-31.

2.1.2.2.7 P-T Limits and USE References

1. Framatome ANP Report BAW-2425, Rev. 1, *Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of R. E. Ginna for Extended Life through 54 Effective Full Power Years*, H. P. Gunawardane, June 2002.

2. WCAP-15885, *R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation*, July 2002.

3. WCAP-14684, *R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation*, June 1996.

4. Ginna Design Analysis DA-ME-2003-024, *Evaluation of Reactor Vessel Beltline Welds' RT_{NDT} for Pressurized Thermal Shock during Period of Extended Operation*, June 10, 2003

Table 2.1.2-1 Calculated Maximum Neutron Exposure of the Reactor Pressure Vessel				
EFPYs	Azimuthal Location			
	0°	15°	30°	45°
Peak Vessel Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC* 31	2.88E+19	1.82E+19	1.32E+19	1.18E+19
29.0 – EOC 32	2.99E+19	1.89E+19	1.37E+19	1.23E+19
30.4 – EOC 33	3.15E+19	1.99E+19	1.44E+19	1.29E+19
31.8 – EOC 34	3.31E+19	2.08E+19	1.50E+19	1.36E+19
33.2 – EOC 35	3.45E+19	2.18E+19	1.57E+19	1.42E+19
37.0	3.83 E+19	2.42E+19	1.76E+19	1.59E+19
52.0	5.22E+19	3.33E+19	2.43E+19	2.21E+19
54.0	5.42E+19	3.46E+19	2.53E+19	2.30E+19
Intermediate-to-Nozzle-Shell-Girth Weld and Nozzle Shell Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC 31	1.13E+18	7.15E+17	5.17E+17	4.61E+17
29.0 – EOC 32	1.17E+18	7.41E+17	5.36E+17	4.79E+17
30.4 – EOC 33	1.25E+18	7.86E+17	5.67E+17	5.09E+17
31.8 – EOC 34	1.33E+18	8.31E+17	5.99E+17	5.39E+17
33.2 – EOC 35	1.41E+18	8.87E+17	6.40E+17	5.77E+17
37.0	1.64 E+18	1.03E+18	7.50E+17	6.78E+17
52.0	2.47E+18	1.57E+18	1.15E+18	1.05E+18
54.0	2.58E+18	1.65E+18	1.21E+18	1.10E+18

Note:
The future projections account for the EPU from 1520 MWt to 1811 MWt beginning from cycle 33.
* EOC – end-of-cycle

Table 2.1.2-2
Calculated Integrated Neutron Exposure of the Ginna Surveillance Capsules Removed to Date

Capsule & Withdrawal Time	Fluence[a]	Capsule Lead Factor[a,b]
V (@ 1.4 EFPYs)	5.87×10^{18} n/cm^2, (E > 1.0 MeV)	2.96
R (@ 2.6 EFPYs)	1.02×10^{19} n/cm^2, (E > 1.0 MeV)	2.97
T (@ 6.9 EFPYs)	1.69×10^{19} n/cm^2, (E > 1.0 MeV)	1.82
S (@ 17.0 EFPYs)	3.64×10^{19} n/cm^2, (E > 1.0 MeV)	1.79
P (Standby)	N/A[c]	1.91
N (Standby)	N/A[c]	1.81

Notes:

a. These capsule fluence and lead factor values are also documented in WCAP-15885 (Reference 2).

b. Lead factor for capsules remaining in the reactor are based on cycle-specific exposure calculations through fuel cycle 29.

c. Fluence values are not determined until after the capsule is removed from the reactor vessel.

Table 2.1.2-3
Ginna Reactor Vessel Toughness Properties[a]

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}
Nozzle Shell Forging 123P118	0.07	0.68	30°F
Intermediate Shell Forging 125S255	0.07	0.69	20°F
Lower Shell Forging 125P666	0.05	0.69	40°F
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	0.23[b]	0.59[b]	10°F
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	0.25	0.56	-4.8°F
Ginna Surveillance Weld (Heat # 61782)	0.23	0.53	---

Notes:

a. All data were taken from WCAP-15885, Rev. 0 (Reference 2).

b. The best-estimate copper and nickel for this heat is different than that used in Ginna Report DA-ME-2003-024 (Reference 4), which used a generic/worst case copper and nickel. This EPU evaluation, and WCAP-15885 used a weld-heat-specific copper and nickel best-estimate average, which is also used by Turkey Point. The initial RT_{NDT} used herein is also a heat-specific measure value, while DA-ME-2003-024 used a generic value. Given that the values used here are actual measured values, they are more applicable than the generic, and are thus acceptable.

Table 2.1.2-4
Predicted End-of-License Extension EOLR (54 EFPYs) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOLR Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lbs)	Projected USE Decrease (%)	Projected EOLR USE (ft-lbs)
Nozzle Shell Forging 123P118	0.07	0.175	117	13	102
Intermediate Shell Forging 125S255	0.07	3.67	91	26	67
Lower Shell Forging 125P666	0.05	3.67	114	26	84
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	0.23	0.175	65[a]	24	49.4
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	0.25	3.67	80	52	38

Notes:
a. Turkey Point initial USE for same weld heat.

2.1.3 Pressurized Thermal Shock

2.1.3.1 Regulatory Evaluation

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to ensure that adequate fracture toughness is provided for supporting reactor operation. Ginna Nuclear Power Plant, LLC's (Ginna) review covered the PTS methodology and the calculations for the reference temperature, (RT_{PTS}), at the expiration of the license, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS are based on:

1. GDC-14, insofar as it requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture

2. GDC-31, insofar as it requires that the reactor coolant pressure boundary be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized

3. 10CFR50.61, insofar as it sets fracture toughness criteria for protection against PTS events

Specific review criteria are contained in the SRP, Section 5.3.2, and other guidance provided in Matrix 1 of RS-001.

Ginna Current Licensing Basis

As noted in *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station reactor vessel material surveillance program relative to conformance to:

- GDC-14, is described in Ginna UFSAR, Section 3.1.2.2.5, "GDC-14, Reactor Coolant Pressure Boundary."

- GDC-31, is described in Ginna UFSAR, Section 3.1.2.4.2, "GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary." As described in this UFSAR section, information on pressurized thermal shock is provided in Section 5.3.3 of the UFSAR.

In addition to the evaluations described in the UFSAR, the Ginna Station's continued compliance with 10CFR50.61 was evaluated for plant license renewal in Section 4.2.2, "Pressurized Thermal Shock," of NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant.

In its license renewal submittal, Ginna provided RT_{PTS} analyses, including the chemistry factor and the predicted RT_{PTS} value, through 54 EFPYs for each forging and weld in the Ginna RV beltline. To verify the predicted RT_{PTS} value, the staff requested a description of the analysis performed to determine the neutron fluence. Ginna's letter dated April 11, 2003 provided WCAP-15885 (Reference 1) that documents the methodology and validation of the results as compared to surveillance capsule measured data. The staff concluded that the methodology generally complies with the guidance in RG 1.190, and that the validation for the Ginna 32 and 54 EFPY fluence values are acceptable for determining the impact of neutron fluence on RV materials in the beltline region.

This methodology, however, has not been qualified for calculations above or below the beltline region. For regions above or below the core, the expected uncertainties would be higher than those in the beltline region, but no greater than a factor of 2. As a result of extending the license, Ginna determined that the weld between the intermediate shell and the nozzle shell would receive a neutron fluence greater than 10^{18} n/cm2 (E>1 MeV). This weld is 10 inches above the top of the core. Because the methodology has not been qualified for determining the neutron fluence for this weld, the staff requested that Ginna perform a RT_{PTS} calculation using a factor of 2 increase in the neutron fluence. This analysis is contained in Attachment 4 to Ginna's letter dated June 10, 2003. In the RT_{PTS} calculation described therein, it was demonstrated that this weld will not be limiting throughout the period of extended operation.

In its June 10, 2003 letter, Ginna changed its method for determining the RT_{PTS} value for the limiting weld, to one based on the use of Ginna surveillance data. 10CFR50.61 identifies two methods of determining the chemistry factor and RT_{PTS} value—one based on the amount of copper and nickel in the weld and one based on the use of credible surveillance data. When the surveillance data are deemed credible, they must be used to determine the material-specific chemistry factor. Ginna chose to use surveillance data in determining the chemistry factor, but needed to demonstrate the credibility of the data. The chemistry factor identified in the June 10 letter is 161.9 °F, and the chemistry factor identified for this weld in the Reactor Vessel Integrity Database (RVID) is 158.7 °F, which is based on the surveillance data. Although the difference in the chemistry factor calculated by Ginna and that in the RVID is small, the staff requested to

review the surveillance data and methodology utilized to determine the chemistry factor and to confirm that the results satisfy 10CFR50.61. The staff further noted that this analysis differed from that identified in UFSAR Section A3.1.2, so Ginna was also requested to provide an update to this UFSAR section. (This was Open Item 4.2.2-1 of Ginna's License Renewal Application).

In Attachment 3 to its letter dated December 9, 2003, Ginna provided the surveillance data that were contained in Section 2 of WCAP-15885 (Reference 1). The neutron fluence for each capsule had been revised in accordance with the methodology in RG 1.190, and the surveillance weld chemistry had been revised based on additional chemical analysis performed on irradiated samples. Attachment 4 to that letter provided an evaluation of the surveillance data. The staff confirmed that the scatter for the surveillance weld metal, which is the limiting material for the Ginna reactor vessel, is within 28 °F of the best fit line through the surveillance data, in accordance with the methodology in RG 1.99, Revision 2, and agreed that the surveillance data generally satisfy the criteria in paragraph (c)(2)(I) of 10CFR50.61, and thus were credible. In addition, the staff agreed that the RT_{PTS} value at the expiration of the extended license is expected to be 271 °F for the intermediate shell to the lower shell weld, which is the limiting material in the Ginna RV. And because the RT_{PTS} value for the intermediate shell to lower shell weld is less than 300 °F, the Ginna RV is adequately protected against PTS events. The required future update to the UFSAR documented in Ginna's letter dated December 9, 2003, provides an adequate description of the analysis performed for PTS, resolving Open Item 4.2.2-1 for the License Extension Application.

2.1.3.2 Technical Evaluation

2.1.3.2.1 Introduction

Pressurized Thermal Shock is discussed in Ginna UFSAR Sections 5.3.3.5, "Pressurized Thermal Shock" and Section 4.2.2, "Pressurized Thermal Shock," of NUREG-1786, the SER.

Reactor vessel integrity is potentially impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the proposed EPU have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the end-of-license renewal (EOLR) PTS evaluation.

2.1.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

EPU Fluence Projections

The calculated (projected) fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. These fluence projections for various effective full-power years (EFPYs) are presented in Table 2.1.3-1. Updated surveillance capsule fluence

values are also provided in Table 2.1.3-2. Note that capsule fluence values listed in Table 2.1.3-2 are not impacted by the EPU because these capsules are already removed from the vessel and thus will never be subjected to the higher power. The capsules listed with a fluence of N/A will be impacted, thus new removal dates must be calculated.

Typically, fluence values are used to calculate the EOLR transition temperature shift (EOLR RT_{PTS}) in the PTS equation from 10CFR50.61. The calculated fluence projections used in the EPU evaluation complied with Regulatory Guide 1.190. As these calculations are performed on a plant-by-plant basis, there is no generic topical for their application, however, the methodology used conforms to Regulatory Guide 1.190.

Chemistry Factor Values

The chemistry factors (CFs) and the fluence factor (FF) are used to determine the ΔRT_{PTS}. The CFs used in this evaluation are presented in Table 2.1.3-3 and are documented in WCAP-15885 (Reference 1). Note that WCAP-15885 also contains the copper and nickel weight percent values that were used to calculate the chemistry factor values.

Initial Reference Temperature, Nil-Ductility Temperature (RT_{NDT})

The initial RT_{NDT} values are the baseline reference temperature for each material and were used to determine the EOLR RT_{PTS} along with the ΔRT_{PTS} and margin. The initial RT_{NDT} values used in this evaluation are presented in Table 2.1.3-3 and are also documented in WCAP-15885 (Reference 1).

Inlet Temperature

As presented in Section 1.1, "Nuclear Steam Supply System Parameters," herein, the reactor vessel inlet temperature (T_{COLD}) will change with the proposed EPU. T_{COLD} changes from 543.1°F to 540.2°F.

Acceptance Criteria

The EPU RT_{PTS} values for all beltline materials must not exceed the screening criteria of the PTS Rule for EOLR. Specifically, the RT_{PTS} values of the base metal (plates or forgings) and longitudinal welds must not exceed 270°F, while the girth weld metal RT_{PTS} values must not exceed 300°F through the EOLR of 54 EFPYs.

For T_{COLD}, the acceptance criteria are provided in Regulatory Guide 1.99, Revision 2, which is the basis for 10CFR50.61, and state that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement."

Thus, T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of 10CFR50.61 to remain valid.

2.1.3.2.3 Description of Analyses and Evaluations

The limiting condition on reactor vessel integrity known as PTS can occur during a severe system transient such as a loss-of-coolant accident (LOCA) or a steam line break. Such transients can challenge the integrity of a reactor vessel under the following conditions:

- Severe overcooling of the inside surface of the vessel wall followed by high repressurization
- Significant degradation of vessel material toughness caused by radiation embrittlement
- Presence of a critical-size defect in the vessel wall

The PTS concern arises if one of these transients should act on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event could produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the NRC issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-life plant operation. All PWR vessels in the U.S. have been required to evaluate vessel embrittlement in accordance with the criteria through end of life.

The NRC subsequently amended its regulations for light water reactors (LWRs) changing the procedure for calculating radiation embrittlement. The revised PTS rule was published in the Federal Register, December 19, 1995, with an effective date of January 18, 1996. This amendment made the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The PTS rule establishes the following requirements for all domestic, operating PWRs:

- For each PWR that has had an operating license issued, the licensee will have projected values of RT_{PTS} accepted by the NRC, for each reactor vessel beltline material for the end-of-life fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any license renewal term, if applicable for the plant.

- The RT_{PTS} screening criteria values for the beltline region are:

  - 270°F for plates, forgings, and axial weld materials
  - 300°F for circumferential weld materials

- RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the end-of-life fluence for the material. Equation 1 is used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \qquad \text{Equation 1}$$

Where,

$RT_{NDT(U)}$ = Reference temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M was evaluated from the Equation 2.

$$M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2} \qquad \text{Equation 2}$$

σ_i is the standard deviation for $RT_{NDT(U)}$.

σ_i = 0°F when $RT_{NDT(U)}$ is a measured value
σ_i = 17°F when $RT_{NDT(U)}$ is a generic value

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings: (σ_Δ not to exceed one half of ΔRT_{NDT})

σ_Δ = 17°F when surveillance capsule data are not used
σ_Δ = 8.5°F when surveillance capsule data are used

For welds: (σ_Δ not to exceed one half of ΔRT_{NDT})

σ_Δ = 28°F when surveillance capsule data are not used
σ_Δ = 14°F when surveillance capsule data are used

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradiation, and was calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \qquad \text{Equation 3}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF was determined from Tables 1 and 2 of the PTS Rule. Surveillance data deemed credible is used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The 54 EFPYs EOLR fluence projections were used in calculating the Ginna-specific RT_{PTS}.

Equation 4 is used for determining RT_{PTS} using Equation 3 with end-of-life fluence values for determining RT_{PTS}

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \qquad \text{Equation 4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, plant-specific information that could affect the level of embrittlement is considered. For the Ginna calculation this information included, but was not limited to, the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program are integrated into the RT_{NDT} estimate if the plant-specific surveillance data is deemed credible. Material-specific values of CF for surveillance materials are determined from Equation 5.

$$CF = \frac{\sum[A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum[f_i^{(0.56-0.20\log f_i)}]} \qquad \text{Equation 5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differed from the vessel weld, i.e., differed from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} would be adjusted for differences in copper and nickel content by multiplying them by the ratio of the CF for the vessel material to that for the surveillance weld.

RT_{PTS} values were calculated in the Ginna Design Analysis, DA-ME-2003-024 (Reference 2), for the two limiting beltline material girth welds.

2.1.3.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for PTS analysis of all the beltline materials. The aging evaluations approved by the NRC in NUREG-1786 for the beltline materials with respect to PTS remain valid for EPU conditions, as described below.

2.1.3.2.5 PTS Results

An evaluation of the impact of the EPU on PTS was performed for Ginna. Given implementation of the proposed EPU, from Table 2.1.3-1 it is noted that the neutron fluence projections for Ginna have increased versus those calculated fluence projections reported in WCAP-15885 (Reference 1) and in DA-ME-2003-024 (Reference 2). PTS calculations were performed for all the beltline materials of the Ginna reactor vessel under EPU conditions using the rules from 10CFR50.61. The results of these calculations are presented in Table 2.1.3.4. Note that the limiting material for Ginna at a license renewal time of 54 EFPY is the intermediate to lower shell girth weld. At this EFPY the Ginna licensing basis RT_{PTS} value, using surveillance capsule data, is 273°F. Based on these results, all RT_{PTS} values remained below the NRC screening criteria values using the projected EPU fluence projections through the EOLR, 54 EFPYs.

2.1.3.2.6 PTS Conclusions

The fluence projections under the proposed EPU conditions, while considering actual power distributions incorporated to date, exceed the fluence projections used in the latest PTS evaluation for the Ginna reactor vessel for license renewal. The effect of the higher fluence projections is minimal for PTS, raising the PTS value for the limiting material from 270.6°F to 273°F, a value below the 300°F allowable. Since T_{COLD} is to be maintained between 525° and 590°F, the equations for calculating RT_{PTS} remain valid without any adjustments.

Ginna has reviewed the evaluation of the effects of the proposed EPU on the PTS for the plant and concludes that it has adequately addressed changes in neutron fluence and their effects on PTS. Ginna further concludes that the evaluation has demonstrated that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-14, GDC-31, and 10CFR50.61, following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU acceptable with respect to PTS.

2.1.3.2.7 PTS References

1. WCAP-15885, *R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation*, July 2002.

2. DA-ME-2003-024, *Evaluation of Reactor Vessel Beltline Welds' RT_{NDT} for Pressurized Thermal Shock during Period of Extended Operation*, June 10, 2003.

Table 2.1.3-1 Calculated Maximum Neutron Exposure of the Reactor Pressure Vessel				
EFPYs	Azimuthal Location			
	0°	15°	30°	45°
Peak Vessel Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC* 31	2.88E+19	1.82E+19	1.32E+19	1.18E+19
29.0 – EOC 32	2.99E+19	1.89E+19	1.37E+19	1.23E+19
30.4 – EOC 33	3.15E+19	1.99E+19	1.44E+19	1.29E+19
31.8 – EOC 34	3.31E+19	2.08E+19	1.50E+19	1.36E+19
33.2 – EOC 35	3.45E+19	2.18E+19	1.57E+19	1.42E+19
37.0	3.83 E+19	2.42E+19	1.76E+19	1.59E+19
52.0	5.22E+19	3.33E+19	2.43E+19	2.21E+19
54.0	5.42E+19	3.46E+19	2.53E+19	2.30E+19
Intermediate-to-Nozzle-Shell-Girth Weld and Nozzle Shell Fluence @ the Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]				
27.6 – EOC 31	1.13E+18	7.15E+17	5.17E+17	4.61E+17
29.0 – EOC 32	1.17E+18	7.41E+17	5.36E+17	4.79E+17
30.4 – EOC 33	1.25E+18	7.86E+17	5.67E+17	5.09E+17
31.8 – EOC 34	1.33E+18	8.31E+17	5.99E+17	5.39E+17
33.2 – EOC 35	1.41E+18	8.87E+17	6.40E+17	5.77E+17
37.0	1.64 E+18	1.03E+18	7.50E+17	6.78E+17
52.0	2.47E+18	1.57E+18	1.15E+18	1.05E+18
54.0	2.58E+18	1.65E+18	1.21E+18	1.10E+18

Note: The future projections account for the EPU from 1520 to 1811 MWt beginning from cycle 33.

*EOC – end-of-cycle

Table 2.1.3-2
Calculated Integrated Neutron Exposure of the Ginna Surveillance Capsules Removed to Date

Capsule & Withdrawal Time	Fluence[a]	Capsule Lead Factor[a,b]
V (@ 1.4 EFPYs)	5.87×10^{18} n/cm^2, (E > 1.0 MeV)	2.96
R (@ 2.6 EFPYs)	1.02×10^{19} n/cm^2, (E > 1.0 MeV)	2.97
T (@ 6.9 EFPYs)	1.69×10^{19} n/cm^2, (E > 1.0 MeV)	1.82
S (@ 17.0 EFPYs)	3.64×10^{19} n/cm^2, (E > 1.0 MeV)	1.79
P (Standby)	N/A[c]	1.91
N (Standby)	N/A[c]	1.81

Notes:

a. These capsule fluence and lead factor values are also documented in WCAP-15885 (Reference 1).

b. Lead factor for capsules remaining in the reactor are based on cycle-specific exposure calculations through fuel cycle 29.

c. Fluence values are not determined until after the capsule is removed from the reactor vessel.

Table 2.1.3-3
Summary of the Ginna Beltline Material CF* Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	CF		Initial RT_{NDT}
	Position 1.1[a]	Position 2.1[a]	
Nozzle Shell Forging 123P118	44.0°F	---	30°F
Intermediate Shell Forging 125S255	44.0°F	16.6°F	20°F
Lower Shell Forging 125P666	31.0°F	28.3°F	40°F
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	167.6°F[b]	180.8°F[c]	10°F[b]
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	170.4°F	161.9°F	-4.8°F
Ginna Surveillance Weld (Heat # 61782)	158.9°F	---	---

Notes:

a. All CFs are taken from WCAP-15885 (Reference 1).

b. The best-estimate copper and nickel for this heat is different than that used in Ginna Design Report DA-ME-2003-024 (Reference 2), which used a generic/worst-case copper and nickel. This EPU evaluation and WCAP-15885 used a weld-heat-specific copper and nickel best-estimate average, which is also used by Turkey Point for the same heat of weld. The initial RT_{NDT} used herein is also a heat-specific measure value, while DA-ME-2003-024 used a generic number. Given that the values used here are actual measured values, they are more applicable than the generic and thus acceptable.

c. This was determined using Turkey Point surveillance data and adjusted for Ginna (i.e., based on temperature and chemistry differences). See WCAP-15885 for the calculation.

*CF – chemistry factor

Table 2.1.3-4
RT_{PTS} Calculations for Ginna Beltline Region Materials at 54 EFPYs (EOLR)

Material	Fluence, f (n/cm^2, E>1.0 MeV)[a]	FF[b]	CF (°F)[c]	ΔRT_{PTS} (°F)[d]	Margin (°F)[e]	$RT_{NDT(U)}$ (°F)[f]	RT_{PTS} (°F)[g]
Nozzle Shell Forging 123P118	0.258×10^{19}	0.632	44.0	27.8	27.8	30	86[k]
Intermediate Shell Forging 125S255	5.42×10^{19}	1.418	44.0	62.4	34	20	116
Using Surveillance Capsule Data	5.42×10^{19}	1.418	16.6	23.5	34[h]	20	78
Lower Shell Forging 125P666	5.42×10^{19}	1.418	31.0	44.0	34	40	118
Using Surveillance Capsule Data	5.42×10^{19}	1.418	28.3	40.1	17	40	97
Nozzle-to-Intermediate-Shell-Girth Weld (Heat # 71249)	0.258×10^{19}	0.632	167.6[j]	105.9	56	10[j]	172[k]
Using Surveillance Capsule Data	0.258×10^{19}	0.632	180.8[j]	114.3	56[h]	10	180[k]
Intermediate-Shell-to-Lower-Shell-Girth Weld (Heat # 61782)	5.42×10^{19}	1.418	170.4	241.6	56	-4.8	293
Using Surveillance Capsule Data	5.42×10^{19}	1.418	161.9	229.6	48.3[i]	-4.8	273[L]

Notes:

a. The fluence, f, was taken from the peak azimuthal location, see Table 2.1.3-1.

b. $FF = f^{(.28 - 0.1*\log f)}$, where f is the clad/base metal interface fluence.

c. CF is taken from Table 2.1.3-2.

d. $\Delta RT_{PTS} = CF * FF$.

e. Margin = $2*(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$.

f. Initial RT_{NDT} values are measured values, see Table 2.1.3-3.

g. $RT_{PTS} = RT_{NDT(u)} + \Delta RT_{PTS}$ + Margin (°F).

h. Data is "not credible."

i. Based on additional test by Babcock and Wilcox and documented in the Ginna PTLR*.

j. Position 1.1 CF and initial RT_{NDT} differs from Ginna PTS Report DA-ME-2003-024 (Reference 2) (see Table 2.1.3-3 for explanation). The Position 2.1 CF was added to

this evaluation versus DA-ME-2003-024 because it was documented in WCAP-15885 (Reference 7). This is conservative since it is higher and the data are not credible (i.e., higher margin).

k. If the fluence value for the Nozzle area was doubled, as suggested in DA-ME-2003-024, then the RT_{PTS} values for the Nozzle Forging and Nozzle Forging to Intermediate Shell Forging Girth Weld would be 100°F and 213°F (Position 2.1), respectively. The Position 1.1 RT_{PTS} value for the Nozzle Forging to Intermediate Shell Forging Girth weld would be 203°F.

l. Ginna Licensing Basis.

*PTLR – Pressure and Temperature Limits Report

2.1.4 Reactor Internals and Core Support Materials

2.1.4.1 Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components that perform safety functions or whose failure could affect safety functions performed by other structures, systems, and components. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system). The Ginna Nuclear Power Plant, LLC (Ginna) review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internals and core support materials are based on GDC-1 and 10CFR50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277 and BAW-2248.

Ginna Current Licensing Basis

The Ginna Station reactor vessel internals (RVIs) consists of two basic assemblies:

1. Upper internals assembly that is removed during each refueling operation to obtain access to the reactor core. The top of this assembly is clamped to a ledge below the vessel-head mating surface by the reactor vessel head. The core barrel fuel alignment pins of the lower internals assembly guide the bottom of the upper internals assembly.

2. Lower internals assembly that can be removed, if desired, following a complete core unload. This assembly is clamped at the same ledge below the vessel head mating surface.

Additional details of the RVIs are provided in Sections 3.9.5 and 4.2.1 of the Ginna UFSAR.

The RVI components under consideration include lower core plate and fuel pins; lower support forging and columns; core barrel and flange; radial keys and clevis inserts; baffle and former assembly; core barrel outlet nozzle; secondary core support; diffuser plates; upper support plate assembly; upper core plate and fuel alignment pins; upper support columns; rod cluster control assembly (RCCA) guide tubes and flow downcomers; guide tube support pins; upper core plate alignment pins; hold-down spring; head/vessel alignment pins; thermal shield; BMI columns and flux thimbles; head cooling spray nozzles; upper instrumentation column, conduit, and supports; and bolting (for upper support column, guide tube, clevis insert, lower support column, and baffle/former assembly).

The intended functions of the reactor vessel internals and core supports are core support, flow distribution, guidance and support of RCCAs, vessel shielding, guidance and support of instrumentation, and guidance and support of thermocouples.

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), Section 3.1, the GDC general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970's the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station reactor internals and core support materials relative to conformance to:

- 10CFR50.55a(a)(1) is described in UFSAR section 3.2.1, "Classification of Structures, Components, and Systems – Introduction." As part of SEP Topic III-1, the original codes and standards used in the design of the Ginna Station were compared with later licensing criteria, including 10CFR50.55a. The objective was to assess the capability of Ginna Station SSCs to perform their safety functions as judged by the later standards. Although several areas were identified where requirements had changed, all areas were satisfactorily resolved, and SEP Topic III-1 was closed.

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1. General Design Criterion 1 - Quality Standards and Records, wherein it is noted that all structures, systems, and components (SSCs) of the facility were classified according to their importance. The classification of structures and equipment is discussed in Section 3.2. SSCs were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. The quality control and quality assurance program for Ginna Station construction, and the current quality assurance program, under which systems were installed and are maintained, are described in UFSAR Section 17.1, and 17.2, respectively.

- GDC-10 is described in Ginna UFSAR section 3.1.1.2.1 which states that the reactor core design, in combination with coolant, control, and protection systems, provides margin to ensure that fuel is not damaged during Modes 1 and 2 or as a result of anticipated operational occurrences.

The Ginna Current Licensing Basis is detailed in the topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", prepared under the Westinghouse Owner's Group Life Cycle Management & License Renewal Program. The RVI components described in the topical report bound the Ginna RVIs.

The NRC conducted an aging management review of the Ginna RVI components utilizing the topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals". The NRC reviewed Ginna's License Renewal Application, the supporting information in the Ginna UFSAR, and Ginna's responses to the License Renewal RAIs to determine whether any SSCs that should be within the scope of license renewal were not identified by the applicant and concluded that no omissions could be identified. In addition, the NRC performed an independent assessment to determine whether any components that should be subject to an Aging Management Review (AMR) were not identified by the applicant and concluded that no omissions could be found. On the basis of this review, the NRC concluded that Ginna has appropriately identified those portions of the reactor vessel and its associated (supporting) SSCs that are within the scope of license renewal, as required by 10 CFR54.5(a), and that the applicant has appropriately identified those portions of the reactor vessel and its associated (supporting) SSCs that are subject to an AMR, as required by 10CFR 54.21(a)(1).

2.1.4.2 Technical Evaluation

2.1.4.2.1 Introduction

This section of the report summarizes the evaluations, and their results, of the potential materials degradation issues arising from the effect of extended power uprate (EPU) on the performance of reactor internals and core support materials at R. E. Ginna Station.

The Westinghouse Owners' Group (WOG) Life Cycle Management & License Renewal Program prepared topical report WCAP–14577, License Renewal Evaluation: Aging Management for Reactor Internals, The topical report describes the aging degradation mechanisms to determine the aging effects. All identified effects are evaluated to identify potential degradation of RVI intended functions. The evaluation also included the time-limited aging analyses (TLAAs). All effects and TLAAs that require management during an extended period of operation are identified in the report. The report has been utilized in the NRC aging management review of the Ginna RVI components.

The NRC review of the WOG topical report concluded that the report provides an acceptable demonstration that the applicable effects of aging on reactor vessel internals components will be adequately managed for the WOG plants, such that there is a reasonable assurance that the RVI components will perform their intended functions in accordance with the current licensing basis during the period of extended operation. The EPU evaluation considered potential changes in the aging effects due to the change in the service conditions resulting from the proposed EPU conditions. These are considered below:

The primary objective of the EPU assessment was to ensure that the new EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on the RVI components during years 40-60, nor change the manner in which the component aging will be managed by the aging management program credited in the topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", and accepted by the NRC in the SER.

The relevant potentially impacted degradation (aging) mechanisms are:

- Integrity of reactor vessel fuel cladding materials,
- Transgranular (TGSCC), and intergranular stress corrosion cracking (IGSCC) of stainless steels,
- Primary water Stress Corrosion Cracking (PWSCC) of Alloy 600 and Alloy X-750 components, and
- Irradiation Assisted Stress Corrosion Cracking (IASCC) and Void Swelling of austenitic steel material internals.

An assessment of these aging mechanisms is considered in the following subsections.

2.1.4.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Proposed EPU Service Conditions

A review of the EPU design parameters (LR section 1.1, Nuclear Steam Supply System Parameters, and Table 1-1) indicates that the following changes in the RCS chemistry conditions, and neutron fluence levels will occur during operations after the EPU implementation:

- The reactor coolant lithium/boron program is coordinated such that an elevated 7.2 pH value is maintained during the fuel cycle (up to 1500 ppm boron), with a maximum lithium level of less than or close to 3.5 ppm.

Westinghouse estimated the maximum fast neutron (E > 0.1 MeV) exposure of the R. E. Ginna reactor internals for operating periods of 32 and 54 effective full power years (EFPY). These estimates were based, in part, on work that was completed to support pressure vessel integrity evaluations for the Extended Power Uprate (EPU) program. These estimated maximum exposure values are summarized in Table 2.1.4-1.

- The following fast neutron fluence levels (E>1.0 Mev) are expected during continued service under uprated condition:

  - Fluence/EFPY 1.4e+21 n/cm^2
  - Fluence/32 EFPY 4.6e+22 n/cm^2
  - Fluence/54 EFPY 8.6e+22 n/cm^2

These maximum exposures occur on the inside surface of the baffle plates opposite the central sections of the reactor core.

2.1.4.2.3 Description of Analyses and Evaluations

The effect of changes in service conditions due to the proposed EPU on the performance of the reactor vessel internals materials is discussed in the following paragraphs.

Materials Specifications, Weld Controls and NDT Inspections

The NRC's acceptance criteria for reactor internals and core support materials are based on GDC-1 and 10CFR50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Ginna's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277 and BAW-2248.

Stress Corrosion Cracking

The two degradation mechanisms that are operative in the internals austenitic stainless steels are intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). Susceptible materials, sensitized microstructure, and the presence of oxygen are required for the occurrence of IGSCC, while the introduction of halogens such as chlorides and the presence of oxygen are prerequisites for the occurrence of TGSCC. The principal method of preventing IGSCC and TGSCC is by water chemistry control. Westinghouse specifies that the reactor coolant chemistry be rigorously controlled, particularly with regard to oxygen, chlorides and other halogens. Ingress from other species, such as demineralizer resins, is carefully monitored, and corrective actions are taken to preclude exposure. In addition, startup transient oxygen levels are minimized as Ginna follows the recommendations of US NRC Reg. Guide 1.44 in that oxygen controls are established prior to elevated temperature operation.

Primary water stress corrosion cracking (PWSCC) is another form of IGSCC degradation that has been observed in Alloy 600 and Alloy X-750 materials in PWR applications. The RCCA guide tube support pins and clevis insert bolts at Ginna are fabricated from X-750 material; the clevis inserts are manufactured from Alloy 600 material.

The cracking of X-750 material is attributed to a combination of high stress and undesirable microstructure. The heat treatment specification for the replaced splint pin material and the support pin design at Ginna was to provide a more PWSCC resistant microstructure and lower stress condition, such that PWSCC exposure of the X-750 guide tube support pins is not considered significant. The Alloy X-750 clevis insert bolts in older plant designs experienced cracking in some plants after approximately 13 years of operation. However the degradation of clevis insert bolts would not be expected to result in a loss of intended function since the design geometry is such that the insert sits in a constrained groove and degradation of the bolts would not be expected to cause the displacement of the clevis insert from its original position.

The Alloy 600 clevis inserts experience lower fluence, temperature, and stresses in comparison to the support pins. The clevis inserts experience essentially compressive stress and no failures have been reported. Furthermore, like the clevis insert bolts, a failure of the clevis inserts would not result in a loss of intended function due to the nature of the design. Therefore, the effects of PWSCC on the clevis inserts are not considered significant.

The topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", considered the potential SCC degradation and concluded that the effects of all forms of SCC are not significant for Alloy 600, X-750, and stainless steel RVI components. The NRC review of the topical report concluded that there is a reasonable assurance that the RVI components will perform their intended functions in accordance with the current licensing basis during the period of extended operation.

The proposed EPU chemistry program at Ginna suggested operating at elevated 7.2 pH level (up to 1500 ppm Boron), while the Lithium level is maintained at less than or close to 3.5 ppm (Reference 2).The chemistry changes resulting from the EPU do not involve introduction of any of the (stress, oxygen or halogen) contributors, therefore no impact on the stress corrosion cracking material degradation is expected in the RVI components as a result of the EPU.

Fuel-Cladding Corrosion Effects

The proposed Ginna EPU lithium, boron, and pH management program was reviewed. The proposed chemistry program for EPU suggested operating at an elevated 7.2 pH level (up to 1500 ppm Boron), while the Lithium level is maintained at less than or close to 3.5 ppm. These conditions are bounded by the proposed Electric Power Research Institute (EPRI) chemistry guidelines (Reference 1). Since these guidelines are specifically designed to prevent fuel-cladding corrosion effects such as fuel deposit buildup and Alloy 600 PWSCC, there will be no adverse effect on fuel-cladding corrosion. Experience with operating plants as well as with the guidelines provided by EPRI (Reference 1) suggest that increasing initial lithium concentrations up to 3.5 ppm with controlled boron concentrations to maintain pH values between 6.9 to 7.4 has not produced any undesirable material integrity issues that could be statistically defined from the database of lab results available in 2003.

IASCC

Irradiation embrittlement is possible in the reactor internals components fabricated from austenitic stainless steel and nickel-based alloys with expected neutron fluences in excess of 1 x 10^{21}n/cm2 (E > 0.1 MeV). If the expected neutron fluence is less than approximately 1 x 10^{21} n/cm2 (E > 0.1 MeV), then the changes in mechanical properties due to neutron exposure are insignificant. The reactor internals components with fluences greater than 1 x 10^{21} n/cm2 (E > 0.1 MeV) (e.g., lower core barrel, baffle/former assembly, baffle/former bolts, lower core plate and fuel pins, lower support forging, clevis bolts) are potentially susceptible to irradiation embrittlement.

The EPU expected maximum fast neutron (E > 0.1 MeV) exposure levels of the R. E. Ginna reactor internals for operating periods of 32 and 54 effective full power years (EFPY) are listed in Section 2.1.4.2.2 above. These values are consistent with the values reported in Section 3.1.1.2 of the License Renewal Application (LRA) topical report. Section B2.1.27 of the LRA identifies the following RVI components as being exposed to the highest in-core neutron radiation fields and hence most susceptible to crack initiation and growth due to IASCC and loss of fracture toughness due to neutron irradiation embrittlement and/or void swelling.:

- Lower core plate and fuel alignment pins
- Lower support columns
- Core barrel and core barrel flange in active core region
- Thermal shield
- Bolting-lower support column, baffle-former, and barrel-former

Data from power reactor irradiation of Type 304 and Type 316 stainless steel are available from several studies (References 2, 3, and 4). Embrittlement, as evidenced by increases in yield strength and decreases in uniform and total elongation, is common in these materials after irradiation. Studies (References 2 and 3) showed that embrittlement of stainless steel can occur at fluences as low as 1 x 10^{21} n/cm^2 (E > 0.1 MeV) in the more susceptible stainless steel materials such as 304SS. These same studies showed that the rate of change in mechanical properties is reduced at fluences above 2 x 10^{22} n/cm^2 (E > 0.1 MeV).

No instance of service related internals degradation has been recorded that can be directly attributed to irradiation embrittlement. However, the end-of-life fluence level for some internals components is approximately 1 x 10^{23} n/cm2 to 1 x 10^{27} n/cm2 (E > 0.1 MeV), therefore Ginna will continue to participate in the industry Materials Reliability Program/IssuesTask Group efforts on reactor internals and monitor developments in this area.

The NRC's review (R.E.Ginna License Renewal SER) concluded that Ginna's Generic Aging Lessons Learned (GALL) process identified in the LRA is consistent with the GALL Report (NUREG-1801) and that RGE has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation as required by 10CFR54.21(a)(3).

Void Swelling

The Void swelling is the gradual increase in size (physical dimension) of the RVI stainless steel component caused by the formation and growth of helium-vacancy clusters into voids due to the effect of irradiation. Although the effects of swelling can be potentially significant for those components which experience significant neutron irradiation while operating at elevated temperatures, the actual plant operations do not appear to produce the conditions necessary for significant swelling. Recent data from Point Beach and Farley suggested very small (0.01% to 0.03%) amounts of swelling in baffle bolts. Extrapolation of these data using a simple square law suggest no concern with respect to void swelling until the end of extended life in U.S. PWRs. Fuel management schemes to reduce neutron leakage from the core have reduced one of the major factors contributing to swelling, and mechanisms such as creep and stress relaxation serve to reduce some of the adverse effects. The topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", examined the effects of swelling and concluded that any actual swelling of the susceptible internals will not prevent them from performing their intended function during the license renewal period.

Industry data on swelling are currently being evaluated as part of the WOG and Materials Reliability Program (MRP). At present there have been no indications from the different bolt removal programs or functional 'evaluations' that there are any discernible effects attributable to swelling. Ginna will continue to participate and follow up industry efforts to investigate swelling effects on the reactor vessel internals.

Thermal Aging

Thermal aging of cast stainless steel can lead to precipitation of additional phases in the ferrite and growth of existing carbides at the ferrite/austenitic boundaries that can result in loss of ductility and fracture toughness of the material. The susceptibility to thermal aging is a function of the material chemistry, aging temperature, and time at temperature. All the cast duplex stainless steel reactor internals in the Westinghouse-designed NSSS are made from CF-8 or CF-8A materials which contain low or zero Molybdenum and are less susceptible to thermal aging.

At Ginna Station, the lower core support column and mixing vanes of the flow mixer are potentially fabricated from the cast austenitic stainless steel material. Although these components are potentially susceptible to thermal aging embrittlement under prolonged exposure to elevated temperatures, the chemistry content and the service temperatures (354°F – 617°F) at Ginna are not favorable to produce adequate loss of toughness to have any significant impact on the structural integrity

The topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", conducted an evaluation of the effects of thermal aging and concluded that the effects of thermal aging are insignificant to all of the reactor internals components and aging management of this effect will not be required during an extended period of operation.

2.1.4.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The NRC's acceptance criteria for reactor internals and core support materials are based on GDC-1 and 10CFR50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Ginna's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277 and BAW-2248.

On the basis of the review and audit of the Ginna'a Reactor Vessel Internals License Renewal SER, the NRC concluded that those portions of the program for which Ginna claimed consistency with GALL program are consistent with GALL. This includes a commitment to develop a reactor vessel internals inspection program for use during the scheduled 2009 reactor vessel ten-year ISI. Furthermore NRC's review of the exceptions to the GALL program found that Ginna has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10CFR54.21(a)(3). The NRC's review of the UFSAR supplement for the Aging Management Program (AMP) found that it provided an adequate summary description of the program as required by 10CFR54.21(d).

All EPU evaluations and discussions included in this Licensing Report addressed meeting the licensing basis for a time period of up to 54 EFPY or 60 years of service

2.1.4.2.5 Results

The results of the reactor vessel internals material degradation assessment showed that no materials degradation issues will result from the proposed power uprate at Ginna Station. On this basis it is concluded that the new EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on their components during 60 years of operation, nor will the EPU change the manner in which the component aging will be managed by the aging management program credited in the topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Vessel Internals", and accepted by the NRC in the SER.

2.1.4.2.6 References

1. EPRI TR-1002884, Volume 1, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Rev. 5, September 2003.

2. Kangilaski, M., "The Effects of Neutron Radiation on Structural Materials," REIC Report No. 45, Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio (June 1967).

3. Robbins, R. E., et. al., "Post Irradiation Tensile Properties of Annealed and Cold Worked AIOSI – 304 Stainless Steel," Trans American Nuclear Society, pp 488-489 (Nov. 1967).

4. Bloom, E. E., "Mechanical Properties of Materials in Fusion Reactor First-Wall and Blanket Systems," Journal of Nuclear Materials, 85 and 86, pp 795-804 (1979).

2.1.4.3 Conclusions

Ginna has evaluated the effects of the proposed EPU on the susceptibility of reactor internals and core support materials to degradation mechanisms and concludes that it has identified appropriate degradation management programs to address the effects of changes in operating temperature, RCS chemistry and neutron fluence on the integrity of reactor internals and core support materials. Ginna further concludes that the licensee has demonstrated that the reactor internals and core support materials will continue to be acceptable and will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-1 and 10CFR50.55a following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU acceptable with respect to reactor internals and core support materials.

Table 2.1.4-1

Operating Time [EFPY]	Fluence (E > 0.1 MeV) [n/cm^2]
32	9.72E+22
54	1.65E+23

2.1.5 Reactor Coolant Pressure Boundary Materials

2.1.5.1 Regulatory Evaluation

The reactor coolant pressure boundary (RCPB) defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The Ginna Nuclear Power Plant, LLC (Ginna) evaluation of reactor coolant pressure boundary materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for reactor coolant pressure boundary materials are based on:

- 10CFR50.55a and GDC-1 insofar as they require that structures, systems, and components important-to-safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC-14, insofar as it requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture

- GDC-31, insofar as it requires that the reactor coolant pressure boundary be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized

- 10CFR50, Appendix G, which specifies fracture toughness requirements for ferritic components of the reactor coolant pressure boundary

Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletins 01-01, 02-01 and 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Ginna Current Licensing Basis

The principal components of the RCPB include the reactor vessel, pressurizer, steam generators, reactor coolant pumps, and the essential Class 1 piping and valves (including the regenerative and letdown heat exchangers). The RCS consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam

generator. Additional details of the RCS are provided in Sections 5.1, 5.2, and 5.4 of the UFSAR.

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), Section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970's the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station reactor coolant pressure boundary materials relative to conformance to:

- 10CFR50.55a(a)(1) is described in UFSAR section 3.2.1, "Classification of Structures, Components, and Systems – Introduction." As part of SEP Topic III-1, the original codes and standards used in the design of the Ginna Station were compared with later licensing criteria, including 10CFR50.55a. The objective was to assess the capability of Ginna Station SSCs to perform their safety functions as judged by the later standards. Although several areas were identified where requirements had changed, all areas were satisfactorily resolved, and SEP Topic III-1 was closed.

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1. General Design Criterion 1 - Quality Standards and Records, wherein it is noted that all structures, systems, and components (SSCs) of the facility were classified according to their importance. The classification of structures and equipment is discussed in Section 3.2. SSCs were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. The quality control and quality assurance program for Ginna Station construction, and the current quality assurance program, under which systems were installed and are maintained, are described in UFSAR Section 17.1, and 17.2, respectively.

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review of this topic as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, *The Integrated Plant Safety Assessment Report, completed in August 1983.* Conformance to the requirements of GDC-4 is also described in the following:

— Ginna UFSAR, Section 3.11
  - Environmental Design of Mechanical and Electric Equipment

— Ginna UFSAR, Section 3.6
  - Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping
    - Pipe Breaks Inside Containment (SEP, Topic III-5.A)
    - Pipe Breaks Outside Containment (SEP, Topic III-5.B)

— Ginna UFSAR, Section 3.5
  - Missile Protection

- GDC-14 is described in Ginna UFSAR section 3.1.2.2.5, General Design Criterion 14 - Reactor Coolant Pressure Boundary. This UFSAR section states that all piping components and supporting structures of the reactor coolant system were designed as Class I (see UFSAR Sections 3.1.1.1.1 and 3.7.1.1.1 for description of the original Class designations used during plant design) and later reevaluated as Seismic Category I equipment as defined in Section 3.7. All pressure containing components of the reactor coolant system were designed, fabricated, inspected, and tested in conformance with the code requirements listed in UFSAR Table 5.2-1.

- GDC-31 is described in Ginna UFSAR, Section 3.1.2.4.2, GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary.

- 10CFR50, Appendix G is described in UFSAR section 3.2.2.1.2 for Accumulators, in UFSAR sections 5.3.2.2 and 5.3.2.3 regarding Pressure-Temperature limits, in UFSAR section 5.3.3 for reactor vessel integrity, in UFSAR section 5.4.5.3.2 for RHR overpressure protection, and in UFSAR sections 5.2.2.2 and 7.6.1 regarding protection during low power operation using the Low Temperature Overpressure Protection (LTOP) system.

The Ginna Station reactor coolant pressure boundary materials are addressed in Section 5.2.3 and Table 5.2-2 of the Ginna UFSAR. The reactor coolant pressure boundary materials are selected for the expected environment and service conditions and procured, fabricated and inspection in accordance with applicable sections of the ASME Boiler and Pressure Vessel Code.

In addition to the evaluations described in the UFSAR, the Ginna Station's reactor coolant pressure boundary was evaluated for plant license renewal, which is documented in sections 2.3.1.1, 3.1, 4.2 and 4.3 of NUREG-1786, the Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant, May 2004.

The Ginna Current Licensing Basis (CLB) is detailed in the Westinghouse topical report, WCAP-14575-A, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, which has been utilized in the aging

management review of the Ginna RCS. The scope of the RCPB components described in the topical report bounds the Ginna Class 1 piping and associated pressure boundary components. A reconciliation of the final SER for WCAP 14575-A applicant action items is provided in Table 3.2.0-1.2 of the license renewal application.

RCPB materials are further considered in LR section 2.1.6, Leak-Before-Break, and 2.14, The Effects of EPU on License Renewal and License Renewal Programs.

2.1.5.2 Technical Evaluation

2.1.5.2.1 Introduction

This section of the report summarizes the evaluations, and their results, of the potential materials degradation issues arising from the effect of the R. E. Ginna extended power uprate (EPU) on the performance of reactor coolant pressure boundary component materials.

The EPU evaluation assessed the potential effect of changes in the Reactor Coolant System (RCS) chemistry (impurities), pH conditions, and EPU service temperatures on the integrity of primary component pressure boundary materials during service. The evaluation includes:

- An assessment of the potential effect of water chemistry changes on the i) general corrosion (wastage) of carbon steel components and ii) stress corrosion cracking (SCC) of system austenitic stainless steel materials, and the management strategy of any issues there from

- An assessment of the effect of change in the service temperature on i) primary water stress corrosion cracking (PWSCC) of Alloy 600/182/82 nickel base alloys and ii) thermal aging of cast stainless steel (CASS) materials, and the management strategy of any issues there from

These assessments are discussed in the following subsections.

2.1.5.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Proposed EPU Service Conditions

A review of the EPU design parameters (LR section 1.1, Nuclear Steam Supply System Parameters, and Table 1-1) indicates that the following changes in the RCS chemistry and service temperature conditions will occur during operations after the EPU implementation:

- The EPU reactor coolant lithium/boron program is coordinated such that an elevated 7.2 pH value is maintained during the fuel cycle (up to 1500 ppm boron) while maintaining a maximum lithium level of less than or close to 3.5 ppm.

- A maximum increase ΔT in the peak steady state service temperature of 8.6°F at the reactor vessel closure head and hot leg locations and a decrease ΔT in service temperature of 3.1°F at the bottom head location will occur due to the uprating. This is summarized in Table 2.1.5-1.

2.1.5.2.3 Description of Analyses and Evaluations

The effect of change in service conditions (temperature and water chemistry) due to the proposed EPU on the performance of the reactor coolant pressure boundary materials is discussed in the following paragraphs.

General Corrosion/Wastage of Carbon Steel Components

The Ginna EPU reactor coolant lithium/boron program is coordinated such that an elevated 7.2 pH value is maintained during the fuel cycle (up to 1500 ppm boron) while maintaining a maximum lithium level of less than or close to 3.5 ppm. Experience with operating plants as well as with the guidelines provided by EPRI (PWR Primary Water Chemistry Guidelines: Vol. 1, Rev. 5, EPRI Palo Alto CA: 2003, TR-1002884) suggest that increasing initial lithium concentrations of up to 3.5 ppm with controlled boron concentrations to maintain pH values between 6.9 to 7.4 does not produce any undesirable material integrity issues

The Ginna Boric Acid Corrosion Control (BACC) program is discussed in Section B2.1.6 of the WCAP-14575-A, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components. The NRC reviewed Ginna's BACC program and found Ginna's RAI responses acceptable since Ginna expanded the BACC program scope to become consistent with Generic Aging Lessons Learned (GALL) report, incorporated lessons learned from Davis-Besse, and addressed NRC's generic communications. On the basis of its review and audit findings the NRC concluded that Ginna demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation as required by 10CFR54.21(a)(3).

SCC of Austenitic Stainless Steels

The two degradation mechanisms that are operative in the pressure boundary austenitic stainless steel (base and weld) materials in the RCPB are intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). Susceptible materials, sensitized microstructure, and the presence of oxygen are required for the occurrence of IGSCC, while the introduction of halogens such as chlorides and the presence of oxygen are prerequisites for the occurrence of TGSCC.

The EPU reactor coolant lithium/boron program is coordinated such that an elevated 7.2 pH value is maintained during the fuel cycle (up to 1500 ppm boron) while maintaining a maximum lithium level of less than or close to 3.5 ppm.

The chemistry changes resulting from the EPU do not involve introduction of any of these contributors so that no effect on material degradation is expected in the stainless steel components as a result of the EPU.

Alloy 600/82/182 Components at R. E. Ginna

1. Alloy 600 and Alloy 82/182 weld deposit are present in the Ginna RCS at the following locations:

   i) Bottom-mounted instrumentation penetrations in the bottom head of the reactor vessel. The instrument nozzles are Alloy 600, welded to the ID of the head with partial penetration welds using 82/182 weld deposit.

   ii) Radial core support lugs in the lower shell of the reactor vessel. The core support lugs are Alloy 600 welded to the interior surface of the shell with 82/182 weld deposit.

   iii) Primary tube sheet surface of the replacement steam generators. The primary surface of each RSG tube sheet (carbon steel) is weld overlaid with alloy 82 weld deposit.

2. Alloy 690 and Alloy 52/152 weld deposit is present at the following locations:

   i) U-tubes of the replacement steam generators. The tubing for the RSGs was Alloy 690 TT (thermally treated).

   ii) Weld-deposited "butter" layers on the weld preparations of the primary inlet and outlet nozzles in the channel heads of the RSGs are Alloy 52/152.

   iii) Divider plates in the channel heads of the RSGs are Alloy 690 plate material welded to the primary tube sheet surface of each generator with Alloy 52/152.

   iv) CRDM nozzles, vent nozzle and instrument nozzles in the replacement reactor vessel head. All nozzles are Alloy 690 TT (thermally treated) and welded to the ID of the head with partial penetration welds using Alloy 52 weld deposit.

PWSCC of Nickel Base Alloy 600/ 82/182 and Alloy 690/52/152 Sub-Component Materials

Laboratory and field data suggests that Nickel base alloys are susceptible to Primary Water Stress Corrosion Cracking (PWSCC) (Methodologies to Assess PWSCC Susceptibility of Primary Component Alloy 600 Locations in Pressurized Water Reactors, Gutti V. Rao Proceedings of the Sixth International Symposium on "Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors," August 1–5, 1993, Baha Resort Hotel, San Diego, CA.). Service experience with the Alloy 600 base metal and the associated Alloy 82/182 weld materials in PWRs over the past several years has shown that they are subject to PWSCC (Survey of Boric Acid Corrosion Events, Samantha Crane and William Cullen U.S. NRC,

October 22, 2004). A superior PWSCC resistant Alloy 690 material is being utilized in the industry to replace the Alloy 600 components in service. The sub-component locations of Alloy 600 and Alloy 690 and their associated (82/182 and 52/152) welds are listed in the preceding section. At Ginna the reactor vessel head containing Alloy 600/82/182 penetrations was replaced during 2003 with a new RV head consisting of the superior PWSCC resistant Alloy 690/52/152 CRDM penetrations.

The Alloy 600/82/182 PWSCC susceptibility is a thermally activated process and the industry Materials Research Program (MRP) has been tracking PWSCC in the RV head Alloy 600 components by examining the integrated time-temperature history (Reference 1). The time-temperature history has been evaluated using a term called Effective Degradation Years (EDY) based on a common reference temperature.

The EDY value is the operating time normalized to a reference temperature of 600°F. The standard Arrhenius activation energy dependence on temperature is applied to each time period with a distinct head temperature, with the thermal activation energy for crack initiation applied:

$$EDY = \sum_{j=1}^{n} \left\{ \Delta EFPY_j \exp\left[-\frac{Q_i}{R} \left(\frac{1}{T_{head,j}} - \frac{1}{T_{ref}} \right) \right] \right\}$$

where:

EDY	=	total effective degradation years, normalized to a reference temperature of 600°F
$\Delta EFPY_j$	=	effective full power years accumulated during time period j
Q_i	=	activation energy for crack initiation (50 kcal/mole)
R	=	universal gas constant (1.103×10-3 kcal/mol-°R)
$T_{head,j}$	=	100% power head temperature during time period j (°R = °F + 459.67)
T_{ref}	=	reference temperature (600°F = 1059.67°R)
n	=	number of time periods with distinct 100% power head temperatures since initial head operation

Effect of EPU on the PWSCC Susceptibility of Reactor Vessel Closure Head Penetrations at Ginna Station

At Ginna the Reactor Vessel Closure Head with Alloy 600/82/182 penetrations was replaced during 2003 with a new head with Alloy 690/52/152 CRDM penetrations. Laboratory and field experience to date suggests that Alloy 690 and associated Alloy 52/152 welds are resistant to PWSCC. On this basis, even though an increase of 8.6°F (Table 2.1.5-1) in the closure head service temperature is predicted due to the proposed EPU at Ginna, the proposed uprating is not expected to have any impact on the PWSCC degradation of the Alloy 690/52/152 Reactor Vessel Closure Head Penetrations. However, since only a limited amount of field data is available on the Alloy 690/52/152, to ensure safe management of the potential PWSCC issue, the NRC has recommended the following closure head inspection plan for the replacement heads.

Inspection Requirements for Replacement Heads with Alloy 690 Nozzles

NRC ORDER EA-03-009

The NRC Order EA-03-009 dated February 20, 2004 stipulated the following requirements for the interim inspection of replacement reactor pressure vessel (RPV) closure heads.

No RPV closure head and closure head penetration nozzle inspections shall be required for the outage during which the RPV head was replaced. Thereafter, until the head reaches 8 EDY, the RPV head and head penetration nozzle inspections shall be performed as follows:

- An inspection of the RPV closure head meeting the following requirements be completed every third refueling outage or every 5 years which ever occurs first:

  Bare metal visual examination of 100 percent of the RPV head surface (including 360° around each RPV head penetration nozzle). For RPV heads with the surface obscured by support structure interferences which are located at RPV head elevations down slope from the outermost RPV head penetration, a bare metal visual inspection of no less than 95 percent of the RPV head surface may be performed provided that the examination shall include those areas of the RPV head upslope and downslope from the support structure interference to identify any evidence of boron or corrosive product. Should any evidence of boron or corrosive product be identified, the licensee shall examine the RPV head surface under the support structure to ensure that the RPV head is not degraded.

- A nonvisual NDE inspection of each head penetration, meeting the requirements of either i), ii) or iii) below, be completed at least every 4 refueling outages or every 7 years, which ever occurs first:

  i) Ultrasonic testing of the RPV head penetration nozzle volume (i.e., nozzle base material) from 2 inches above the highest point of the root of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) to 2 inches below the lowest point at the toe of the J-groove weld on a horizontal plane perpendicular to the nozzle axis (or the bottom of the nozzle if less than 2 inches) OR from 2 inches above the highest point of the root of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) to 1.0-inch below the lowest point at the toe of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) and including all RPV head penetration nozzle surfaces below the J-groove weld that have an operating stress level (including all residual and normal operation stresses) of 20 ksi tension and greater. In addition, an assessment shall be made to determine if leakage has occurred into the annulus between the RPV head penetration nozzle and the RPV head low-alloy steel.

  ii) Eddy current testing or dye penetrant testing of the entire wetted surface of the J-groove weld and the wetted surface of the RPV head penetration nozzle base material from at least 2 inches above the highest point of the root of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) to 2 inches below

the lowest point at the toe of the J-groove weld on a horizontal plane perpendicular to the nozzle axis (or the bottom of the nozzle if less than 2 inches); OR from 2 inches above the highest point of the root of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) to 1.0-inch below the lowest point at the toe of the J-groove weld (on a horizontal plane perpendicular to the nozzle axis) and including all RPV head penetration nozzle surfaces below the J-groove weld that have an operating stress level (including all residual and normal operation stresses) of 20 ksi tension and greater.

iii) A combination of (i) and (ii) above, to cover equivalent volumes, surfaces and leak paths of the RPV head penetration nozzle base material and J-groove weld as described in (i) and (ii). Substitution of a portion of a volumetric exam on a nozzle with a surface examination may be performed with the following requirements:

(a) On nozzle material below the J-groove weld, both the outside diameter and inside diameter surfaces of the nozzle must be examined.

(b) On nozzle material above the J-groove weld, surface examination of the inside diameter surface of the nozzle is permitted provided a surface examination of the J-groove weld is also performed.

Recently Proposed Industry Guidance for the Inspection of RPV Closure Head Penetrations

On September 4, 2004 the Nuclear Energy Institute (NEI), the Industry and MRP presented Inspection Guidance (Reference 1) for the RPV closure head based on safety assessment of closure head penetrations, to the NRC. The following requirements apply to replacement heads having Alloy 690 nozzles with Alloy 52/152 J-groove attachment welds.

- An initial bare metal visual examination shall be performed before or during the third refueling outage after installation of the replacement head, or within 5 calendar years of replacement, whichever occurs first. Repeat bare metal visual examinations shall be performed at least every third refueling outage or every 5 calendar years, whichever occurs first.

- Prior to installation of the replacement head the penetrations in the replacement head shall be characterized using NDE techniques. Such information may be valuable for interpretation of data collected during in-service examinations.

- All plants having replacement heads with Alloy 690 nozzles attached with Alloy 52/152 J-groove welds shall perform an initial in-service Level 1, 2, or 3 nonvisual NDE within 10 calendar years following head replacement. (This 10 calendar year period may be extended by as much as 1 year to enable the inspection to coincide with a plant outage.) However, for this inspection to satisfy the initial in-service non-visual examination

requirement, it shall be performed no earlier than 6 calendar years after head replacement.

- All plants having replacement heads with Alloy 690 nozzles attached with Alloy 52/152 J-groove welds shall perform a repeat nonvisual NDE (Level 1, 2, or 3) 10 calendar years following the most recent nonvisual NDE. This 10 calendar year period may be reduced or extended by as much as 1 year to enable the inspection to coincide with a plant outage.

- The MRP is currently working with the American Society of Mechanical Engineers (ASME) to develop a code case which would be applicable for the reactor vessel closure head and penetration nozzles (Summary of September 8, 2004 Meeting with Nuclear Energy Institute (NEI) and Electric Power Research Institute on Reactor Vessel Closure Head (RVCH) Penetrations Safety Assessment, Memorandum from Joseph L. Birmingham, NRC to Cathy Haney, NRC, October 12, 2004).

After one cycle of operation, Ginna performed a visual inspection of the replacement reactor vessel closure head during the Spring 2005 refueling outage, and concluded that the new closure head maintained RCPB integrity.

Ginna will continue to monitor the Industry programs and recommendations to manage the issue for the new vessel head and take appropriate actions as necessary.

Effect on the PWSCC Susceptibility of Alloy 600/82/182 BMI Penetrations

The Bottom-mounted Instrument (BMI) penetrations at Ginna are made of PWSCC susceptible Alloy 600/82/182 materials. A review of the service temperature data (Table.2.1.5-1) suggests that the power uprating at Ginna actually reduces the PWSCC susceptibility due to a 3.2°F decrease in service temperature at the BMI. However since the BMIs are fabricated from PWSCC susceptible Alloy 600/82/182 material, the Ginna BMIs are subject to the NRC Bulletin 2003-02 requiring certain inspections for the safe management of the BMI PWSCC issue. In support of this inspection plan, MRP (MRP-2004-04) recommended the following:

> Licensees/Utilities with an upcoming 10-year reactor vessel ISI should plan to supplement the lower vessel head bare metal visual inspections with volumetric inspections. The inspections considered include UT of the nozzle and either enhanced visual of the J-groove weld or ECT of the J-groove weld, depending on vendor demonstrated capabilities.
>
> All other plants should continue with visual inspections per previous MRP-2003-017 recommendations.

The Ginna station conducted visual inspections during the September 2003 and March 2005 outage inspections and reported no evidence of leakage.

Thermal Aging

Thermal aging of cast stainless steel can lead to precipitation of additional phases in the ferrite and growth of existing carbides at the ferrite/austenitic boundaries that can result in loss of ductility and fracture toughness of the material. The susceptibility to thermal aging is a function of the material chemistry, aging temperature and time at temperature.

At Ginna a small increase (8.6°F) in the hot leg temperature was assessed due to the EPU. The effect of this change in the service temperature on the thermal aging is considered.

The topical report WCAP-14575-A, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, indicates that thermal aging causes reduction in fracture toughness of the CASS component material and hence reduction in the critical flaw size that could lead to component failure. The impacted RCPB CASS components include primary piping and its welds, valve bodies and pump casings. The topical report WCAP-14575-A proposed programs to manage the effects of thermal aging of CASS components during the period of extended operation. The NRC's assessment of these programs is contained in Section 3.3.3 of the SER.

2.1.5.2.4 Evaluation of the impact of EPU on the Renewed Plant Operating License Evaluations and License Renewal Programs

On the basis of the review and audit of Ginna's Reactor Coolant Pressure Boundary materials License Renewal SER, the NRC concluded that those portions of the program that Ginna claimed to be consistent with the GALL program were found to be consistent with the GALL. Furthermore NRC's review of the exceptions to the GALL program found that Ginna has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10CFR54.21(a)(3).

On the basis of the above, the NRC further concluded that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of subcomponents subject to an AMR, such that there is a reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10CFR54.29(a).

The evaluation of the small increase in temperature on the effectiveness of the aging management programs was evaluated and determined that the program will remain adequate.

All EPU evaluations and discussions included in this LR section ,addressed meeting the licensing basis for a time period of up to 54 EFPY or 60 years of service.

2.1.5.2.5 Results

Based on the results of the assessment of the potential materials degradation issues resulting from the proposed Power Uprating at Ginna, It is concluded that:

- No new material degradation issues of carbon steel boric acid corrosion are expected due to the EPU water chemistry. The NRC reviewed Ginna's Boric Acid Corrosion Control program and found Ginna's RAI responses acceptable.

- The risk for PWSCC of the Reactor Vessel Closure Head Penetrations is expected to be minimal since the vessel head was replaced during September 2003 with a new head fabricated with Alloy 690/52/152 penetrations. As of April 2005, there had been only one complete cycle of operation with the replacement reactor vessel closure head. Ginna will continue to monitor the Industry program to manage the issue for the new vessel head.

- The risk for PWSCC of the Alloy 600/82/182 BMI penetrations will actually decrease due to the proposed uprating since the service temperature of the BMIs will be lower by 3.2°F. Visual examinations of the BMIs were conducted during the September 2003 and March 2005 outages and found no evidence of leakage.

- The effect of a small increase in the hot leg temperature on the thermal aging of piping and welds was assessed. Ginna will follow the WOG recommended Aging Management Program (AMP) to address the impact of thermal aging embrittlement on the LBB evaluations for the period of extended operation.

- The NRC's review (R. E. Ginna License Renewal SER) concluded that Ginna's Generic Aging Lessons Learned (GALL) process identified in the License Renewal Application (LRA) is consistent with the GALL Report (NUREG-1801) and that Ginna has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation as required by 10CFR54.21(a)(3).

- The chemistry changes resulting from the EPU do not involve introduction of any of the contributors to SCC of austenitic stainless steel, therefore no material degradation is expected in the stainless steel components as a result of the EPU.

The results of the reactor coolant pressure boundary material degradation assessment showed that no new materials degradation issues will result from the proposed power uprating at Ginna Station. On this basis it is concluded that the new EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on their components during 60 years of operation, nor will the EPU change the manner in which the component aging will be managed by the aging management program credited in the LRA and accepted by the NRC in the SER.

2.1.5.2.6 References

1. EPRI-MRP-117, "Inspection Plan for Reactor Vessel Closure Head Penetrations in US Power Plants," July 2004.

2.1.5.3 Conclusion

Ginna has reviewed the effects of the proposed EPU on the susceptibility of reactor coolant pressure boundary materials to known degradation mechanisms, and concludes that it has identified appropriate degradation management programs to address the effects of changes in the coolant chemistry and operating temperature due to EPU on the integrity of reactor coolant pressure boundary materials. Ginna further concludes that it has demonstrated that the reactor coolant pressure boundary materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-1, GDC-4, GDC-14, GDC-31, 10CFR50, Appendix G, and 10CFR50.55a. Therefore, Ginna finds the proposed EPU acceptable with respect to reactor coolant pressure boundary materials.

Table 2.1.5-1 Summary of Service Temperature Changes in the RV Closure Head and Bottom-Mounted Instrumentation (BMI) Penetrations Due to the Proposed 19.5% Power Uprate

Core Power Level (MWt)	Location	Temperature (°F)		Maximum Change in the Steady State Peak Temperature Due to Uprating (delta T °F)
		High	Low	
1520 (Current)	RV Upper Head	590.6	576.8	
1520 (Current)	BMI Penetration	543.1	528.0	
1811 (19.5% Uprate)	RV Upper Head & Hot Leg	599.2	587.9	+8.6
1811 (19.5% Uprate)	BMI Penetration	539.9	528.0	-3.2

2.1.6 Leak-before-Break

2.1.6.1 Regulatory Evaluation

Leak-before-break (LBB) analyses provide a means for eliminating the dynamic effects of postulated pipe ruptures for a piping system from the design basis. The NRC approval of LBB for a plant permits the licensee to: remove protective hardware along the piping system (e.g., pipe whip restraints and jet impingement barriers); and redesign pipe-connected components, their supports, and their internals. Ginna Nuclear Power Plant, LLC's (Ginna) review for LBB included:

- Direct pipe failure mechanisms (e.g., water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions)
- Indirect pipe failure mechanisms (e.g., seismic events, system overpressurizations, fires, flooding, missiles, and failures of structures, systems, and components in close proximity to the piping)
- Deterministic fracture mechanics and leak detection methods

The NRC's acceptance criteria for LBB are based on draft SRP, Section 3.6.3; NUREG-1061 Volume 3; and GDC-4, insofar as it allows for the exclusion of dynamic effects of postulated pipe ruptures from the design basis.

Specific review criteria are contained in the draft SRP, Section 3.6.3;and NUREG-1061 Volume 3; other guidance is provided in Matrix 1 of RS-001.

Ginna Current Licensing Basis

Unresolved Safety Issue A-2, asymmetric loss-of–coolant accident (LOCA) loads for the Ginna primary loop piping licensing basis is discussed in *Ginna Updated Final Safety Analysis Report* (UFSAR), Sections 3.6.1.3.2.13 and 5.4.11.1.2. The current LBB licensing basis (see Section 4.7.7 of the Ginna License Renewal Safety Evaluation Report) (NUREG-1786) for the Ginna primary loop piping is based on WCAP-15837 (Reference 1), The LBB current licensing basis for the Ginna pressurizer surge line is discussed in the Ginna UFSAR, Section 3.6.1.3.2.14. A recent LBB analysis for the pressurizer surge line is documented in WCAP-16311-P (Reference 2) and was submitted to the NRC on September 20, 2004. The Residual Heat Removal (RHR) lines LBB current licensing basis is discussed in UFSAR Section 3.6.1.3.2.3 and was approved by the NRC (Reference 8). The accumulator lines LBB analysis (References 9, 10, and 11) was submitted the NRC on September 30, 2004. The Reactor Coolant Pump (RCP) casings analysis for the ASME Code Case N-481 analysis is documented in WCAP-15873 (Reference 3) and was approved by the NRC (see section 4.7.7 of the Ginna License Renewal Safety Evaluation Report, NUREG-1786).

2.1.6.2 Technical Evaluation

2.1.6.2.1 Introduction

The original structural design basis of the RCS for the Ginna Station required consideration of dynamic effects resulting from postulated pipe breaks and the need to incorporate protective measures for such breaks into the design. Subsequent to the original Ginna Station design, an additional concern regarding asymmetric blowdown loads was raised as described in *Unresolved Safety Issue A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System)* and Generic Letter 84-04. Reference 4 provided the NRC safety evaluation for the analyses submitted by Westinghouse for a group of plants that included the Ginna Station to resolve Unresolved Safety Issue A-2, asymmetric LOCA. Research by the NRC and industry, coupled with operating experience, determined that safety could be negatively impacted by placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives resulted in demonstrating that LBB criteria can be applied to RCS piping based on fracture mechanics technology. A letter from D. C. Dilanni (NRC) to R. W. Kober (Ginna) (Reference 5) confirmed that the asymmetric blowdown loads resulting from double-ended pipe breaks in primary loop piping need not be considered as a design basis for the Ginna Station.

The current structural design basis of the Ginna Station includes the application of LBB methodology to eliminate consideration of the dynamic effects resulting from pipe breaks in the reactor coolant system (RCS) primary loop piping, pressurizer surge line, RHR lines and the accumulator lines. The purpose of this LR section is to describe the evaluations performed to demonstrate that the elimination of these breaks from the structural design basis continues to be valid following implementation of the EPU, and that lines (primary loop piping, pressurizer surge line, RHR and the accumulator lines) for which Ginna credits LBB continue to comply with the requirements of GDC-4, the draft SRP section 3.6.3 and NUREG-1061, Volume 3.

To demonstrate the elimination of primary loop piping, pressurizer surge line, RHR and the accumulator lines pipe breaks, the following objectives had to be achieved:

- Demonstrate that margin exists between the "critical" flaw size and a postulated flaw that yields a detectable leak rate.
- Demonstrate that there is sufficient margin between the leakage through a postulated flaw and the leak detection capability.
- Demonstrate margin on the applied load.
- Demonstrate that fatigue crack growth is negligible.

These objectives were met in the current LBB analyses.

To support the EPU at the Ginna Station, the current LBB analyses were evaluated to address the proposed EPU conditions.

2.1.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The loadings, operating pressure, and temperature parameters for the EPU were used in the evaluation.

The parameters that are important in the evaluation are the piping forces, moments, normal operating temperature, and normal operating pressure. These parameters were used as input in the evaluation. The normal EPU operating temperature range and normal operating pressure conditions are provided in Table 1-1 of this Licensing Report.

Acceptance Criteria

The LBB acceptance criteria are based on the Draft SRP, Section 3.6.3 and NUREG-1061 Volume 3. The LBB recommended margins are as follows:

- Margin of 10.0 on leak rate
- Margin of 2.0 on flaw size
- Margin on loads of 1.0 (if using faulted load combinations by absolute summation method) or $\sqrt{2}$.

2.1.6.2.3 Description of Analyses and Evaluations

Primary Loop Piping

Westinghouse performed a plant-specific LBB analysis for the Ginna Station primary loop piping in 2002 for the Ginna Station License Renewal Program. The results of the analysis were documented in WCAP-15837 (Reference 1) and were approved by the NRC (see Section 4.7.7 of the Ginna License Renewal Safety Evaluation Report, NUREG-1786).

The recommendations and criteria proposed in NUREG-1061, Volume 3, and the draft SRP, Section 3.6.3 are incorporated in the evaluation. The primary loop piping dead weight, normal thermal expansion, and safe shutdown earthquake (SSE) and pressure loads due to the EPU conditions were employed. The EPU normal operating temperature range and pressure were used in the evaluation. The evaluation results demonstrated that all the LBB recommended margins for the primary loop piping continue to be satisfied for the EPU conditions.

Pressurizer Surge Line Piping

The original pressurizer surge line LBB analysis was approved by the NRC in a letter from D. M. Crutchfield, (NRC) to J. E. Maier (Ginna) (Reference 6). The original analysis did not consider the effects of thermal stratification since the thermal stratification phenomenon in the surge line was not a concern at that time. The pressurizer surge line is known to be subjected to thermal stratification and the effects of thermal stratification for the Ginna Station pressurizer surge line have been evaluated and documented in WCAP-12928 (Reference 7).

The Ginna pressurizer surge line analysis for the application of LBB considering the effects of the thermal stratification is documented in WCAP-16311-P (Reference 2) and is under review by the NRC. As documented in Section 2.2.2.1, NSSS Piping, Components and Supports, of this report, the current design basis pressurizer surge line loads and results including the effects of thermal stratification remain applicable for the EPU. The effects of the EPU conditions in Table 1-1 of this Licensing Report, on the Ginna Station surge line have been evaluated and determined to continue to comply with the LBB margin requirements. Therefore, the Ginna pressurizer surge line evaluation results determined that the conclusions of the current LBB analysis shown in WCAP-16311-P remain valid for the EPU conditions.

RHR and Accumulator Lines

The Ginna RHR lines LBB analysis was approved by the NRC in a letter from G. S. Vissing (NRC) to R. C. Mecredy (Ginna) (Reference 8). The accumulator lines LBB analysis is under review by the NRC. Section 2.2.2.1, NSSS Piping, Components and Supports, of this report concluded that there is an insignificant impact on the auxiliary piping systems attached to the Reactor Coolant Loop (RCL) due to the EPU conditions, since the pressures and temperatures are not changed. Therefore, the effects of the EPU conditions on the Ginna Station RHR and accumulator lines have an insignificant impact. The evaluation results determined that the conclusions of the current LBB analyses for the Ginna Station RHR and accumulator lines remain valid for the EPU conditions.

Limitations on the Application of LBB

The LBB approach should not be considered applicable to high energy fluid system piping, or portions thereof, that operating experience has indicated particular susceptibility to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer or low and high cycle (i.e., thermal, mechanical) fatigue. For the Ginna Station LBB applications, these limitations are addressed in the WCAP-15837 (Reference 1, for primary loop piping), WCAP-16311-P (Reference 2, for the pressurizer surge line) and other applicable reports (for RHR and the accumulator lines) which were reviewed (and under review) and approved by the NRC.

2.1.6.2.4 Evaluation of RCP Casings

The Reactor Coolant Pump (RCP) casings analysis for the ASME Code Case N-481 is documented in WCAP-15873 (Reference 3) and was approved by the NRC (see Section 4.7.7 of the *Ginna License Renewal Safety Evaluation Report, NUREG-1786).* The effects of the EPU conditions on the Ginna Station RCP Casings have been evaluated. All six criteria of Code Case N-481 (fracture toughness, stress analysis, operating history, flaw postulation, flaw aspect ratios, and load comparisons) were met at EPU conditions. Therefore, it determined that the RCP casings loads continue to comply with the ASME Code Case N-481 requirements for the extended period of plant operation.

2.1.6.2.5 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the primary loop piping LBB and the RCP pump casings has been evaluated. The 40-year design transients and cycles (NUREG-1786) remain valid for the 60-year life of the plant and therefore the fatigue crack growth (FCG) analyses documented in WCAP-15837 (Reference 1) and in WCAP-15873 (Reference 3) remain valid for the EPU conditions. The current analyses used the end of life fracture toughness values due to the thermal aging effect on the cast stainless steel material and therefore fracture toughness values used in the current analyses are also valid for the EPU conditions. The aging evaluations documented in WCAP-15837 (Reference 1) for the primary loop piping and in WCAP-15873 (Reference 3) and approved by the NRC in the License Renewal Safety Evaluation Report (SER) Section 4.7.7 for the R. E. Ginna Station (NUREG -1786) remain valid for the EPU conditions.

Ginna has thus evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the primary loop piping LBB and the RCP pump casings analysis. The aging evaluations approved by the NRC in NUREG-1786 for the primary loop piping LBB and the RCP pump casings analyses remain valid for EPU conditions.

2.1.6.2.6 Results

The evaluation results demonstrated the following:

<u>Leak Rate</u> – A margin of 10.0 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 0.25 gpm

<u>Flaw Size</u> – A margin of 2.0 or more exits between the critical flaw size and the leakage flaw size

<u>Loads</u> – A margin of 1.0 (using faulted load combinations by absolute summation method) or $\sqrt{2}$ exists.

The evaluation results demonstrated that the LBB conclusions provided in current LBB analyses for the Ginna Station remain unchanged for the EPU conditions.

It is therefore concluded that the LBB acceptance criteria continue to be satisfied for the Ginna Station primary loop piping, pressurizer surge line, RHR and the accumulator lines at the EPU conditions. All the recommended margins continue to be satisfied and the conclusions shown in the current LBB analyses remain valid. It was therefore concluded that the dynamic effects of primary loop piping, pressurizer surge line, RHR and the accumulator lines pipe breaks need not be considered in the structural design basis of the Ginna Station at the EPU conditions.

The Ginna Station RCP Casings also continue to comply with the ASME Code Case N-481 requirements at the EPU conditions.

2.1.6.2.7 References

1. WCAP-15837, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant for the License Renewal Program, April 2002.

2. WCAP-16311-P, Rev. 0, Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant, August 2004.

3. WCAP-15873, A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R. E. Ginna Nuclear Power Plant for the License Renewal Program, May 2002.

4. Letter from D. G. Eisenhut, NRC, to R. W. Kober, RG&E, Subject: Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe breaks in PWR Primary Main Loops (Generic Letter 84-04), February 1, 1984.

5. Letter from D. C. Dilanni, NRC, to R. W. Kober, RG&E, Subject: Generic Letter 84-04, September 9, 1986.

6. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: IPSAR Section 4.13, Effects of Pipe Break on Structures, Systems and Components Inside Containment for the R. E. Ginna Nuclear Power Plant, June 28, 1983.

7. WCAP-12928, Structural Evaluation of the Robert E. Ginna Pressurizer Surge line, Considering the Effects of Thermal Stratification, May 1991.

8. Letter from G. S. Vissing (NRC) to R. C. Mecredy (RG&E) Subject: Staff Review of the Submittal by Rochester Gas and Electric Company to Apply Leak-Before-Break Status to Portions of the R. E. Ginna Nuclear Power Plant Residual Heat Removal System Piping, February 25, 1999.

9. SIR-99-036, Leak-Before-Break Evaluation of Portions of the Accumulator A and B Piping at R. E. Ginna Nuclear Power Station, June 1999, by Structural Integrity Associates.

10. Letter from J.A. Widay, Ginna, LLC to R.L. Clark, USNRC, "Fracture Mechanics Analysis per GDC-4; September 30, 2004.

11. Letter from T.A. Marlow, Ginna, LLC to D.M. Skay, USNRC, "Application of 10CFR50.90 Process for Use of Fracture Mechanics per GDC-4; May 28, 2005.

2.1.6.3 Conclusion of Leak-Before-Break

Ginna has reviewed the evaluation of the effects of the EPU conditions on the LBB analyses for the Ginna Station and determined that the changes in the primary system pressure and temperature range and the associated effects on the LBB analyses have been adequately addressed. Ginna further determined that the evaluations demonstrated that the LBB analyses will continue to remain valid following implementation of the EPU and that lines that credit LBB will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-4. Therefore, Ginna finds the EPU acceptable with respect to all aspects of LBB for the Ginna Station.

The Ginna Station RCP Casings will also continue to comply with the ASME Code Case N-481 requirements at the EPU conditions.

2.1.7 Protective Coating Systems (Paints) - Organic Materials

2.1.7.1 Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and from contamination by radionuclides and also provide wear protection during plant operation and maintenance activities. Ginna Nuclear Power Plant, LLC's (Ginna) review covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on 10CFR50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components. The criteria are also based on Regulatory Guide 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP, Section 6.1.2.

Ginna Current Licensing Basis

The application of Ginna coating systems inside the containment (and elsewhere in the plant) pre-dated Regulatory Guide 1.54, Rev. 0, and ANSI Standard N101.4. Ginna maintains the coatings inside the containment in accordance with the Protective Coatings Monitoring and Maintenance Program discussed in License Renewal Application section B2.1.24. All maintenance activities for the protective coatings inside the containment are performed in accordance with the Quality Assurance Program for Station Operations.

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station coating systems' design relative to conformance to:

- 10 CFR50 Appendix B is discussed in Ginna UFSAR sections 1.8.2.35 and 17.2.

- The Ginna Quality Assurance Program for Station Operation is discussed in UFSAR Section 17.2. The Ginna Station QA procedures' review and approval processes and administrative controls are in accordance with the requirements of 10CFR50 Appendix B. .

  The Quality Assurance Program for Station Operation states that, for new coatings and configuration changes to existing coatings, which have the potential to adversely affect a safety-related function, the quality assurance requirements of 10CFR50, Appendix B, in conjunction with engineering specifications, are used.

- Regulatory Guide 1.54, (Rev.0) is discussed in Ginna UFSAR section 1.8.2.32.

  The Ginna Station was constructed before Regulatory Guide 1.54 was issued. The Quality Assurance Program for Station Operation states that, for new coatings and configuration changes to existing coatings, which have the potential to adversely affect a safety-related function, the quality assurance requirements of 10CFR50, Appendix B, in conjunction with engineering specifications are used instead of the detailed requirements included in Regulatory Guide 1.54, Rev. 0, and its referenced standard, ANSI N101.4-1972. For surface preparation, application, and inspection the regulatory requirements of ANSI N101.2-1972 are referenced.

Other Ginna UFSAR sections that discuss coatings are:

- Ginna UFSAR section 3.8.1.6.5.5 discusses the liner painting materials and thicknesses (2.5 mils carbozinc 11 and 1.5 mils zinc chromate alkyd primer to spec TT-P-645A).

- Ginna UFSAR section 6.3.2.1.1 provides the response to Generic Letter 98-04 concerning commitments to ensure the Service Level I coatings inside the containment do not detach from their substrate during a design basis LOCA. The Ginna response to GL 98-04 has been accepted by the NRC [Reference: NRC Letter from Guy S. Vissing to RGE, Completion of Licensing Action for Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, July 14, 1998, R.E. Ginna Nuclear Power Plant (TAC No. MA4051)", November 19, 1999.].

- Details regarding protective coatings are also provided in Ginna UFSAR sections 6.1.2.4.2 (chlorides in ECCS), 6.1.2.6 (compatibility of coatings with post accident environment), 6.2.1.1.2 (containment design), and UFSAR Table 6.1-2 (protective coatings on carbon steel structures, equipment and concrete).

- SEP Topic VI-1, Organic Materials and Post Accident Chemistry, is discussed in UFSAR Section 6.1.2.9, Organic Material Evaluations. It states that the organic

materials used in the plant and described in Sections 6.1.2.6 (coatings) and 6.1.2.8 (other organics) are acceptable and will not interfere with the operation of engineered safety features under accident conditions, and that they will maintain their integrity and remain in serviceable condition after exposure to the severe environmental conditions of the design basis accident.

In addition to the evaluations described in the Ginna UFSAR, the coating systems were evaluated for the Ginna Station License Renewal.

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The Protective Coatings Monitoring and Maintenance Program in section B2.1.24 of the SER was provided as part of the discussion of programs used to manage aging effects.

In the License Renewal Application SER, section 3.0.3.12.7, the Ginna Coating System Monitoring and Maintenance Program was not credited as a stand alone aging management program. Degradation of coated base material is identified through inspections performed by the systems monitoring, ASME Section XI, Subsection IWE and IWL Programs, as well as the one time inspections of opportunity performed during routine maintenance. Where coatings are found degraded, they are repaired in accordance with the corrective action program.

2.1.7.2 Technical Evaluation

2.1.7.2.1 Introduction

The protective coating systems are discussed in Ginna UFSAR sections 1.8.2.32, (Regulatory Guide Compliance), 3.8.1.4.7.6 and 3.8.1.6.5.5 (Containment Liner), 6.2.1.1.2 (Supplementary Accident Criteria), 6.1.2.6 (Post Accident Environment Compatibility), and 6.3.2.1.1 (Recirculation of Containment Sump Water). Other sections address painting for radiation decontamination (section 12.3.5.2) and for HEPA filters compatibility (section 6.5.1.2.2.4).

The coating system function is to provide corrosion and erosion control and to facilitate decontamination in the event of radioactive material leakage into the containment (or other buildings).

Failure of coating systems inside the containment before and/or during a design bases accident that requires the recirculation of water from the containment sump could degrade or fail the accident response systems' functions by clogging the sump screens. NRC Generic Letter 98-04, required plant operators to address the qualification of the coating systems inside containment. The Ginna response stated the Ginna dose due to a severe DBA is less than 1E9 Rads, with many areas inside containment receiving a dose of less than 1E7 Rads.

Resolution of GL 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, which discusses the clogging of containment emergency sumps, is ongoing between Ginna and the NRC and is not addressed as a power uprate issue in this report.

The Protective Coating Monitoring and Maintenance Program includes all coatings inside the containment. The program ensures preventive actions are taken to remove and replace degraded paint identified during monitoring inspections. Corrective actions are performed in accordance with 10CFR50 Appendix B procedures. The procurement, application and maintenance requirements of 10CFR50 Appendix B are implemented through specification of appropriate technical and quality requirements for the Service Level 1 coatings program.

2.1.7.2.2 Acceptance Criteria

The performance of coating systems is affected by environmental aging parameters such as normal service temperature; peak accident temperature; post accident operation time, temperature and total service; normal operation and accident radiation doses; as well as by environmental and other conditions existing during the surface preparation and application. The coating system performance under chemical spray conditions is evaluated.

The coating systems inside the containment are controlled by Ginna's QA program . They are the only coating systems potentially impacted by the EPU due to EPU changes in environmental parameters.

Coating systems that meet the requirements of the plant coatings' technical specifications, and that are treated using processes in accordance with 10CFR50 Appendix B quality assurance requirements, are qualified when the EPU parameters are within the conditions for which performance of the coating system is demonstrated to be acceptable. The coating systems are inspected and their condition monitored to determine when appropriate actions for replacement of problem areas are needed.

2.1.7.2.3 Description of Analyses and Evaluations

The coatings were evaluated by comparing the EPU parameters with the qualification acceptance parameters under which the coating systems were previously accepted by the NRC.

The normal operating temperature is not significantly impacted by the EPU and remains below the current containment temperature limit.

The containment peak accident temperature for the EPU is less than the current EQ envelope temperature. The long-term post accident temperature is higher at 24 hours for a short time after which it is lower than the envelope. Evaluation of material compatibility to the post accident design basis environment (286°F accident peak and

152°F for 30 days) was made for the Ginna coating system of Carbozinc-11 primer and Phenoline 305 top coat. Industry tests of these coatings have been made (to 320°F accident peak and 150°F to 175°F for 60 days) and successfully passed two accident transients without deterioration or loss of adhesion.

EPU containment pressure remains below the qualification pressure of the equipment and protective coating systems. Refer to LR section 2.3.1, Equipment Qualification of Electrical Equipment and 2.6, Containment Review Considerations for additional discussions of the containment accident environments.

The total integrated radiation dose increases for EPU conditions but is bounded by the coating system qualification dose levels. The dose increase inside the containment is discussed in LR section 2.10.1, Occupational and Public Radiation Doses. The maximum scaling factor used to develop the dose increase at EPU power levels and 18 month fuel burn-up is 37%. The total integrated dose for the EPU including the 60 year operating time dose, the accident, and the post accident dose over 1 year is 2.93E8 Rads. This is less than the 1E9 Rads dose level referenced in the Ginna response to GL 98-04.

As discussed in LR section 2.3.1 (Equipment Qualification of Electrical Equipment) the post accident pH of containment spray remains below 10.5 following implementation of EPU Therefore the EPU has no impact on the coatings within the containment from exposure to chemical spray.

2.1.7.2.4 Results

The containment temperature, pressure and chemical spray conditions associated with EPU normal operation, design basis accident conditions and post accident operations are within the coating qualification conditions currently approved by the NRC.

The NRC accepted the Ginna response to GL 98-04. The response included the radiation dose comparison of the Ginna dose being less than the 1E9 Rads, with many areas inside Containment receiving a dose of less than 1E7 Rads. Therefore, there is reasonable assurance the EPU radiation dose of 2.93E8 Rads does not exceed the known radiation resistance of the Ginna coatings inside the Containment.

License renewal did not result in any changes to the Protective Coatings Monitoring and Maintenance Program because it is an ongoing maintenance activity rather than an age management program.

2.1.7.3 Conclusions

The evaluation of the effects of the EPU on protective coating systems addressed the impact of changes in conditions following a LOCA and their effects on the protective coatings. Ginna concludes that the protective coating systems inside containment will continue to be acceptable and meet their safety related functions following

implementation of the proposed EPU. The coating systems applied under the monitoring and maintenance program will continue to meet the requirements of 10CFR50, Appendix B.

2.1.8 Flow-Accelerated Corrosion

2.1.8.1 Regulatory Evaluation

Flow-accelerated corrosion is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to flow-accelerated corrosion, and flow-accelerated corrosion is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to flow-accelerated corrosion depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing flow-accelerated corrosion effects, in most cases, cannot be achieved. Loss of material by flow-accelerated corrosion will, therefore, occur. Ginna Nuclear Power Plant, LLC (Ginna) has reviewed the effects of the proposed EPU on flow-accelerated corrosion and the adequacy of the flow-accelerated corrosion program to predict the rate of loss so that repair or replacement of damaged components will be made before they reach critical thickness. The licensee's flow-accelerated corrosion program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by flow-accelerated corrosion.

The evaluation will also include the Ginna Erosion/Corrosion Program, where the FAC program is a subprogram to the E/C Program.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. As addressed in Ginna UFSAR section 10.7.8, Ginna has developed an erosion/corrosion program for single and two-phase systems consistent with the requirements of NUREG 1344 and the NUMARC Working Group on Piping Erosion/Corrosion Summary Report, dated June 11, 1987. The program is designed to ensure that erosion/corrosion does not result in unacceptable degradation of the

structural integrity of carbon steel piping systems. The program is documented in the Ginna Station Erosion/Corrosion Program Manual and includes the following:

- Frequency of inspection criteria
- Acceptance criteria
- Inspection/expansion criteria
- Repair/replacement criteria
- Corrective action

The Ginna Erosion/Corrosion Program is responsive to Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," and implements the guidelines in EPRI Report, NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program." Other source / development documents include NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants" and NRC Information Notice 93-21, "Summary of NRC Observations Compiled During Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs."

In addition to the evaluations described in the Ginna UFSAR, the Flow-Accelerated Corrosion Program was evaluated for the Ginna Station License Renewal. The evaluations are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The Flow-Accelerated Corrosion Program is addressed in Section 3.0.3.6 of the License Renewal Safety Evaluation Report.

2.1.8.2 Technical Evaluation

2.1.8.2.1 Introduction

This section addresses the following Erosion/Corrosion Program topics:

- Program scope and attributes
- Piping / component inspection
- Evaluation of inspection data
- Component repair / replacement
- Service water E/C program
- Small bore E/C program
- Pressure vessel E/C program

Erosion/Corrosion (E/C) Program Scope / Attributes

The scope of the Erosion/Corrosion Program encompasses piping, pressure vessels, and storage tanks containing both single phase and two phase fluids. Specifically, the following systems and equipment are within the scope of the Ginna Erosion/Corrosion Program:

- Carbon steel piping systems subject to flow-accelerated corrosion (FAC)
- Tanks and heat exchangers connected to piping systems within the scope of the E/C Program
- Service water system
- Buried radioactive piping
- Bulk chemical storage tanks and connected fill and delivery piping as defined in New York State regulations
- Portions of systems and components that have been found to exhibit erosion based on operating experience.

The activities associated with ensuring that flow-accelerated corrosion does not result in unacceptable degradation of the structural integrity of carbon steel piping systems are included in the FAC subprogram (herein referred to as the "FAC program"). The elements / activities associated with the FAC program are identified and addressed in this section.

All piping and equipment for the systems identified above are considered susceptible to erosion/corrosion unless excluded by one of the following screening criteria:

- Piping material

  Piping systems with stainless steel and low-alloy with nominal chromium content equal to or greater than 1-1/4% material can be excluded from the FAC program. However, considering mechanical degradation mechanisms in piping systems, 2% or higher chromium content in material was conservatively adopted in the Ginna FAC Program procedures as the exclusion criterion. It should be noted that even traces of chromium (0.04% and higher) in material content reduces wear-rate levels significantly in piping systems susceptible to flow-accelerated corrosion. Therefore, alloy analyses were performed to determine the chromium percentage in material composition on all components covered by the FAC Program that are inspected with ultrasonic testing technique. Measured chromium percentages were imported to CHECWORKS to improve program analysis results.

- Piping system content

The FAC phenomenon occurs in de-oxygenated clean water typically found in the secondary system piping. Because superheated or "dry" steam conditions have been shown not to cause erosion/corrosion regardless of temperature or pressure levels, the main steam lines between the steam generator and the turbine are excluded from the FAC program. However, periodic NDE inspections are performed on main steam piping to confirm acceptable conditions.

Service water is lake water that is not clean or de-oxygenated. However, the service water system is susceptible to numerous erosion/corrosion mechanisms. Accordingly, the service water system is included in the E/C program.

- System usage

  With defined exceptions (e.g., portions of low usage piping systems with flashing flow conditions when used), systems which see less than 2% plant operating time can be excluded from the FAC program.

- System operating conditions

  All single phase systems with operating temperatures less than or equal to 212°F are excluded from the FAC program. The 212°F temperature criteria does not apply to two phase systems, and it does not apply to the service water system.

- Industry / Ginna experience

  Systems known to be susceptible to erosion/corrosion based on industry or Ginna experience are included in the E/C program regardless of other exclusion criteria. This includes erosion/corrosion-susceptible low temperature or low usage lines which would be otherwise excluded.

- Pipe size

  Piping with a nominal diameter greater than 2 inches is classified as large bore piping. Piping with a nominal diameter greater than or equal to ¾ inch and less than or equal to 2 inches is classified as small bore piping. With the exception of piping in the service water system, piping with a nominal diameter less than ¾ inch is excluded from the E/C program.

The following large bore systems have been found to be susceptible to FAC through the screening process and are therefore monitored in the FAC program:

- Feedwater
- Condensate
- Heater drains

- Moisture separator reheater drains
- Steam generator blowdown
- Extraction steam
- Gland Steam

The following small bore systems / lines are monitored in the small bore E/C program:

- Main steam drains
- Heater drains
- Gland steam
- MSR excess steam
- Valve warm-up lines

The service water system is monitored in the service water E/C program.

The pressure vessel E/C program includes the following tanks and heat exchangers:

- Steam generator blowdown tank
- Heater drain tank
- Preseparator drain tanks
- Low pressure feedwater heaters
- High pressure feedwater heaters
- Moisture separator reheaters
- Spent fuel pool heat exchangers

Piping / Component Inspection

A component in a susceptible piping system is selected for inspection if any one of the following considerations is applicable:

- The CHECWORKS analysis indicates a high wear-rate or low predicted remaining life (time remaining until t_{min}, the calculated minimum allowable wall thickness).

- The component is susceptible to erosion/corrosion based on previous industry and/or Ginna experience, or judgment of the responsible erosion/corrosion engineer.

- The component is selected as a result of sample expansion based upon the inspection results at other locations.

- The component requires re-inspection based on previous inspection results and projected time to degrade to t_{min}.

These items are described in the following subsections:

CHECWORKS

The CHECWORKS computer code, Version 1.0G, developed by EPRI, is used to evaluate piping systems susceptible to flow-accelerated corrosion mechanisms. Most of the lines modeled are large bore lines; however, small bore pipes are also modeled in system segments. Parameters including operating conditions, routing geometry, piping properties, and specifications are inputted for CHECWORKS program evaluations.

The primary objective of the CHECWORKS Program evaluation is to ensure that the inspections are focused on the areas most likely to be experiencing degradation so the likelihood of catastrophic failures is minimized. Results from this analysis identify recommended initial locations that should be inspected to determine the condition of the piping system as a whole. Data derived from the inspection of the recommended initial locations is used to develop a plant-specific erosion/corrosion model of susceptible piping. Using the plant specific model, CHECWORKS refines the ranking locations and provides quantified estimates of erosion/corrosion rates and times until minimum code allowable wall thicknesses will be reached. The CHECWORKS models are updated based on periodic NDE inspections.

Table 2.1.8-2 provides a comparison of wall thickness predicted by CHECWORKS with the measured wall thickness for several representative components from the various systems that are included in the Ginna Erosion/Corrosion Program.

Industry Experience Locations

Components which have displayed a susceptibility for erosion/corrosion in other power plants are given consideration in the inspection point selection process. These "industry experience" points are typically downstream of components which cause flow restriction or otherwise add turbulence (e.g., downstream of control valves and orifices). A review of the inspection results is performed to assure plant-specific experience has been included in the selection of components for examination.

Ginna Experience Locations

The Ginna Station maintains record of piping components that have experienced erosion/corrosion degradation which required past repair or replacement. These locations are considered in the E/C Program. Requiring continuing inspection at these points depends on the replacement material used.

Engineering Based Selections

Piping systems that are not analyzed using CHECWORKS have inspections at locations based upon engineering experience. A conservative number of locations is specified to ensure that any significant variations in susceptibility are represented and bounded. CHECWORKS and non-CHECWORKS selections are based on the following:

- Steam cycle and service water systems with similar flow and operating conditions may be grouped together. Each group is considered separately when determining required inspections.

- Historical inspection data available for each system is considered.

- In each sub-system, the number of changes in design pressure, pipe diameter, and pipe class is determined. A sufficient sample of points is selected to ensure that each segment has been suitably inspected.

- A qualified engineer experienced in the use of CHECWORKS reviews isometric drawings or performs walkdowns to determine inspection locations.

- In addition to the inspection of "standard" components, special consideration is given to components known to be particularly susceptible (e.g., control valves, component discharge nozzles, orifices, steam traps).

- Databases are used to trend inspection results and evaluate wear rates for non-CHECWORKS systems.

Inspection Techniques

Ultrasonic testing (UT) provides the most quantitative wear trend data of a component to assess wall thickness and predict future wear. Therefore, UT is the primary non-destructive examination method used for detecting pipe wall thinning. For locations where ultrasonic testing cannot provide adequate results, alternate methods may be used, including radiographic testing, visual, and video examinations.

Evaluation of Inspection Data

Evaluations of UT data to establish the component's initial thickness and determine the measured wear is very complex. Complications may include: not knowing the initial thickness of component, counter-bore, backing ring, bad measured reading, obstructions such as pipe hangers, lugs, etc. There are four methods that can be used to calculate wear of piping components in the CHECWORKS program by using single and multiple measured inspection data:

- Point to Point – where multiple thickness measurements are available.
- Band Method
- Area Method
- Moving Blanket Method

Sample expansion is the process by which additional components are selected for inspection due to the detection of erosion/corrosion-related wall thinning exceeding specified limits. The purpose of this effort is to determine the extent of located thinning.

Components which have been inspected are re-examined on a schedule consistent with the calculated component life (time to t_{min}). Re-examination is scheduled for the refueling outage preceding the time when t_{min} will be reached.

Component Repair / Replacement

The process / criteria for determining repair / replacement of a component is outlined as follows:

- The minimum wall thickness, t_{min}, is determined, based on the hoop stress due to internal design pressure and/or longitudinal stress due to internal design pressure plus bending moment due to dead weight.

  The following factors are considered in the calculation of t_{min}:

  - Minimum wall thickness based on code requirements (ANSI B31.1, 1973 Edition with Summer 1973 Addendum / ASME B&PV Code, 1986 Edition)
  - Current wear rate
  - Effective full power years (EFPY) until the next refueling outage
  - Line correction factor for the CHECWORKS analysis

- The component is inspected; wall thickness is measured by nondestructive examination methods (t_{meas}).
- If the measured wall thickness is greater than 87.5% of the nominal wall thickness, the component is acceptable for continued service.
- If t_{meas} is less than t_{min}, an Action Report is generated and a structural integrity evaluation of the component is performed. Elements of the component structural evaluation include:
  - Using the higher value of the wear rates determined by (1) CHECWORKS and (2) measured data, the predicted minimum wall thickness at the end of the future service period, usually the next refueling outage, is calculated.
  - If the predicted minimum wall thickness is less than or equal to 30% of the nominal wall thickness, the component is rejected and actions are taken to repair/replace it. Additional components are required to be inspected to determine the extent of the wall thinning (sample expansion).
  - For instances where the predicted minimum wall thickness is less than or equal to 87.5% of the nominal wall thickness and greater than 30% of the nominal wall thickness, the structural evaluation is continued.

- If the results of the component structural evaluation are not acceptable, actions are taken to repair / replace the component.

Note: Although the E/C Program addresses analytical methods of evaluation, including use of ASME Code Cases N-480 and N-597, for conservatism, degraded components are replaced and/or repaired prior to invoking these analytical evaluations.

The existing criteria for repair / replacement of piping are consistent with the guidelines in EPRI Report, NSAC-202L-R2.

Service Water E/C Program

The elements of the E/C Program described above are applicable to the service water system, unless specific exception is indicated. The service water E/C program is intended to evaluate erosion/corrosion degradation mechanisms affecting the reliability of the safety and non-safety related portions of the service water system. Aspects of the E/C Program which apply specifically to the service water system include the following:

- The service water system corrosion mechanisms (e.g., pitting, microbiologically induced corrosion) are quite varied and less amenable to erosion/corrosion software prediction tools such as CHECWORKS. E/C of service water systems is plant specific and requires that site specific inspection data be collected and studied to understand and predict wear. As a result, Ginna is collecting inspection results and building a database that allows identification of areas most susceptible to erosion/corrosion mechanisms and prioritization of inspection efforts.

- Both ultrasonic testing and radiographic examination techniques are used in the service water system.

- Service water corrosion can be accelerated in stagnant flow areas where debris, silt bio-fouling, and/or scale has collected. This buildup can often be visually identified during maintenance activities and/or radiographic examinations. One aspect of the service water E/C program is to record this visual inspection data and incorporate this data into the service water erosion/corrosion database. This database is a repository for inspection results, design information for inspected components, minimum wall requirements for components to be inspected, and corrosion rates for inspected components. Recording the data allows trending of component wall thicknesses, fouling mechanisms, and corrosion mechanisms.

Small Bore E/C Program

The elements of the E/C Program described above are applicable to small bore piping, unless specific exception is indicated. Aspects of the E/C Program which apply specifically to small bore piping include the following:

- Piping systems used for instrumentation or instrument sensing lines are excluded from the small bore E/C program.

- Lines which are not required for plant operation and which can be isolated after a postulated erosion/corrosion failure are excluded from the small bore E/C program.

- Both ultrasonic testing and radiographic examination techniques may be used for small bore inspections. The majority of small bore inspections are performed using radiographic testing, since it can be performed while on-line and it provides adequate inspection information for small bore piping.

- A database is maintained to manage in-scope small bore piping information, such as size, material, pressure, temperature, and inspection results.

Pressure Vessel E/C Program

The elements of the E/C Program described above are applicable to the tanks and heat exchangers included in the pressure vessel E/C program, unless specific exception is indicated. Aspects of the E/C Program which apply specifically to the pressure vessel E/C program include the following:

- Eddy current inspection of heat exchanger tubing is outside the scope of the program.

- The high pressure turbine crossunder piping to the moisture separator reheaters is included in the pressure vessel E/C program scope.

- The extraction steam lines to the No.1, No. 2, and No.3 low pressure feedwater heaters are included in the pressure vessel E/C program scope.

- Tanks and heat exchangers erosion/corrosion is managed on a component specific basis. Examination methods are ultrasonic testing or visual inspection. Based on the inspections, the level of erosion/corrosion is characterized as insignificant, slight, moderate, or severe.

2.1.8.2.2 Description of Analyses and Evaluations

As addressed in LR section 2.5.5.1, Main Steam System, the EPU will result in small changes in operating pressures and temperatures, and an approximately 18% increase in flow rates / velocities in the main steam lines from the steam generators to the high pressure turbine stop valves. The highest calculated velocity at EPU conditions is well below the industry guidelines. As indicated in Section 2.1.8.2.1 above, because superheated or dry steam conditions have been shown not to cause erosion/corrosion regardless of temperature or pressure levels, the main steam lines between the steam generator and the turbine are excluded from the FAC program. Because dry steam

conditions exist at the EPU power level, this conclusion will remain valid after the EPU is implemented.

As addressed in LR section 2.5.5.1, the EPU will result in changes in operating pressures and temperatures in the main steam lines to the moisture separator reheaters, the reheat steam from the high pressure turbine to the moisture separator reheaters (cold reheat / crossunder piping), and the reheat steam from the moisture separator reheaters to the low pressure turbines (hot reheat / crossover piping). The calculated velocities at EPU conditions for these lines are shown to be essentially unchanged from current conditions; the highest calculated velocity is in the crossover piping and is well below the industry guideline. The crossover piping has not experienced any significant erosion/corrosion due to the dry steam conditions at the current power level; there is no change in steam conditions in this piping at the EPU power level.

As identified in Section 2.8.1.2.1 above, the crossunder piping is included in the Erosion/Corrosion Program. This piping has been subjected to localized elevated mechanical and chemical degradation mechanisms that were caused by the initial repair process of stainless steel weld-overlays on degraded areas. Currently, during every outage, thorough visual inspection is performed, and like-for-like weld material is used for repair of degraded areas. The EPU velocity in this piping is only slightly higher than at the current operating conditions. The moisture content is relatively unchanged from current conditions. Therefore, there is no change in the potential for erosion / corrosion in this piping for EPU conditions. Monitoring per the E/C Program will be continued after the EPU.

For lines in the following systems, the EPU will result in changes in operating pressures, temperatures, and fluid flow velocities and affect steam quality in the applicable lines:

- Extraction steam
- Heater drains, including moisture separator reheater drains
- Condensate and Feedwater (LR section 2.5.5.4, "Condensate and Feedwater System")

Prior to implementing the EPU all of the Ginna CHECWORKS models will be updated to incorporate operating pressures and temperatures, fluid velocities, and steam quality data derived from the EPU heat balances.

As indicated in Section 2.8.1.2.1 above, single phase systems with operating temperatures less than or equal to 212°F are excluded from the FAC program. A review of lines in the systems listed above was performed to determine if, at EPU conditions, the operating temperature of any single phase lines / components not currently in the FAC Program increased from below to above 212°F, the temperature threshold for FAC susceptibility. It was determined that the condensate piping from feedwater heaters 2A / 2B outlet nozzles to feedwater heaters 3A / 3B inlet nozzles increased from ~208°F at current conditions to ~217°F at EPU conditions. This piping and nozzles are not currently included in the FAC Program. Therefore, the feedwater heaters 2A / 2B outlet

nozzles, the feedwater heaters 3A / 3B inlet nozzles, and the condensate piping between feedwater heaters 2A / 2B and 3A / 3B will be added to the FAC Program Due to this change, the feedwater heater 2B outlet nozzle to feedwater heater 3B inlet nozzle was inspected during the 2005 refueling outage.

For lines in the systems listed above a comparison was also made of the calculated piping / component (e.g., nozzle) velocities with industry standard velocity criteria as a measure of whether there is an increased potential for erosion/corrosion.

> Note: It has been shown that an increased fluid velocity results in an increase in wear rate. The wear rate predictions of piping components from CHECWORKS analyses include all operating parameters, such as temperature, steam quality, fluid velocity, that are also affected at elevated levels by the EPU. Based on the influence a particular parameter (e.g., steam quality) has on wear rate, the change in that parameter due to the EPU may be the dominant factor affecting the resulting wear rate of a line / component.

The majority of the line / component velocities were found to be within the industry standard velocity criteria. A summary of the results of the review of lines / components with EPU velocities exceeding industry standard velocity criteria follows:

- Affected lines in the extraction steam system are included in the E/C Program. Affected components (turbine nozzle outlet expanders) in the extraction steam system are included in the FAC program. Monitoring per these programs will be continued after the EPU.

- Affected single phase lines in the heater drains system are included in the FAC program. Monitoring per the FAC program will be continued after the EPU.

- Affected lines in the condensate and feedwater system, except the condensate polisher bypass line and the condensate cooler / hydrogen cooler bypass line, are included in the FAC program. Monitoring per the FAC program will be continued after the EPU. Although velocity in the condensate polisher bypass line and the condensate cooler / hydrogen cooler bypass line exceeds industry criteria, the temperature of these lines at EPU conditions is below the 212°F temperature threshold for FAC susceptibility. Therefore, these lines remain excluded from the FAC program.

- Feedwater heaters (FWHs) 1A / 1B and 2A / 2B extraction steam inlet nozzles are included in the E/C Program. Non-destructive examination (NDE) will be performed on either the FWHs 1A and 2A or FW Hs 1B and 2B extraction steam inlet nozzles in the 2006 refueling outage]. Based on NDE results, expansion of inspection to the other FWH nozzles may be performed. Monitoring per the E/C Program will be continued after the EPU.

- Feedwater heaters 1A / 1B shells are included in the E/C Program. Non-destructive examination is planned to be performed on either FWH 1A or 1B inlet shell in the 2006 refueling outage. Based on NDE results, expansion of

inspection to include other low pressure FWHs is possible. Monitoring per the E/C Program will be continued after the EPU.

- Feedwater heaters 1A / 1B, 4A / 4B, and 5A / 5B drain inlet nozzles are included in the FAC Program. The FWH 4A drain inlet nozzle was previously replaced and inspected, with acceptable results. The FWH 4B drain inlet nozzle material is chrome-moly steel, which is very resistant to FAC mechanisms. NDE is to be performed on selected drain inlet nozzles of FWHs 1A / 1B, 2A / 2B, and 5A / 5B in RFO2006 [Commitment – CATs Item 12552]. Based on NDE results, expansion of inspection to include additional drain inlet nozzles in these FWHs is possible. Monitoring per the FAC program will be continued after the EPU.

- Drain outlet nozzles of all feedwater heaters are included in the FAC Program. Non-destructive examination was performed on FWH 4A drain outlet nozzle and downstream piping in the 2005 refueling outage. Based on results of this examination, expansion of inspection to FWH 4B drain outlet nozzle was not performed. Monitoring per the FAC program will be continued after the EPU.

- Feedwater heaters 1A / 1B condensate / feedwater inlet nozzles are not currently included in the FAC Program. These nozzles are part of single-phase lines which at EPU conditions are below the 212°F temperature threshold for FAC susceptibility,. Therefore, these nozzles remain excluded from the FAC program. However, since the velocity in these nozzles exceeds industry criteria, these nozzles will be monitored per the E/C Program after the EPU.

- Feedwater heaters 2A / 2B condensate / feedwater outlet nozzles, and feedwater heaters 3A / 3B condensate / feedwater inlet nozzles are not currently included in the FAC Program. As addressed above, these nozzles will be added to the FAC Program, based on increase in the operating temperature from below 212°F at current plant conditions to above 212°F at EPU conditions. These nozzles will be inspected in the first refueling outage after implementation of the EPU.

- Feedwater heaters 3A / 3B condensate / feedwater outlet nozzles, feedwater heaters 4A / 4B condensate / feedwater inlet and outlet nozzles, and feedwater heaters 5A / 5B condensate / feedwater inlet and outlet nozzles are included in the FAC Program. Monitoring per the FAC program will be continued after the EPU.

As indicated in Section 2.1.8.2.1 above, the steam generator blowdown system is included in the FAC Program. As addressed in Ginna UFSAR Section 10.7.5.1, to reduce the erosion/corrosion potential and permit periods of increased blowdown flow, the blowdown pipe size was increased from 2 inches to 3 inches. Portions of the system have been replaced with piping / components fabricated from chrome-moly steel. Up through 2002 Ginna operated with a normal operating continuous steam generator blowdown flow of ~100 gpm per steam generator. Since 2002 Ginna has operated with a reduced continuous steam generator flow of 40-80 gpm per steam generator without any detrimental impact on secondary side chemistry. As addressed in LR section 2.1.10, Steam Generator Blowdown System, the increased steam and feedwater flow rates at

EPU conditions do not significantly affect the concentration of impurities throughout the turbine cycle.

Although steam generator blowdown flow may need to be slightly increased from the present operating range due to the EPU operating conditions, the required steam generator flow will be less than the 100 gpm flow per steam generator used historically prior to 2002. Therefore, there are no changes to the design steam generator blowdown rates as a result of the EPU. Since the blowdown flow is not changed significantly for the EPU, the velocities in the blowdown lines are not significantly affected by the EPU. There is only a small change in steam generator outlet temperature at EPU conditions. Accordingly, no appreciable increased potential for flow accelerated corrosion exists with the steam generator blowdown system at EPU conditions. Monitoring of the steam generator blowdown system per the requirements of the FAC program will continue after the EPU.

As addressed in LR section 2.5.4.2, Service Water System, service water system pressures and flow rates are not affected by the EPU; temperature will increase slightly for normal system operating modes. Accordingly, there is no increased potential for erosion/corrosion in service water piping / components at EPU conditions. Monitoring and inspection of the service water system in accordance with the E/C Program will continue after the EPU.

The EPU will result in an increase in steam flow and velocities in the HP Turbine gland steam leak-off piping due to the higher HP turbine exhaust pressure. Since the gland steam piping is periodically monitored as part of the existing E/C Program, any increase in long term wear of the piping due to EPU will be identified by the existing monitoring program.The EPU may result in increased flow rates in some small bore piping, e.g., main steam drains. As discussed in Section 2.1.8.2.1 above, monitoring and inspection of small bore lines is based on engineering experience and judgment. Monitoring / inspection of small bore lines in accordance with the E/C Program will continue after the EPU.

The EPU does not affect erosion/corrosion aspects of buried radioactive piping or bulk chemical storage tanks and connected piping.

An assessment of the potential for erosion/corrosion of the moisture separator reheaters at EPU conditions has been performed. Based on inspections of similarly designed moisture separator reheaters at other plants and on a technical analysis, which considered the affects of temperature, velocity, moisture, and flow configuration, it was concluded that significant erosion/corrosion will not occur in the moisture separator reheaters at EPU conditions. Ginna will continue to inspect the shells and nozzles of the moisture separator reheaters before and after the EPU to confirm this conclusion

As addressed in LR section 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System, the EPU does not affect the flow rates of the spent fuel pool cooling system. Accordingly, the EPU does not affect erosion/corrosion aspects of the spent fuel pool heat exchangers.

Monitoring of the heater drain tank and preseparator drain tanks per the E/C Program will be continued after the EPU. Any changes in wear due to the EPU would be identified by the periodic inspections.

For several representative components which are susceptible to flow-accelerated corrosion, Table 2.1.8-1 provides a comparison of the wear rates for both the existing full power plant conditions and the EPU conditions as analytically calculated by the EPRI CHECWORKS Program. Although the actual wear rates measured in the plant can differ from those calculated by CHECWORKS, the percent change in the CHECWORKS results can be used to estimate the expected change in actual component wear rates due to implementation of the EPU. Table 2.8.1-2 provides the NDE measured component wear rates; and, also provides a comparison of predicted wall thicknesses versus measured wall thicknesses for all of the components listed in Table 2.1.8-1.

Elements of the Erosion/Corrosion Program, including the FAC program, described in Section 2.1.8.2.1 above, including the component repair / replacement process and criteria, will continue to be utilized following the EPU.

For plant modifications associated with the EPU (e.g., modifications to moisture separator reheater 2nd and 4th pass drain piping), impact of the modifications on the Erosion/Corrosion Program will be addressed as part of the plant change process. For new components and affected preexisting components, their satisfaction of the E/C Program's inclusion/exclusion criteria following EPU will be checked, and they will be subject to the program depending on those findings.

Evaluation of Impact on Renewed Plant Operating License Evaluations and Licensing Renewal Programs

The License Renewal Safety Evaluation Report concludes that the Ginna Flow-Accelerated Corrosion Program is consistent with the requirements of the "Generic Aging Lessons Learned (GALL) Report", NUREG1801. Changes to the piping/equipment included in the Erosion/Corrosion Program as a result of the EPU are within the scope of the existing program and in compliance with program criteria. Therefore, the EPU does not affect the evaluation / conclusions in the License Renewal Safety Evaluation Report regarding the flow accelerated corrosion program, and no new aging effects requiring management are identified.

2.1.8.3 Conclusions

Ginna has evaluated the effect of the proposed EPU on the flow-accelerated corrosion analysis for the plant and concludes that the impact of changes in the plant operating conditions on the flow-accelerated corrosion analysis have been adequately addressed. Ginna further concludes that it has been demonstrated that the updated analyses will predict the loss of material by flow-accelerated corrosion and will ensure timely repair or

replacement of degraded components following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to flow-accelerated corrosion.

Table 2.1.8-1

Comparison of Current and EPU Analytical Wear Rates

System	Line Description	Component ID	Component Geometry	Pipe Specification	CHECWORKS Current Wear-Rate 100% Power (mils/year)	CHECWORKS Current Wear-Rate 117% (EPU) Power (mils/year)	Increase in Wear Rate Due to EPU
Main Feedwater	FW Pump Discharge to #5 FW Heater	M1-19	Pipe	14" Sch 100	0.975	1.141	17.03%
Main Feedwater	#5 FW Heater Outlet to FW Control Valve	M2-35	Elbow	20" Sch 100	0.489	0.573	17.18%
Heater Drain	#4 FW Heater Drain to HD Tank	M3-014A	Pipe	12" Sch 80	0.950	1.178	24.00%
Condensate	#4 FW Heater Outlet to FW Pump Suction	M5-27B	Pipe	18" Sch Std	0.224	0.263	17.41%
Heater Drain	HD Tank Outlet to HD Pump Suction	M7A-10	Pipe	10" Sch 40	0.603	0.702	16.42%
MSR Drain	MSR 2nd Pass Drain to #5 FW Heater	M11A-75	Pipe	6" Sch 80	0.575	0.629	9.39%
Extraction Steam	Pre-separator Tank to #4 FW Heater	M21-41	Pipe	16" Sch Std	4.924	5.401	9.69%
Heater Drain	#5B FW Heater Drain to #4B FW Heater	M41B-44	Pipe	6" Sch 80	0.334	0.343	2.69%
Heater Drain	Pre-separator Tank Drain to HD Tank	M45-22	Pipe	6" Sch 80	1.012	1.252	23.72%
Main Feedwater	FW Control Valve to S/G 1A	M84-11	Elbow	14" Sch 100	1.515	1.773	17.03%

TABLE 2.1.8-2

Comparison of Predicted and Measured Wall Thickness

Line Description	Comp. ID	Pipe Specification	Nominal Thickness or NDE Measurement	CHECWORKS Current Wear-Rate (mils/year)	CHECWORKS Line Correction Factor	Predicted Thickness[1]	NDE (UT or RT) Measured Thickness	NOTES
FW Pump Discharge to #5 FW Heater	M1-19	14" Sch 100	$t_{m\text{-}RFO1989}$=0.907"	1.1	0.293	$t_{p\text{-}RFO2005}$=0.848"	$t_{m\text{-}RFO2005}$=0.897"	0.17% $Cr_{RFO2005}$
#5 FW Heater Outlet to FW Control Valve	M2-35	20" Sch 100	$t_{m\text{-}RFO1989}$=1.325"	1.3	0.628	$t_{p\text{-}RFO2003}$=1.296"	$t_{m\text{-}RFO2003}$=1.292"	0.077% $Cr_{RFO2003}$
#4 FW Heater Drain to HD Tank	M3-14A	12" Sch 80	$t_{nom\text{-}1970}$=0.688"	1.1	0.655	$t_{p\text{-}RFO2005}$=0.637"	$t_{m\text{-}RFO2005}$=0.672"	No base line, 0.06% $Cr_{RFO2005}$
#4 FW Heater Outlet to FW Pump Suction	M5-27B	18" Sch Std	$t_{nom\text{-}1970}$=0.375"	0.2	0.147	$t_{p\text{-}RFO2005}$=0.327"	$t_{m\text{-}RFO2005}$=0.309"	Mihama Event, 0.11% $Cr_{RFO2006}$
HD Tank to HD Pump Suction	M7A-10	10" Sch 40	$t_{nom\text{-}1970}$=0.365"	0.5	0.192	$t_{p\text{-}RFO2003}$=0.340"	$t_{m\text{-}RFO2003}$=0.336"	No base line, 0.031% $Cr_{RFO2003}$
MSR 2nd Pass Drain to #5 FW Heater	M11A-75	6" Sch 80	$t_{nom\text{-}1970}$=0.432"	0.7	1.298	$t_{p\text{-}RFO2003}$=0.414"	$t_{m\text{-}RFO2003}$=0.403"	No base line & Cr measurements
Preseparator Tank to #4 FW Heater	M21-41	16" Sch Std	$t_{nom\text{-}1970}$=0.375"	4.7	0.899	$t_{p\text{-}RFO2005}$=0.192"	$t_{m\text{-}RFO2005}$=0.217"	No base line, 0.06% $Cr_{RFO2003}$
#5B FW Heater Drain to #4B FW Heater	M41B-44	6" Sch 80	$t_{nom\text{-}1970}$=0.432"	0.3	0.147	$t_{p\text{-}RFO2003}$=0.357"	$t_{m\text{-}RFO2003}$=0.413"	No base line, 0.094% $Cr_{RFO2003}$
Preseparator Tank Drain to HD Tank	M45-22	6" Sch 80	$t_{nom\text{-}1984}$=0.432"	1.3	1.025	$t_{p\text{-}RFO2005}$=0.405"	$t_{m\text{-}RFO2005}$=0.365"	No base line & Cr measurements
FW Control Valve to S/G 1A	M84-11	14" Sch 100	$t_{nom\text{-}1970}$=0.938"	1.5	0.656	$t_{p\text{-}RFO2005}$=0.858"	$t_{m\text{-}RFO2005}$=0.821"	No base line & Cr measurements

NOTE: (1) Predicted Thickness = Nominal (or NDE) Thickness – [Current Wear Rate * Years of Service / Line Correction Factor]

2.1.9 Steam Generator Tube Inservice Inspection

2.1.9.1 Regulatory Evaluation

Steam generator (SG) tubes constitute a large part of the reactor coolant pressure boundary (RCPB). SG tube inservice inspection (ISI) provides a means for assessing the structural and leaktight integrity of the SG tubes through periodic inspection and testing of critical areas and features of the tubes. The Ginna Nuclear Power Plant, LLC (Ginna) staff's review in this area covered the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed EPU on plugging limits, potential degradation mechanisms (e.g., flow-induced vibration), plant-specific alternate repair criteria, and redefined inspection boundaries. The NRC's acceptance criteria for SG tube ISI are based on 10CFR50.55a requirements for periodic inspection and testing of the RCPB. Specific review criteria are contained in SRP Section 5.4.2.2 and other guidance provided in Matrix 1 of RS-001. Additional review guidance is contained in Ginna Technical Specification 5.5.8 for SG surveillance requirements, Regulatory Guide 1.121 for SG tube plugging limits, GL 95-03 and Bulletin 88-02 for degradation mechanisms, and NEI 97-06 for structural and leakage performance criteria, all of which form the basis for alternate repair criteria or redefined inspection boundaries.

Ginna Current Licensing Basis
ISI of SGs is discussed in UFSAR section 5.4.2.2. SG ISI is conducted in accordance with the Inservice Inspection (ISI) Program document. A program of periodic SG inspections, designed to meet the guidance of Regulatory Guide 1.83, is conducted to provide assurance of acceptable SG performance. The ISI program for the RCPB is discussed in Section 5.2.4. As part of the response to NRC Generic Letter 97-06, Ginna committed to develop a secondary side inspection program to ensure that degradation of SG internals does not adversely affect tube integrity (Reference 1).

In addition to the evaluations described in the Ginna UFSAR, the Steam Generator Integrity Program was evaluated for the Ginna Station License Renewal. The evaluations are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The Steam Generator Integrity Program is addressed in Section 3.1.2.3.5 of the License Renewal Safety Evaluation Report. As discussed in the plant license renewal application, Ginna has an aging management program for the steam generators that implements the criteria of Nuclear Energy Institute document NEI 97-06, *Steam Generator Program Guidelines*. These guidelines incorporate a balance of prevention, inspection, evaluation, repair, and leakage monitoring activities.

2.1.9.2 Technical Evaluation

SG process parameters will change as a result of the proposed EPU. Parameters that are expected to change include temperatures, steam pressure, steam and feedwater flows, void fraction distributions, circulation ratio.

The replacement SGs installed at Ginna in 1996 were designed with additional heat transfer capability to accommodate a future power uprate. The replacement SGs have a higher heat transfer capability, incorporating a larger surface area and adjusting for the reduced thermal conductivity of the Alloy 690 tubing compared to the Alloy 600 tubing in the original SGs. The replacement SGs incorporate many other design and material improvements, and the impacts of the EPU on the replacement steam generators have been assessed and found to be acceptable.

The process of SG tube ISI and integrity assessment will not change as a result of the EPU. The tube integrity assessment process begins with an assessment of potential degradation mechanisms and selection of applicable non-destructive examination techniques that will be used during the ISI to determine if any degradation exists. After performing the ISI, a condition monitoring assessment is performed to determine if there may have been structural or leakage integrity issues during the operating interval since the previous inspection. After employing conservative growth rates, an operational assessment is performed to ensure that structural and leakage integrity performance criteria will be met during the operating interval until the next inspection. Tubes that are not projected to meet the structural and/or leakage integrity criteria are then removed from service by plugging, or repaired using an approved method.

Although the process parameter changes due to the EPU may impact the initiation and growth rates of various degradation mechanisms, these changes are considered per the above assessments, and will be incorporated into the selection of the type of NDE program action.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

While the proposed EPU conditions will change process parameters, there are no changes being made to the SGs of a material or structural nature. The potential effects of process parameters are subject to an existing aging management review program, the SG Surveillance Program. No unevaluated material changes to the SGs are being made that would change the scope of the SGs with respect to license renewal. Therefore, no new aging effects requiring management are identified.

2.1.9.3 Conclusion

The Ginna Staff has evaluated the effects of the proposed EPU on SG tube integrity and concludes that the evaluation has adequately assessed the continued acceptability of the plants TSs under the proposed EPU conditions and has identified appropriate degradation management inspections to address the effects of in temperature, differential pressure, and flow rates on SG tube integrity. The Ginna staff further concludes that SG tube integrity will continue to be maintained and will continue to meet the performance criteria in NEI 97-06 and the requirements of 10CFR50.55a following implementation of the proposed EPU.

Section 2.1.9 References

1. Letter from R. Mecredy, RG&E to G.S. Vining, NRC, Subject: Response to NRC Generic Letter 97-06, "Degradation of Steam Generator Internals", dated March 30, 1998

2.1.10 Steam Generator Blowdown System

2.1.10.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of steam generator tubes. The steam generator blowdown system removes steam generator secondary-side impurities and thus, assists in maintaining acceptable secondary-side water chemistry in the steam generators. The design basis of the steam generator blowdown system includes consideration of expected and design flows for all modes of operation. The Ginna Nuclear Power Plant, LLC's (Ginna) review focused on the ability of the steam generator blowdown system to remove particulate and dissolved impurities from the steam generator secondary side during normal operation, including anticipated operational occurrences (main condenser in-leakage and primary-to-secondary leakage).

The NRC's acceptance criteria for the steam generator blowdown system are based on GDC-14, insofar as it requires that the reactor coolant pressure boundary be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture and of gross rupture.

Specific review criteria are contained in SRP section 10.4.8.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station steam generator blowdown design relative to conformance with:

- GDC 14 Reactor Coolant Pressure Boundary, is described in Ginna UFSAR section 3.1.2.2.5, which requires that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low

probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. This regulatory requirement is applicable to the steam generator blowdown system insofar as the system is directly connected to the secondary (steam) side of the steam generators.

The Steam Generator Blowdown System is discussed in IPSAR Section 4.12.3 (Topic III-4.C), and Section 4.13.2 (and 2.5.2 of the NUREG-0821 Supplement) (Topic III-5.A).

Ginna UFSAR sections that address the steam generator blowdown system include:

- Ginna UFSAR section 11.5, discharge effluent radiation monitoring, requires radioactivity monitoring of discharge flow paths prior to their final release point.

- Ginna UFSAR section 3.7.3.7, seismic piping upgrade, which upgraded certain Seismic Category I Piping systems to more current requirements and provided a seismic data base for use with modifications, the inservice inspection program, and NRC requests for information. The 2 inch and 3 inch diameter lines from the steam generators through the penetrations to the outside isolation valves were included in the upgrade program.

- Ginna UFSAR section 10.7.8, erosion / corrosion, which addressed this concern in single and two phase flow systems, consistent with the requirements of NUREG 1344 and the NUMARC erosion / corrosion report, dated June 11, 1987, to ensure that erosion / corrosion does not result in unacceptable degradation of the structural integrity of high energy carbon steel piping systems.

- Ginna UFSAR section 6.2.4, containment isolation features to isolate the steam generator blowdown lines penetrating containment to ensure that the total leakage of activity will be within design limits in the event of an accident.

Additional steam generator blowdown system details are provided in Ginna UFSAR Sections 3.6.1, Postulated Piping Failures In Fluid Systems Inside Containment, 3.6.2 Postulated Piping Failures in Fluid Systems Outside Containment, 10.7.7, Secondary Chemistry Control and 9.3.2, Sampling Systems.

In addition to the information described in the Ginna UFSAR, the Ginna Station's steam generator blowdown system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

The above SER, discusses the portion of steam generator blowdown system directly connected to the steam generators in section 2.3.4 Main and Auxiliary Steam. Plant programs used to manage the aging effects associated with the steam generators are discussed in section 3.4 of the SER.

2.1.10.2 Technical Evaluation

2.1.10.2.1 Introduction

The steam generator blowdown system is described in Ginna UFSAR section 10.7.5. The steam generator blowdown system design functions are:

- To blow down fluid at a continuous rate for chemistry control of each steam generator
- To blow down fluid at a surge rate, nominally at a frequency of once a week, and when needed to recover from abnormal chemistry
- To recover both the blowdown water and its heat capacity
- To provide for containment isolation of blowdown lines penetrating containment.

Continuous blowdown from the steam generators reduces accumulation of solids that result from the boiling process. The normal blowdown flow rate that has proven to be effective is approximately 100 gpm continuous flow for each steam generator. The blowdown system is designed to surge from the continuous flow rate to a periodic surge blowdown rate of 150 gpm for each steam generator for a period of three to five minutes, nominally at a frequency of once a week. The surge flow rates, duration, and frequency are a function of steam generator corrosion product buildup and plant operating condition.

The 100 gpm blowdown flow historically was sent partially to the circulating water canal for discharge and partially to the condenser to be cleaned up by the all-volatile treatment (AVT) system (condensate polishing demineralizer system) and reused. However, the operating protocol for the AVT system has changed and the condensate polishing demineralizer portion of the AVT system is not regularly used. It can be put in service to clean up and recycle blowdown when unusual accumulations of impurities occur.

2.1.10.2.2 Description of Analyses and Evaluations

The steam generator blowdown system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations were conservatively performed for an analyzed NSSS thermal power of 1781 MWt. The existing design parameters of the systems / components listed below were compared with the EPU conditions.

- Normal and surge blowdown flow rates

- Operating and design pressures and temperatures

- Monitoring of liquid effluents (steam generator blowdown) prior to release, is addressed in LR section 2.10.1, Occupational and Public Radiation Doses.

- Fluid velocities and the potential for increased erosion / corrosion. The erosion / corrosion monitoring program is evaluated in LR section 2.1.8, Flow Accelerated Corrosion.

- Safety related valve closure and testing requirements (containment isolation) are addressed in LR section 2.2.4, Safety Related Valves and Pumps.

- The review of piping / component supports is described in LR section 2.2.2.2, Balance Of Plant Piping and Supports (All Non-Class 1).

- Protection against dynamic effects, including missiles, pipe whip and discharging fluids is addressed in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and LR section 2.5.1.3, Pipe Failures

- Environmental qualification of the containment isolation valves is addressed in LR section 2.3.1, Environmental Qualification of Electrical Equipment.

2.1.10.2.3 Results

The increased steam and feedwater flow rates at EPU conditions do not significantly affect the concentration of impurities throughout the turbine cycle nor increase the effect of the impurities on the steam generators. Blowdown rate subsequent to implementing EPU will be within the historical 100 gpm/steam generator rate. Therefore, no changes to the design steam generator blowdown flow rates or operating modes are needed as a result of the EPU.

Historically, Ginna operated with a normal blowdown flow of approximately 100 gpm per steam generator. Presently, Ginna is operating with a steam generator blowdown flow of only approximately 40-80 gpm per steam generator. This is because the condensate polishing demineralizer portion of the AVT system is not being used. In this operating mode, steam generator blowdown is entirely discharged to the circulating water system rather than being partially "recycled" by returning some blowdown to the main condenser for treatment, as is required when running AVT. Thus the concentration factor is lower and less blowdown is needed to maintain chemistry. Makeup water is added to the turbine cycle at the same rate as blowdown is discharged.

For the EPU, the steam generator blowdown flow will be maintained between 40 - 100 gpm since the steam generator impurities are not expected to be significantly higher at EPU conditions.

Since the flow velocity and temperature in the steam generator blowdown piping at EPU are unchanged from the original design parameters, the potential for erosion / corrosion will not increase due to the EPU. This is further discussed in LR section 2.1.8.2.2, Flow-Accelerated Corrosion. The blowdown system is now, and will continue to be monitored by the Erosion / Corrosion Program after EPU.

At EPU conditions, the operating temperatures and pressures in the steam generators, steam generator blowdown tank and interconnecting piping and valves increase slightly due to the higher Tave and steam generator operating pressure. However, the existing design pressure and temperature of the steam generators, 1085 psig / 557^{0}F, remain bounding for EPU conditions since these values are based on the no-load operating condition which does not change at EPU. Therefore, the design conditions for the steam generator blowdown piping and components connected to the steam generators also remain bounded for EPU conditions.

The steam generator blowdown lines penetrating containment are provided with air-operated isolation valves which are designed to close for containment isolation post-accident. The maximum blowdown flow rates and pressures experienced by these valves at EPU do not exceed the existing valve design capabilities and; therefore, these valves continue to meet their containment isolation design function.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The steam generator blowdown operating flow rates and process conditions are within the original design parameters of the system. There are no system / component modifications necessary. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. There are no new or previously unevaluated materials in the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.1.10.3 Conclusions

The effects of the proposed EPU on the steam generator blowdown system have been evaluated for changes in system flow and impurity levels with the conclusion that they continue to meet the Ginna Station current licensing basis requirements with respect to GDC-14 following implementation of the proposed EPU. The ability of the steam generator blowdown system to remove particulate and dissolved impurities from the

steam generator secondary side during normal operation, including anticipated operational occurrences, will not be affected. No modifications to the steam generator blowdown system are required for EPU. Therefore, the proposed EPU is acceptable with respect to the steam generator blowdown system.

2.1.11 Chemical and Volume Control System

2.1.11.1 Regulatory Evaluation

The chemical and volume control system (CVCS) and boron recovery system (BRS) provide means for:

- Maintaining water inventory and quality in the reactor coolant system (RCS)
- Supplying seal-water flow to the reactor coolant pumps (RCPs) and pressurizer auxiliary spray
- Controlling the boron neutron absorber concentration in the reactor coolant
- Controlling the primary water chemistry, reducing coolant radioactivity level, and recycling coolant for demineralized water makeup for normal operation

Note: Ginna Station does not use CVCS to provide high pressure injection flow for emergency core cooling in the event of postulated accidents.

Ginna Nuclear Power Plant, LLC (Ginna) reviewed the safety-related functional performance characteristics of CVCS components. The NRC's acceptance criteria are based on GDC-14 insofar as it requires that the reactor coolant pressure boundary be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture, and on GDC-29, insofar as it requires that the reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in anticipation of operational occurrences.

Specific review criteria are contained in the SRP, Section 9.3.4.

Ginna Current Licensing Basis

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), section 3.1, the GDC used during the licensing of the Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in UFSAR, sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station CVCS relative to conformance to:

- GDC-14 is described in UFSAR, section 3.1.2.2.5, "GDC-14 – Reactor Coolant Pressure Boundary." As described in this UFSAR section, all piping components and supporting structures of the RCS were designed as Class 1 and later re-evaluated as Seismic Category 1 equipment as defined in UFSAR, section 3.7.

- GDC-29 is described in UFSAR, section 3.1.2.3.10, "GDC-29 – Protection Against Anticipated Operational Occurrences." As described in this UFSAR section, the Ginna protection and reactivity control systems are designed, constructed, operated, and maintained to ensure high reliability in regard to their required safety functions in any anticipated operational occurrences.

As described in UFSAR, section 9.3.4.2, the CVCS supports reactor coolant pressure boundary material integrity by maintaining the RCS water chemistry necessary to meet pressurized water reactor (PWR) RCS water chemistry technical specifications.

As described in UFSAR section 9.3.4.1.1, the CVCS supports reactivity control in addition to the reactivity control achieved by the control rods. This is done by regulating the concentration of boric acid solution neutron absorber in the RCS.

As described in UFSAR section 6.2.4.4 the CVCS supports containment isolation system functions of limiting the release of potentially radioactive materials to the environment through CVCS piping segments which penetrate the containment.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's CVCS was evaluated for plant license renewal. System and system component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

With respect to the above SER, the CVCS is described in LR section 2.3.3.1. The programs used to manage the aging effects associated with auxiliary systems are discussed in section 3.3 of the Ginna Licensing Renewal SER.

2.1.11.2 Technical Evaluation

2.1.11.2.1 Introduction

The CVCS is described in the Ginna UFSAR, section 9.3.4. The system is designed to perform the following functions. Note: the CVCS functions related to RCS boron control and containment isolation are safety-related.

- To control the reactor coolant inventory, chemistry conditions, activity level, and boron concentration
- To provide seal-water injection flow to the reactor coolant pumps
- To process reactor coolant effluent for reuse of boric acid and makeup water
- To provide pressurizer auxiliary spray
- To support containment isolation

To perform these functions, continuous feed and bleed is maintained between the RCS and the CVCS. Water is let down from the RCS, through a regenerative heat exchanger (HX), to minimize thermal loss from the RCS. The pressure is reduced through orifices and further cooling occurs in the non-regenerative HX followed by a second pressure reduction. Water is returned to the RCS by the charging system, which also provides seal injection flow to the reactor coolant pumps.

The chemistry of the letdown flow may be altered by passing the flow through demineralizers that remove ionic impurities. A filter removes solids, and the gases dissolved in the coolant are removed in the volume control tank. The boric acid concentration in the coolant is changed by the reactor makeup portion of the CVCS as required for reactivity control. Excess coolant may be diverted into the boron recycle portion of the CVCS for reprocessing into pure water and concentrated boric acid.

2.1.11.2.2 Description of Analysis and Evaluations

The CVCS was evaluated to ensure the system is capable of performing its intended functions for the range of Nuclear Steam Supply System (NSSS) design parameters approved for EPU (LR, section 1.1, Nuclear Steam Supply System Parameters, Table 1-1). The evaluation was conservatively performed for an analyzed NSSS thermal power of 1817 MWt.

The changes in NSSS design parameters that could potentially affect the CVCS design bases functions include the increase in core power and the allowable range of RCS full-load design temperatures. The increase in core power and the allowable range of RCS full-load design

temperatures may also affect the CVCS design bases requirements related to the core re-load boron requirements. Additionally the allowable range of RCS full load design temperatures may affect the heat loads that the CVCS heat exchangers must transfer to the component cooling water system and in the case of the regenerative heat exchanger to the charging flow.

Regenerative Heat Exchanger

The regenerative HX cools the normal letdown flow from the RCS, which is at RCS T_{cold} temperature. The design inlet (RCS T_{cold}) temperature of the regenerative HX is 552.5°F, which bounds the highest RCS T_{cold} temperature associated with the RCS no-load temperature of 547°F (LR, section 1.0, Table 1-1). The no-load RCS temperature has not changed. Although the full-load EPU T_{cold} temperature will increase above the current values, they will remain less than the design basis T_{cold} and no load T_{cold} values. And therefore the performance of the regenerative HX remains essentially unchanged due to EPU and is acceptable at EPU conditions with no plant changes required.

Non-Regenerative Heat Exchanger

The non-regenerative HX cools the letdown flow from the regenerative HX. Since the change in performance of the regenerative HX is essentially unchanged at EPU conditions, as discussed in the previous section, there is essentially no effect on the performance of the non-regenerative HX. Minor differences in letdown temperature can easily be accommodated within the capability of the non-regenerative HX cooling water temperature control valve, TCV-130. Therefore, it is concluded that acceptable non-regenerative HX performance is provided at the EPU conditions, with no plant changes required.

Excess Letdown Heat Exchanger

The excess letdown HX cools the excess letdown flow from the RCS, which is at RCS T_{cold} temperature. The design inlet (RCS T_{cold}) temperature of the excess letdown HX is 552.5°F, which bounds the highest RCS T_{cold} temperature associated with the RCS no-load temperature of 547°F. Since the no-load RCS temperature has not changed, and the full-load EPU T_{cold} temperature remains below the no load temperature, the performance of the excess letdown HX is acceptable at EPU conditions with no plant changes required.

Seal Water Heat Exchanger

The seal water HX cools the seal return flow from the two RCP No. 1 seals and the excess letdown flow (from the excess letdown HX) if it is in service. The RCP heat load (including the thermal barrier HX) is a function of RCS T_{cold} temperature, while the excess letdown heat load is a function of excess letdown HX performance. Since the no-load RCS temperature has not

changed and the full-load EPU T_{cold} temperature remains below the no load temperature, the performance of the seal water HX is acceptable at EPU with no plant changes required.

Charging, Letdown, and RCS Makeup (Boration, Dilution, purification, and N-16 Delay Time)

As discussed in the previous sections for the various CVCS HXs, there are essentially no effects on their performance at the EPU conditions. Therefore, the charging and letdown flows at EPU conditions are essentially unchanged.

The flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the EPU conditions. However, the makeup system also relies on storage capacity of various sources of water including primary makeup water and boric aid solutions from both the boric acid storage tanks and the refueling water storage tank (RWST).

Primary makeup water is used to dilute RCS boron, to provide positive reactivity control, or to blend concentrated boric acid to match the prevailing RCS boron concentration during RCS inventory makeup operations. Since the flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the EPU conditions as discussed above, the EPU does not affect the capability of the makeup system to perform these system functions.

The boric acid storage tanks (BAST) and RWST provide the sources of boric acid for providing negative reactivity control to supplement the reactor control rods. The EPU is expected to have a small effect on the boration requirements that must be provided by the CVCS boration capabilities. The maximum expected RCS boron concentrations are within the capability of the CVCS. The Westinghouse reload safety evaluation (RSE) process (currently incorporated into Ginna Technical Specifications) is designed to address boration capability for routine plant changes, such as core reloads, and infrequent plant changes such as a plant uprating that result in a change to core operating conditions and initial core reactivity. Therefore, boration capability will be addressed during the RSE process for each reload cycle.

The CVCS letdown flow is fixed and charging flow is varied to control pressurizer water level and RCS inventory. The pressurizer water level is programmed as a function of power level to accommodate RCS coolant expansion. Accordingly this programmed level is being changed based on the EPU NSSS design parameters. However, this change has no impact on the ability of the CVCS to maintain RCS inventory which is accomplished via letdown, charging and makeup.

Analyses have indicated the potential for an increase in crud buildup at EPU. The expected increase in the required charging and letdown flow to provide the additional RCS purification/cleanup is within the current charging and letdown flow capabilities.

The letdown flow path is routed inside containment such that there is adequate decay of N-16 before the letdown fluid leaves the containment building. Since letdown flow is essentially unchanged, as discussed in the previous paragraphs, this radiation protection feature of the CVCS is not affected by the EPU. However it is noted that the letdown line and excess letdown line radiation dose rates from N-16 (for example, amount of N-16) will increase proportional to the increase in reactor power level.

Refer to LR section 2.2.2, Pressure-Retaining components and Component Supports for an evaluation of the CVCS class 1 piping including RCS nozzles and thermal sleeves.

Evaluation of Impact on Renewal Plant Operating License, Evaluations and License Renewal

EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the CVCS at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

2.1.11.3 CVCS Results

The evaluations of the CVCS charging, letdown, and RCS makeup performance show the CVCS is acceptable at the EPU conditions, with no plant changes. Accordingly, the performance of the following CVCS functions (which are accomplished via charging, letdown, and makeup) are acceptable at EPU conditions with no plant changes.

- Control of RCS inventory and activity levels.
- Control of water chemistry required to ensure reactor coolant pressure boundry material integrity in accordance with Ginna Station current licensing basis requirements with respect to GDC 14.
- Control of RCS soluable boron required to control reactivity in accordance with Ginna Station current licensing basis requirements with respect to GDC 29.
- Provision of seal water injection for the reactor coolant pumps (RCPs).
- Providing Pressurizer auxiliary spray

- Deliverance of water to the CVCS boron-recycle subsystem for reprocessing and recovery of boric acid and makeup water.

The CVCS boration capability is addressed during the Reload Safety Evaluation (RSE) process (currently incorporated into Ginna Technical Specifications) for each core re-load cycle (reference WCAP-9272-P-A, *Westinghouse Reload Safety Evaluation Methodology*, F. M. Bordelon et al., July 1985).

The performance of the CVCS components including valves and piping that support containment isolation are not affected by change in RCS design parameters resulting from EPU.

There is a small increase in letdown line dose rates from N-16, proportional to the increase in reactor power level. This small increase has been evaluated in LR section 2.10.1, Occupational and Public Radiation Doses, as being acceptable.

Refer to LR section 2.2.2.1, NSSS Piping, Components and Supports, for results of the evaluation of the CVCS class 1 piping including RCS nozzles and thermal sleeves.

The CVCS support functions provided by the sampling system and waste disposal system are not affected by the change in RCS conditions resulting from the EPU.

2.1.11.4 Conclusion

Ginna has reviewed the evaluation of the effects of the EPU on the CVCS and boron recovery system and concludes that the evaluation adequately addressed changes in the temperature of the reactor coolant and their effects on the CVCS and boron recovery system. Ginna further concludes that the CVCS and boron recovery system continue to be acceptable and continue to meet the requirements of GDC-14 and GDC-29 following implementation of the EPU. Therefore, Ginna finds the EPU acceptable with respect to the CVCS.

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

2.2.1.1 Regulatory Evaluation

Safety-related structures, systems, and components could be impacted by the pipe-whip dynamic effects of a pipe rupture. Ginna Nuclear Power Plant, LLC (Ginna) conducted a review of pipe rupture analyses to ensure that those safety-related structures, systems, and components are adequately protected from the effects of pipe ruptures. Ginna's review covered:

- The implementation of criteria for defining pipe break and crack locations and configurations
- The implementation of criteria dealing with special features, such as augmented In-service inspection programs or the use of special protective devices such as pipe-whip restraints
- Pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects
- The design adequacy of supports for structures, systems, and components provided to ensure that the intended design functions of the structures, systems, and components will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings

Ginna's review focused on the effects that the proposed extended power uprate (EPU) may have on the above items. The NRC's acceptance criteria are based on General Design Criterion (GDC-4), which requires that the structures, systems, and components important to safety be designed to accommodate the dynamic effects of a postulated pipe rupture. Specific review criteria are contained in the Standard Review Plan (SRP), Section 3.6.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR Section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to,

conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station safety related structures, systems and components with respect to potential pipe ruptures and their associated dynamic effects relative to conformance to:

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 – Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant.

Conformance to the requirements of GDC-4, ensuring that safety related structures, systems and components are adequately protected with respect to pipe ruptures and their associated dynamic effects, is addressed in Ginna UFSAR Section 3.6.

UFSAR Section 3.6 describes the design features of Ginna Station that protect essential equipment from the consequences of postulated piping failures both inside and outside containment. Initial analyses were conducted in accordance with guidance and criteria set forth in the AEC Letter from A. Giambusso to E. J. Nelson (RG&E), dated December 18, 1972, concerning high energy pipe breaks outside containment, and the Systematic Evaluation Program Review for Topics III-5.A and III-5.B related to pipe breaks inside and outside containment, respectively.

The analyses showed that, with certain modifications proposed by RG&E, the requirements of GDC-4 were satisfied ensuring that safety related structures, systems and components are adequately protected with respect to pipe ruptures and their associated dynamic effects.

Pipe ruptures were postulated at arbitrary intermediate locations in addition to terminal ends and high stress locations as required at the time by Branch Technical Position (BTP) MEB 3-1 of Standard Review Plan Section 3.6.2 in NUREG 0800. Pipe whip restraints and jet impingement shields were installed as necessary to mitigate the effects of these arbitrary intermediate locations. Generic Letter 87-11, dated June 19, 1987, revised BTP MEB 3-1 to Revision 2 to eliminate the requirement to postulate arbitrary intermediate pipe ruptures and permitted the elimination of pipe whip restraints and jet impingement shields installed to mitigate the effects of arbitrary intermediate pipe ruptures.

Postulated Piping Failures in Fluid Systems Inside Containment

High energy piping lines inside containment were evaluated for the effects of potential pipe breaks.

An effects oriented approach was initially utilized for evaluating the consequences of most potential high energy line breaks. This approach postulates a high energy pipe break inside containment anywhere along the line and analyzes the capability of the remaining systems to safely shut down the reactor.

Other evaluation techniques were considered to evaluate breaks that could not be shown to have acceptable consequences using the effects oriented approach alone. For example, a mechanistic approach based upon breaks at locations of highest stress in the piping segment, in accordance with MEB 3-1, may result in acceptable consequences because these breaks are remote from required equipment or because the postulated pipe breaks are contained. This approach also analyzed failure mechanisms to demonstrate that the consequences were acceptable. For example, a safety-related pipe assumed to be impacted in an effects-oriented analysis may be shown to have sufficient strength to resist whipping using mechanistic methods.

In accordance with 10CFR50, Appendix A, GDC-4, leak-before-break was used to demonstrate that certain high energy lines were designed, constructed, and analyzed so as to have a negligible probability of failure as part of their design basis. The use of leak-before-break is described in section 2.1.6 of this report.

Postulated Piping Failures in Fluid Systems Outside Containment

In December 1972, the NRC staff sent letters (AEC Letter from A. Giambusso to E. J. Nelson (RG&E), dated December 18, 1972) to all power reactor licensees requesting an analysis of the effects of postulated failures of high energy lines outside of containment.

In response to this letter, RG&E submitted an evaluation of the effects of postulated high energy line breaks outside of containment. As a result of that evaluation and subsequent follow-up evaluations, RG&E committed to perform certain station modifications and to implement an augmented in-service program to mitigate the effects of postulated pipe breaks. The augmented in-service program was approved by the NRC in Amendment 7 to the Ginna operating license.

The initial evaluations in response to AEC Letter from A. Giambusso to E. J. Nelson (RG&E), dated December 18, 1972, as well as the evaluations performed in response to SEP Topics III-5.A and III-5.B, are described below:

Pipe break criteria define a high energy fluid system as one where the maximum operating temperature is greater than or equal to 200°F or the maximum operating pressure is greater than or equal to 275 psig. All other piping is considered moderate energy piping. An effects oriented approach to determine the acceptability of plant response to pipe breaks was performed (i.e., each structure, system and component which must function to mitigate the effects of the pipe break and to safely shut down the plant was examined to determine its susceptibility to the effects of the postulated break. Break effects considered were compartment pressurization, pipe whip, jet impingement, spray, flooding, and environmental conditions of temperature, pressure and humidity.

The SEP reevaluation of pipe breaks outside containment considered the zones within the plant, which contain systems required for safe shutdown and/or systems required to mitigate the effects of the postulated breaks. A detailed discussion of postulated pipe break locations and Ginna mitigating strategies is provided in section 2.5.1.3 of this report.

Other Ginna UFSAR sections discussing the design of BOP and Non-class 1 piping and supports that are potentially impacted by pipe rupture and their associated dynamic effects include:

- Ginna UFSAR section 3.2, Classification of Structure, Components, and Systems, provides details with respect to the seismic classification of piping and piping components.

- Ginna UFSAR section 3.7, Seismic Design, and specifically section 3.7.3.1, Seismic Analysis Methods, provides details with respect to the seismic qualification of piping and piping components.

- Ginna UFSAR section 3.9, Mechanical System and Components, and specifically sections 3.9.2.1.8, Seismic Piping Upgrade Program and 3.9.2.1, Dynamic Testing and Analyses - Piping Systems, provide details with respect to the seismic qualification of piping and piping components.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's pipe rupture components were evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the equipment and components credited with mitigating the effects of pipe ruptures and associated dynamic effects are described in section 2.4 and the programs credited with managing that equipment aging are described in section 3.5.2.

2.2.1.2 Technical Evaluation

2.2.1.2.1 Introduction

GDC-4 requires that the safety related structures, systems, and components be designed to accommodate the dynamic effects of a postulated pipe rupture, including pipe whip dynamic effects and jet thrust and impingement effects. The licensing basis for Pipe Ruptures Locations and Associated Dynamic Effects for the Ginna Station is identified in the Ginna UFSAR section 3.6.

Refer to LR section 2.5.1.3, Pipe Failures, for discussion of plant design for protection from piping failures outside containment.

The following discussion relates to postulated pipe breaks inside containment not credited with leak-before-break, which is discussed in LR section 2.1.6. Following the application of LBB, the remaining pipe breaks in the mechanical design basis of the RCS are all less than 10 inch primary, and secondary side branch line breaks interfacing with the RCS. The applicable break locations for the main coolant piping, the pressurizer, the surge line, the reactor vessel (RV), the

steam generators (SGs), and the reactor coolant pumps (RCPs) are discussed in Section 2.2.2 Pressure-Retaining Components and Component Supports. Section 2.2.3 Reactor Pressure Vessel Internals and Core Supports discusses reactor vessel internals (RVI) and fuel design qualifications.

2.2.1.2.2 Description of Analyses and Evaluations

Affected Balance of Plant piping (Non Class1) systems were evaluated to address revised EPU operating conditions. Applicable pipe rupture postulation criteria were reviewed and changes to piping system stress levels resulting from EPU were reconciled against these documents. The evaluations performed for these piping systems did not result in any new or revised break locations, and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for EPU.

Based on the evaluations performed for EPU noted above, the following were demonstrated:

- Existing criterion for defining pipe break and crack locations and configurations is unaffected by EPU.
- Criterion dealing with special features, such as augmented in service inspection programs or the use of special protective devices such as pipe whip restraints is unaffected by EPU.
- Existing pipe whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe whip dynamic effects remains valid for EPU.
- Existing design of structures, systems, and components (SSC's) remains acceptable to protect safety related SSC's from the effects of pipe whip and jet impingement loading for EPU.

Hence, the design features for the Ginna Station that protect safety related structures, systems and components from the consequences of postulated piping failures both inside and outside containment as described in UFSAR sections 3.6.1 (for inside containment) and 3.6.2 (for outside containment) remain valid for EPU.

Pending NRC approval (see LR section 2.1.6), LBB is applicable for the RCS main loop piping and the pressurizer surge line, and also for the accumulator and residual heat removal (RHR) lines, exempting these large diameter breaks from consideration for dynamic effects analysis. For the EPU program, the LOCA hydraulic forcing function forces and associated loop LOCA RV motions from the smaller branch line breaks were used, namely the 3-inch pressurizer spray line on the cold leg, the 2-inch safety injection line on the hot leg, and the 4-inch upper plenum injection line connection to the vessel. Evaluations for the piping locations, in consideration of the EPU conditions are provided in Section 2.2.2 Pressure-Retaining Components and Component Supports.

The applicable piping loads resulting from EPU conditions, as defined by the Table 1-1 parameters, were evaluated and confirmed for continued applicability of LBB (Section 2.1.6,

Leak-Before-Break). The next largest branch lines were the 3-inch pressurizer spray line connected to the cold leg and the 4-inch upper plenum injection (UPI) line. Of these two breaks, the 3-inch pressurizer spray line break was found to be more limiting due to imparted forces and stresses.

The impact on the Loop LOCA hydraulic forcing functions due to the EPU Program is addressed in Section 2.8.5.6.3.3, LOCA Hydraulic Forcing Functions, and the associated loop LOCA RV motions are addressed in Section 2.2.3, Reactor Pressure Vessel Internals and Core Supports.

The results of LOCA reactor vessel displacements and the impact forces calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals (Section 2.2.3, Reactor Pressure Vessel Internals and Core Supports). The core plate motions for both breaks were used in fuel grid crush analysis and to confirm the structural integrity of the fuel (Section 2.8.1, Fuel System Design). The structural integrity evaluation explicitly considers the transition to the Westinghouse 14x14 422VANTAGE+ nine-grid fuel design (see Section 2.8.1, Fuel System Design) to be implemented coincident with the first operating cycle at EPU conditions.

In 1998 the Nuclear Regulatory Commission issued Information Notice (IN) 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants". Ginna participated with the Westinghouse Owners Group to respond to these concerns and evaluate the structural integrity of Ginna's baffle-former bolts. The effect of the EPU has also been evaluated in Section 2.2.3, Reactor Pressure Vessel Internals and Core Supports, to address any potential effects on the conclusions and requirements arising from this Information Notice.

2.2.1.2.3 Results

For Balance of Plant piping (Non Class1) systems, the evaluations for EPU conditions did not result in any new or revised break locations, and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for EPU. Hence, for rupture postulation issues, the Balance of Plant piping (Non Class 1) and support systems continue to meet their licensing basis and satisfy the requirements of GDC-4.

Reanalysis for the LOCA Hydraulic Forcing Functions (Section 2.8.5.6.3.3 LOCA Hydraulic Forcing Functions) under the EPU conditions in Table 1-1, with modeling of the transition to the Westinghouse 14x14 422VANTAGE+ nine-grid fuel design confirm the structural integrity of the reactor vessel internals (Section 2.2.3, Reactor Pressure Vessel Internals and Core Supports) and the fuel (Section 2.8.1, Fuel System Design). No new requirements with regard to reactor vessel internal baffle former bolts have been identified.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The evaluations for EPU conditions did not result in any new or revised break locations, and the existing design basis for pipe break, jet impingement, and pipe whip considerations remains valid for EPU. As a result, there were no modifications to existing plant pipe rupture related support components. Hence, no new aging effects requiring management have been identified as a result of EPU.

2.2.1.3 Conclusion

Ginna has reviewed the evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the evaluations have adequately addressed the effects of the proposed EPU on them. Ginna further concludes that the evaluations have demonstrated that those safety-related structures, systems, and components will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-4 following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU at the Ginna Station acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Introduction

In keeping with the format of RS-001, this LR section is arranged differently than the others. In this section, there is a "Regulatory Evaluation" subsection that generally applies to all of the specific components that are addressed individually in each of the later "Technical Evaluation" subsections. In addition to the generic Regulatory Evaluation, any amplifications or qualifications that are necessary for individual types of components are provided in the Introduction for each component.

There is also a generic "Current Licensing Basis (CLB)" subsection that addresses compliance with the generic Regulatory Evaluation criteria. In addition to the generic CLB subsection, when necessary, a component-specific CLB provides further details pertinent to the component, and explains any exception to the generic CLB.

Regulatory Evaluation

Ginna Nuclear Power Plant, LLC (Ginna) has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code)*, Section III, Division 1, and GDC-1, -2, -4, -14, and -15. Ginna's review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. Although analysis of flow induced vibration effects is not included in the plant's current licensing basis, Ginna's review covered the impact of higher EPU flow rates on flow-induced vibration in certain more susceptible components. Ginna's review also included a comparison of the resulting stresses and cumulative fatigue usage factors against the code-allowable limits. The NRC's acceptance criteria are based on:

- 10CFR50.55a and GDC-1, insofar as they require that structures, systems, and components important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed

- GDC-2, insofar as it requires that those structures, systems, and components important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC-4, insofar as it requires that structures, systems, and components important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC-14, insofar as it requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture

- GDC-15, insofar as it requires that the reactor coolant system be designed with margin sufficient to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation

Specific review criteria are contained in SRP, Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1 and other guidance provided in Matrix 2 of RS-001.

Note: The NRC's regulatory evaluation in RS-001 for Pressure Retaining Components and Component Supports focuses on the ASME Boiler and Pressure Vessel Code. Design of Ginna Station's systems, structures and components generally preceded publication of the ASME codes. Specific design codes utilized as design criteria are detailed in the LR section detailing the specific SSC under review.

Ginna Current Licensing Basis

As noted in *Ginna Updated Final Safety Analysis Report* (UFSAR) Section 3.1, the general design criteria used during the licensing of the Ginna Station predate those provided today in *10CFR50, Appendix A, General Design Criteria for Nuclear Power Plants*. The adequacy of the Ginna design relative to the general design criteria is discussed in *Ginna Updated Final Safety Analysis Report* (UFSAR) Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, *The Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983*. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Ginna UFSAR Sections 3.1, 3.2, 3.5, 3.6, 3.7, 3.8, 3.9, and 5 address the quality assurance, classification and design of safety-related pressure retaining components and component supports.

Specifically, the adequacy of Ginna Station pressure-retaining components and component supports' design relative to conformance to:

- 10CFR50.55a(a)(1) is described in UFSAR section 3.2.1, "Classification of Structures, Components, and Systems – Introduction." As part of SEP Topic III-1, the original codes and standards used in the design of the Ginna Station were

compared with later licensing criteria, including 10CFR50.55a. The objective was to assess the capability of Ginna Station SSCs to perform their safety functions as judged by the later standards. Although several areas were identified where requirements had changed, all areas were satisfactorily resolved, and SEP Topic III-1 was closed.

- Mechanical properties requirements and stress and fatigue evaluation requirements, as applicable, of the ANSI B31.1 Power Piping Code (Reference 1) and the ASME Code, Section III (References 2 and 3) is addressed in UFSAR Section 3.9.

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1. General Design Criterion 1 - Quality Standards and Records, wherein it is noted that all structures, systems, and components (SSCs) of the facility were classified according to their importance. The classification of structures and equipment is discussed in Section 3.2. SSCs were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. The quality control and quality assurance program for Ginna Station construction, and the current quality assurance program, under which systems were installed and are maintained, are described in UFSAR Section 17.1, and 17.2, respectively.

- GDC-2 is described in Ginna UFSAR section 3.1.2.1.2, General Design Criterion 2 - Design Bases for Protection Against Natural Phenomena. As described therein, Ginna Station received NRC acceptance of its response to Generic Letter 87-02, including approval of the methodology for verification of equipment seismic adequacy, including equipment involved in future modifications and replacement equipment.

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review of this topic as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, *The Integrated Plant Safety Assessment Report, completed in August 1983*. Conformance to the requirements of GDC-4 is also described in the following:

  - Ginna UFSAR, Section 3.11
    - Environmental Design of Mechanical and Electric Equipment

- Ginna UFSAR, Section 3.6
  - Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping
    - Pipe Breaks Inside Containment (SEP, Topic III-5.A)
    - Pipe Breaks Outside Containment (SEP, Topic III-5.B)
- Ginna UFSAR, Section 3.5
  - Missile Protection

- GDC-14 is described in Ginna UFSAR section 3.1.2.2.5, General Design Criterion 14 - Reactor Coolant Pressure Boundary. This UFSAR section states that all piping components and supporting structures of the reactor coolant system were designed as Class I (see UFSAR Sections 3.1.1.1.1 and 3.7.1.1.1 for description of the original Class designations used during plant design) and later reevaluated as Seismic Category I equipment as defined in Section 3.7. All pressure containing components of the reactor coolant system were designed, fabricated, inspected, and tested in conformance with the code requirements listed in UFSAR Table 5.2-1.

- GDC-15 is described in Ginna UFSAR section 3.1.2.2.6, General Design Criterion 15 - Reactor Coolant System Design. This UFSAR section states that the reactor coolant system and associated auxiliary, control, and protection systems were designed with sufficient margins so that design conditions are not exceeded during Modes 1 and 2, and it also addresses overpressure protection.

In addition to their GDC compliance as described above, the Ginna Station pressure-retaining components and supports were evaluated for plant license renewal, which is documented in the License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant (NUREG-1786). Systems and system components, including materials of construction, operating history, programs used to manage aging effects, and time limited aging analyses are documented in SER sections 2.3, 3.0, 3.1, 3.2, 3.3, 3.4, 3.5, and 4.3. The fatigue monitoring program manages the time-limited-aging-analysis (TLAA) for metal fatigue and is discussed in section 4.3 of the SER. TLAA was incorporated into the Ginna Current Licensing Basis as part of license renewal.

Under the applicable design codes, the effect of flow induced vibration was not required to be investigated when Ginna was designed and first licensed. However, certain systems and/or components which are more susceptible to flow induced vibration, and which are evaluated to experience higher post-EPU flows of potential significance, have been analyzed. The following subsections within section 2.2.2 address flow induced vibration for their specific components.

2.2.2.1 NSSS Piping, Components, and Supports

2.2.2.1.1 Introduction

The nuclear steam supply system (NSSS) piping, which is the reactor coolant system (RCS) piping, consists of two heat transfer piping loops (loops A and B) connected in parallel to the reactor pressure vessel (RPV). Each loop contains a circulating pump called the reactor coolant pump (RCP) and a steam generator. Each RCS loop consists of three legs: the hot leg from the RPV to the steam generator, the cross-over leg from the steam generator to the RCP, and the cold leg from the RCP to the RPV. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and the instrumentation for operational control. The pressurizer is connected to loop B. Auxiliary system piping connections into the RCS piping are provided as necessary. The RCS piping system is supported by the primary equipment supports of the RCS, namely the RPV supports, the steam generator supports, the RCP supports, and the pressurizer supports.

The NSSS piping, components, and supports, as contained in the *Ginna Updated Final Safety Analysis Report* (UFSAR) sections 3.2, 3.5, 3.6, 3.7, 3.8, 3.9, 5.1, 5.2, and inclusive of the piping discussed in section 5.4; were evaluated for the EPU program. The existing design-basis analyses for reactor coolant loop (RCL) piping (Reference 4), RCL primary equipment supports (Reference 4), and pressurizer surge line including thermal stratification (Reference 5) were reviewed for the effects of input parameters that would change with the implementation of EPU.

Specifically, the following analyses were evaluated and, where necessary, reanalyzed with EPU parameters:

- RCL loss-of-coolant accident (LOCA) analysis using Loop LOCA hydraulic forces for the EPU program and the associated Loop LOCA reactor pressure vessel (RPV) motions for the EPU program
- RCL piping stresses
- RCL displacements at auxiliary piping line connections to the centerline of the RCL at branch nozzle connections and impact on the auxiliary piping systems
- Primary equipment nozzle loads
- RCL piping system leak-before-break (LBB) loads for LBB evaluation
- Pressurizer surge line piping analysis including the effects of thermal stratification

- RCL primary equipment support loads (Reactor Vessel, Steam generator, and Reactor Coolant Pump)

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to NSSS piping, components, and supports, with the following amplification. As discussed in the License Renewal SER section 3.1.2.3.8, a commitment was made to implement a fatigue monitoring program. The scope of the fatigue monitoring program includes the piping discussed in this section. Also, from the License Renewal SER, the 40 year design transient set remains valid (bounding) for 60 years of operation at the pre-EPU power level.

Analysis of flow induced vibration is not included in the NSSS licensing basis for Ginna. However, it was considered for more susceptible components that would experience a significant flow increase under EPU conditions. NSSS piping and components were evaluated and deemed unaffected by EPU conditions due to their heavy construction and small increase in flow, if any.

Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The following four basic sets of input parameters were used in the evaluation for the EPU:

- Design parameters for 1817 MWt Power as shown in Table 1-1 of L R section 1.1, Nuclear Steam Supply System Parameters,
- NSSS design transients in LR section 2.2.6, NSSS Design Transients,
- Loop LOCA hydraulic forcing functions forces in L R section 2.8.5.6.3.5, Technical Evaluation – LOCA Forces, and the associated
- Loop LOCA RPV motions in L R section 2.2.3, Reactor Pressure Vessel Internals and Core Supports.

The acceptance criteria for the Ginna Station RCL piping system are based upon the ANSI B31.1, Power Piping Code, 1967 edition including Summer 1973 Addenda (Reference 1) as used in the current Reference 4 design basis. That edition/addenda of the B31.1 Code, does not require a fatigue evaluation to be performed for the RCL piping system. Since there is no NRC requirement to meet code requirements beyond those in the code of record for the RCL piping, a fatigue evaluation of the RCL piping has not been performed.

Unlike for the RCL piping, for the pressurizer surge line there is a requirement for updating to a newer code. The pressurizer surge line was evaluated to the ASME B&PV Section III, Subsection NB 1986 Code (Reference 2), and includes the fatigue evaluation and the effects of thermal stratification as stipulated in *NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, December 20, 1988.* Bulletin 88-11 states: "...licensees of plants in operation over 10 years (i.e., low power license prior to January 1, 1979) are requested to demonstrate that the pressurizer surge line meets the applicable design codes* and other FSAR and regulatory commitments for the licensed life of the plant, considering the phenomena of thermal stratification and thermal striping in the fatigue and stress evaluations"; where Note * is "Fatigue analysis should be performed in accordance with the latest ASME Section III requirements incorporating high cycle fatigue". As a result of NRC Bulletin 88-11, the analysis code of record for only the surge line was required to be updated to incorporate fatigue evaluations.

The acceptance criteria for the pressurizer surge line thermal stratification analysis are per the ASME B&PV Code (Reference 2), and are as specified in the current design basis in WCAP-12928 (Reference 5).

The acceptance criteria for the primary equipment supports are based upon the ASME B&PV Code (Reference 3) as used in the current Reference 4 design basis.

Nuclear Steam Supply System Performance Capability Working Group Design Parameters

The design parameters for operation at 1817 MWt (NSSS) power, as identified in Table 1-1 of LR section 1.1, Nuclear Steam Supply System Parameters, were used in the thermal analysis of the RCL and used in the evaluation for the pressurizer surge line. The RCL was evaluated for two temperature cases – one for the lower-bound temperature case (Cases 1 and 2), and the second for the upper-bound temperature (Cases 3 and 4), as identified in Table 1-1. The above two thermal cases of the RCL were evaluated to envelope the RCL temperatures and the steam generator tube plugging data specified in Table 1-1.

NSSS Design Transients

The impact on design transients due to the changes in full-power operating temperatures for the EPU program is addressed in L R section 2.2.6, NSSS Design Transients. As used in the current RCL piping system design basis in Reference 4, the design criteria for the RCL piping is the ANSI B31.1 Power Piping Code, (1967 Edition including Summer 1973 Addenda – Reference 1); thus as explained above in the section titled "Input Parameters, Assumptions, and Acceptance Criteria", no fatigue analysis is required for the main RCL. For the pressurizer surge line, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by

ΔT between the pressurizer temperature and the hot-leg temperature. The controlling ΔTs for the pressurizer surge line are associated with the plant heatup and cooldown events which are not affected by the EPU program. It has been reviewed and shown that the temperatures and the design transients affected by the EPU have an insignificant effect on the pressurizer surge line analysis, including the effects of thermal stratification. Therefore, the EPU has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line, and the results in Reference 5 remain valid.

Loop LOCA Hydraulic Forcing Functions Forces and Associated Loop LOCA RPV Motions

The impact of the EPU Program on the Loop LOCA hydraulic forcing functions (HFFs) is addressed in LR section 2.8.5.6.3.5, Technical Evaluation – LOCA Forces, and the associated loop LOCA RPV motions are addressed in LR section 2.2.3, Reactor Pressure Vessel Internals and Core Supports. By virtue of LBB, breaks are not postulated for the RCL main loop piping, the pressurizer surge line, and the accumulator and residual heat removal (RHR) lines (see LR section 2.1.6, Leak-before-Break). Three pipe break LOCA cases and the associated RPV motions were evaluated for the LOCA analyses performed in the current design basis in Reference 4, namely the 10-inch accumulator line, the 10-inch pressurizer surge line, and the 10-inch RHR line. Subsequently, the application of LBB criteria exempted these three large diameter RCS pipe breaks from consideration. For the EPU program, the loop LOCA hydraulic forcing function forces and associated loop LOCA RPV motions from the smaller branch line breaks are used, namely the 3-inch pressurizer spray line on the cold leg, the 2-inch safety injection line on the hot leg, and the 4-inch upper plenum injection line connections to the vessel. The new analysis showed that the loop LOCA hydraulic forcing function forces and the associated loop LOCA RPV motion for the EPU program are bounded by the corresponding loop LOCA forces and RPV motions used in the current design basis LOCA analyses in Reference 4. Therefore the RCL piping LOCA analyses evaluation performed in Reference 4 remains applicable for the EPU program.

Description of Analyses and Evaluations

The design parameters that will change due to the EPU Program were reviewed for impact on the existing RCL piping and consequent impact to the auxiliary lines attached to the RCL centerline at the RCL branch nozzle connections.

As listed in the Ginna UFSAR, Section 3.9.1.2, the current design basis structural analysis of the RCL piping (Reference 4) was performed using the WESTDYN program . As presented in *WCAP-8252 Revision 1, Documentation of Selected Westinghouse Structural Analysis Computer Codes,* the WESTDYN computer code has a NRC SER formally approving the code and method of analysis. The WESTDYN computer program

was used for all applicable deadweight, thermal expansion, operational basis earthquake (OBE) and safe shutdown earthquake (SSE) cases. The Reference 4 design basis WESTDYN computer analysis models for these loading cases were updated to reflect the current as-built plant conditions and replacement steam generators and then used for the EPU analysis.

Like the pre-EPU and pre-RSG design basis analysis in Reference 4, the EPU structural evaluation of the RCL piping system included deadweight, thermal expansion, OBE, and SSE analyses. LOCA analyses from the current design basis in Reference 4 remain applicable for the EPU program.

The deadweight analysis for the EPU was performed considering the weight of the RCL piping and the primary equipment water weight.

The thermal analysis evaluated the RCL for the lower-bound temperature and the upper-bound temperature. The RCL was evaluated for two temperature cases – one for the lower-bound temperature case (Cases 1 and 2), and the second for the upper-bound temperature (Cases 3 and 4), as identified in Table 1-1 of LR section 1.1, Nuclear Steam Supply System Parameters. The above two thermal cases of the RCL were evaluated to envelope the RCL temperatures and the steam generator tube plugging data specified in Table 1-1.

The seismic analysis methods and the SSE and OBE input response spectra for the EPU parameters are the same as used in the current design basis analysis described in Reference 4. The seismic analyses performed considered two cases based on various primary equipment support activity as performed in the design basis analysis in Reference 4, and accounted for the range of operating temperatures as defined by the PCWG parameters for the EPU program in Licensing Report Table 1-1 of LR section 1.1, Nuclear Steam Supply System Parameters.

LOCA and pipe break analyses from the current design basis analysis in Reference 4 remain applicable for the EPU program as discussed in LR section 2.2.2.1, Input Parameters, Assumptions, and Acceptance Criteria, under the heading "Loop LOCA Hydraulic Forcing Functions Forces and Associated Loop LOCA RPV Motions".

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Aging evaluations were performed for the NSSS Piping, Components, and Supports in support of license renewal. These aging evaluations, WCAP-14575-A titled "Aging Management Evaluation for Class I Piping and Associated Pressure Boundary Components" dated December 2000, and WCAP-14422 titled "License Renewal Evaluation: Aging Management Evaluation for Reactor Coolant System Supports"

Revision 2-A, dated December 2000, were approved by the NRC in the License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant (NUREG-1786). Ginna has evaluated the impact of the EPU on these evaluations and the conclusions reached in the Ginna License Renewal Application for the NSSS Piping, Components, and Supports and has determined that the evaluations remain valid for the EPU conditions. In addition to the aging management evaluations, metal fatigue was identified as a TLAA in Section 4.3 of the Ginna License Renewal Application. TLAA (Time Limited Aging Analyses - Metal Fatigue) were performed for license renewal and are documented in the License Renewal Safety Evaluation Report (SER) in section 4.3 and incorporated into the Current Licensing Basis as part of license renewal. For the license renewal and as in section 3.1.2.3.8 in the License Renewal Safety Evaluation Report (SER), Fatigue Monitoring was required for selected locations and the NRC has approved the actions to manage the effects of aging during the period of extended operation on the functionality of the System Components subject to an Aging Management Review.

Ginna has evaluated the impact of the EPU on the fatigue evaluations performed in support of license renewal and has determined that the 40 year design transient set remains valid (bounding) for a 60 year operating term. Further, the fatigue analyses performed to support license renewal bounds and remains valid for the EPU conditions, and since the 40 year design transient set is valid (bounding) for 60 years of operation, the license renewal fatigue analysis bounds and remains valid for the EPU conditions for 60 years of operation.

Finally, as reported in LR section 2.14, Impact of EPU on the Renewed Plant Operating License, the environmental effects of fatigue were evaluated for the reactor vessel components based on the updated fatigue usage factors determined from the EPU evaluations. The cumulative usage factors for the pressurizer surge line, reactor coolant piping charging system nozzle, and reactor coolant piping safety injection nozzle are still below the ASME code limit of 1.0

NSSS Piping, Components, and Supports Results

Based on the evaluations performed for the EPU program NSSS PCWG design parameters, NSSS design transients, loop LOCA hydraulic forcing functions (HFFs) and associated RPV motions, it is concluded that there is no adverse effect on the current design basis RCL piping analyses and the current RCL piping design basis results in Reference 4 remain acceptable for the EPU Program.

The maximum RCL piping stresses for the EPU program and the corresponding code-allowable stress values are presented in Licensing Report Table 2.2.2.1-1, Maximum RCL Piping Stress Summary, and are the same as in the design basis in Reference 4. The stresses are combined in accordance with the methods used in the Reference 4

design basis and as specified in the code criteria as described in Reference 1. From the results tabulated in the Licensing Report in Table 2.2.2.1-1, it can be seen that the RCL piping stresses are within the allowable limits and meet the acceptance criteria (Reference 1) and are acceptable for the EPU program.

The applicable RCL piping primary equipment support loads for the EPU parameters were provided for evaluation and confirmation of acceptability (see LR section 2.2.2.3, Reactor Vessel and Supports, section 2.2.2.5, Steam Generator and Supports, and section 2.2.2.6, Reactor Coolant Pumps and Supports).

The primary equipment nozzle loads for the RSG were compared to the allowables as defined in the equipment design specifications. RCP and RPV nozzle loads were compared to loads previously evaluated and qualified for the design basis in Reference 4. The nozzle loads are acceptable and the EPU Program has no adverse impact on the analysis results.

The applicable RCL piping loads resulting from the range of operating temperatures, as defined by the EPU NSSS PCWG parameters in Table 1-1 of LR section 1.1, Nuclear Steam Supply System Parameters, were provided for evaluation and confirmation of LBB (see LR section 2.1.6, Leak-before-Break).

The impact of the EPU program parameters on RCL piping displacements at the intersection of the centerline of the RCL piping and the auxiliary line piping system branch nozzle connections is insignificant. Therefore, the Auxiliary piping systems attached to the RCL are not significantly affected by the EPU program.

For the pressurizer surge line, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by ΔT between the pressurizer temperature and the hot-leg temperature and has been evaluated.. The controlling ΔTs for the pressurizer surge line are associated with the plant heatup and cooldown events which are not affected by the EPU program.

As discussed in LR section 2.2.2.1, under the heading Input Parameters, Assumptions, and Acceptance Criteria, the current design basis pressurizer surge line analysis results in Reference 5, including the effects of thermal stratification, are applicable for the EPU program and meet the acceptance criteria for the EPU program.

NSSS Piping, Components, and Supports References

1. ANSI B31.1, *Power Piping Code*, 1967 Edition, including Summer of 1973 Addenda.

2. *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code,* Section III, Subsection NB, 1986 Edition.

3. *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code*, Section III, Subsection NF and Appendix F, American Society of Mechanical Engineers, 1974 Edition.

4. Westinghouse Letter: SE&PT-CSE-3041, *Final Report for the Robert Emmett Ginna Nuclear Generating Station Steam Generator Hydraulic Snubber Replacement Program*, October 1, 1992 (Final Report attached to letter, *Evaluation Of The Reactor Coolant System For The Steam Generator Hydraulic Snubber Replacement Program*, September 1992.).

5. WCAP-12928, *Structural Evaluation of Robert E. Ginna Pressurizer Surge Line, Considering the Effects of Thermal Stratification*, May 1991.

NSSS Piping, Components, and Supports Conclusions

The parameters associated with the EPU program have been evaluated for their effects on the following:

- RCL piping stresses
- RCL piping system LBB loads for LBB evaluation
- RCL piping displacements at the junction of the centerline of the RCL piping and the branch nozzle connections of the Auxiliary piping systems to the RCL and Impact on auxiliary piping systems
- Primary equipment nozzle loads
- Pressurizer surge line piping analysis including the effects of thermal stratification
- Primary equipment support loads (Reactor Vessel, Steam Generator, and Reactor Coolant Pump)

The evaluation determined that the parameters associated with the EPU program have no adverse effect on the analysis of the RCL piping system, including impacts to the primary equipment nozzles. RCL piping stresses meet the required stress criteria as summarized in Table 2.2.2.1-1, Maximum RCL Piping Stress Summary. The primary equipment nozzle loads are all acceptable. RCL piping loads transmitted for LBB evaluation for the EPU Program are evaluated in LR section 2.1.6, Leak-before-Break.

The RCL primary equipment support loads meet the required stress criteria as summarized in the LR section 2.2.2.3, Reactor Vessel and Supports, section 2.2.2.5, Steam Generator and Supports, and section 2.2.2.6, – Reactor Coolant Pumps and Supports.

RCL piping displacements at branch nozzles due to the EPU had no significant impact on the auxiliary piping systems that are attached to the RCL (as applicable) and are not affected by the EPU program.

Review has shown that the temperatures and the design transients affected by the EPU have an insignificant effect on the pressurizer surge line design basis analysis, including the effects of thermal stratification. Therefore, the EPU has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line, and the results in Reference 5 remain valid.

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the NSSS Piping, Components, and Supports. The aging evaluations approved by the NRC in the License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant (NUREG-1786) for the NSSS Piping, Components, and Supports remain valid for the EPU conditions. As approved by the NRC per section 4.3 in the License Renewal Safety Evaluation Report (SER), and as in WCAP-14575-A titled "Aging Management Evaluation for Class I Piping and Associated Pressure Boundary Components" dated December 2000, and as in WCAP-14422 titled "License Renewal Evaluation: Aging Management Evaluation for Reactor Coolant System Supports" Revision 2-A, dated December 2000, the evaluations remain valid for the EPU conditions.

The existing fatigue analysis bounds and remains valid for the EPU conditions, and since the 40 year design transient set is valid (bounding) for the 60 years of operation, the existing fatigue analysis bounds and remains valid for the EPU conditions for 60 years of operation.

Further details can also be obtained from the LR section 2.14, Impact of EPU on the Renewed Plant Operating License.

For the impact on the Environmentally Assisted Fatigue Evaluations, see LR section 2.14.3 titled "Impact of EPU on Environmentally Assisted Fatigue Evaluations".

Ginna has identified the aging effects and the aging management programs credited for managing the effects on the RCS (Class I). The programs were verified as adequate under the EPU conditions, thus there is reasonable assurance that the components' intended function will be maintained consistent with the Current Licensing Basis for the period of extended operation.

Therefore, the EPU conditions, and the analyses and evaluations performed in support of the EPU, do not impact the aging management reviews, aging management programs, and TLAAs associated with the NSSS Piping, Components, and Supports for the Ginna Station License Renewal. The NSSS Piping, Components, and Supports continue to meet the current licensing basis with respect to GDC-1, 2, 4, 14, and 15.

Table 2.2.2.1-1
Maximum RCL Piping Stress Summary (from Reference 4) (Based on $K_{Average}$ *)**

ANSI B31.1 Code Equation	RCL Piping	Actual Piping Stress For EPU (ksi)	ANSI B31.1 Code Allowable Stress (ksi)	Percentage of Allowable
Normal – Equation 11 Design Pressure + Deadweight	Hot Leg	[ ]a,c	16.8	[ ]a,c
	Crossover Leg	[ ]a,c	16.8	[ ]a,c
	Cold Leg	[ ]a,c	16.8	[ ]a,c
Upset – Equation 12 Design Pressure + Deadweight + OBE	Hot Leg	[ ]a,c	20.1	[ ]a,c
	Crossover Leg	[ ]a,c	20.1	[ ]a,c
	Cold Leg	[ ]a,c	20.1	[ ]a,c
Emergency – Equation 12 Design Pressure + Deadweight + SSE	Hot Leg	[ ]a,c	30.2	[ ]a,c
	Crossover Leg	[ ]a,c	30.2	[ ]a,c
	Cold Leg	[ ]a,c	30.2	[ ]a,c
Faulted – Equation 12 Design Pressure + Deadweight + [(SSE)2 + (DBPB*)2] 1/2	Hot Leg	[ ]a,c	40.3	[ ]a,c
	Crossover Leg	[ ]a,c	40.3	[ ]a,c
	Cold Leg	[ ]a,c	40.3	[ ]a,c
Maximum Thermal – Equation 13 Maximum Thermal Stress Range** + Seismic Anchor Motion OBE Displacements	Hot Leg	[ ]a,c	27.5	[ ]a,c
	Crossover Leg	[ ]a,c	27.5	[ ]a,c
	Cold Leg	[ ]a,c	27.5	[ ]a,c
Normal + Maximum Thermal – Equation 14 Design Pressure + Deadweight + Maximum Thermal Stress Range** + Seismic Anchor Motion OBE Displacements	Hot Leg	[ ]a,c	44.4	[ ]a,c
	Crossover Leg	[ ]a,c	44.4	[ ]a,c
	Cold Leg	[ ]a,c	44.4	[ ]a,c

Note:
* DBPB = Design-Basis Pipe Break.
** Loss-of-load overtemperature transient effects are included (Reference 4).
*** $K_{average}$ is the average of the support stiffnesses.

2.2.2.2 Balance of Plant Piping and Supports (Non-Class 1)

Introduction

This section of the Licensing Report covers piping and supports that are not included in section 2.2.2.1, NSSS Piping, Components and Supports. Section 2.2.2.1 covers Class 1 reactor coolant loop piping and supports up to the class break at the containment isolation boundary. This section covers Non-Class 1 piping and its supports, whether inside or outside containment.

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to BOP piping components and supports.

Other Ginna UFSAR sections that discuss the design of BOP and Non-class 1 piping and supports include:

- Ginna UFSAR section 3.2, Classification of Structure, Components, and Systems, provides details with respect to the seismic classification of piping and piping components.
- Ginna UFSAR section 3.7, Seismic Design, and specifically section 3.7.3.1, Seismic Analysis Methods, provides details with respect to the seismic qualification of piping and piping components.
- Ginna UFSAR section 3.9, Mechanical System and Components, and specifically sections 3.9.2.1.8, Seismic Piping Upgrade Program and 3.9.2.1, Dynamic Testing and Analyses - Piping Systems, provide details with respect to the seismic qualification of piping and piping components.

Technical Evaluation

Introduction

All BOP and Non-class 1 piping and support systems were evaluated to assess the impact of operating temperature, pressure and flow rate changes that will result due the implementation of EPU. The BOP and Non-class 1 piping affected by EPU were evaluated to the ANSI B31.1 –1973 Code for Power Piping through Summer 1973 Addenda, which is the current code of record for Ginna Station BOP and Non-class 1 piping and support systems as described in UFSAR sections 3.7.3.7.4 and 3.9.2.2.1.

The Ginna Station BOP and Non-class 1 piping and support systems that were evaluated for EPU conditions included the following systems:

- Main Steam
- Feedwater
- Condensate
- Heater Drains
- Moisture Separator Drains
- Extraction Steam
- Circulating Water
- Component Cooling Water
- Auxiliary Feedwater
- Spent Fuel Pool Cooling
- Service Water
- Steam Generator Blowdown
- Radwaste Systems (Liquid and Gaseous)
- Safety Injection
- Containment Spray
- Chemical and Volume Control
- Residual Heat Removal
- Sampling
- Pressurizer Safety & Relief Valve Piping to Pressurizer Relief Tank

For piping systems with a class 1/Non-class 1 interface, an assessment of the class 1 piping and support systems to identify any potential impact that would be imposed upon the Non-class 1 piping due to EPU conditions was performed. Based on the class 1 piping and support evaluations that were performed, no significant impact was identified on the associated Non-class 1 piping and supports due to EPU conditions. Refer to LR section 2.2.2.1, NSSS-Piping and Support (Class 1).

Description of Analyses and Evaluations

System operation at EPU conditions generally results in increased pipe stress levels and pipe support and equipment loads when those SSC's experience higher operating temperatures, pressures or flow rates.

Pre-uprate and EPU operating data (operating temperature, pressure and flow rate) were obtained from heat balance diagrams, calculations, and/or other reference documents.

Thermal "change factors" were determined, as required, to compare and evaluate changes in thermal operating conditions. The thermal "change factors" were based on the following ratio:

- The thermal "change factor" equals the ratio of the power uprate to pre-uprate operating temperature. That is, thermal change factor is $(T_{uprate}-70°F)/(T_{pre\text{-}uprate}-70°F)$.

Based on the magnitude of the calculated thermal change factors, the following engineering activities were performed and/or conclusions reached.

For thermal change factors less than or equal to 1.00 (that is, the pre-uprate condition envelopes or equals the EPU condition), the piping and support system was concluded to be acceptable for EPU conditions.

For thermal change factors greater than 1.00, an additional evaluation was performed to address the specific increase in temperature, in order to determine piping and support system acceptability.

Operating pressure increases due to EPU were very small and mostly affected systems related to the main power cycle (main steam, condensate, feedwater, extraction steam, heater drains). Since the pipe stress evaluations for piping systems at Ginna Station have used the system design pressure in accordance with ANSI B31.1, the small increases in operating pressures were acceptable, as long the EPU operating pressure remains within the current design pressure of the system.

Flow rate increases due to EPU occur mainly in systems related to the main power cycle. The two piping systems of most concern with respect to flow rate increases are main steam and feedwater systems. Flow rate increases and their impact on potential flow induced fluid transient loads were evaluated for the main steam and feedwater piping systems. The evaluation of the main steam system addressed the system flow rate increase and its impact on fluid transient loads (i.e., steam hammer loads) resulting from a turbine stop valve closure event. The evaluation of the feedwater system addressed the flow rate increase and its impact on fluid transient loads (i.e., water hammer loads) resulting from feedwater regulatory valve closure/feedwater pump trip events. The remaining piping systems potentially impacted by EPU do not contain any fast closing valves and have not experienced significant flow induced transients. Hence, flow rate increases due to EPU for these piping systems were determined to be acceptable with respect to potential water hammer loading events.

The turbine cross-over and cross-under piping are addressed in LR section 2.5.5.1, Main Steam.

Changes in piping operating temperatures due to revised heat exchanger heat load requirements (e.g., component cooling and service water heat exchangers) have been considered as part of the piping and support system evaluations.

The BOP and Non-class 1 piping and support systems affected by EPU were evaluated to the ANSI B31.1 –1973 Code for Power Piping through Summer 1973 Addenda, which is the current code of record for BOP and Non-class 1 piping systems as described in UFSAR sections 3.7.3.7.4 and 3.9.2.2.1.

There was no change to seismic inputs (amplified response spectra) or loads resulting from EPU. The existing seismic design basis for all piping and supports remains valid and unaffected by EPU. Hence, BOP and Non-class 1 piping and support seismic loadings will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-2.

The following computer programs were used in performing the EPU piping and pipe support evaluations. These computer programs are not currently described in the UFSAR and were used in performing the EPU piping and support evaluations. These programs are used to calculate stresses and loads using the appropriate equations from the ANSI B31.1 criteria. Using an approved Quality Assurance program these computer programs have been verified and validated and shown to be accurate. Thus the programs are appropriate for use in QA category 1 Nuclear Safety Related applications and can be incorporated into the plant current licensing basis without prior NRC permission.

Computer Program	Program Description
STEHAM-PC	The STEHAM-PC program was used to determine forcing functions for the main steam turbine stop valve closure event. This program determines forcing functions in steam-filled piping systems due to valve opening and closing conditions
WATHAM-PC	The WATHAM-PC program was used to determine forcing functions for the feedwater regulatory valve closure and feedwater pump trip events. This program determines forcing functions in water-filled piping systems due to pump start/stop/trip, check valve closure and valve opening and closing events
PSAP-PC	The PSAP-PC program was used to perform a steady state analysis of a water filled flow system
NUPIPE-SWPC	The NUPIPE-SWPC program was used to perform detailed pipe stress analysis. This program is designed to perform analyses in accordance with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components and the ANSI/ASME B31.1 Power Piping Code
PC-PREPS	The PC-PREPS program was used to perform detailed pipe support evaluations. This program performs a complete structural analysis, performing an AISC code check, weld qualification and baseplate/anchor bolt qualifications
PILUG-PC	The PILUG-PC program was used to perform piping evaluations. This program is a stress analysis program used to calculate stress intensity at the junction of a rectangular attachment perpendicular to round pipe

For BOP and Non-class 1 piping and support systems that required detailed analyses to reconcile EPU operating parameters, a summary of revised stress levels corresponding to EPU conditions is provided in Table 2.2.2.2-1. The results presented include existing stress levels (i.e., pre-uprate), revised pipe stress levels for EPU conditions, allowable

stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile SPU conditions. The design margin provided is based on the ratio of the calculated stress divided by the allowable stress.

Other evaluations of issues that potentially impact BOP and Non-class 1 piping and supports are addressed in the following LR section.

- Protection against dynamic effects, including GDC-4 requirements, of pipe whip and discharging fluids – LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures.

- Protection against internally generated missiles and turbine missiles, including GDC-4 requirements, is discussed in LR section 2.5.1.2, Missile Protection.

- Design of the Reactor Coolant System and related components, including GDC-15 requirements, is discussed in LR section 2.2.2.1, NSSS Piping and Supports, Class 1.

Results

During the review of the present piping stress analysis design bases for the Service Water and Component Cooling Water Systems, some inconsistencies were identified between the operating temperatures assumed in the analyses and the maximum possible operating temperatures. The impact of these differences in operating temperature upon the piping thermal stresses has been evaluated. The evaluations have determined that the existing piping design is acceptable due to the flexibility of the piping systems and high thermal stress margins available in the existing analyses. No plant changes to the existing piping system are expected due to this issue. A Corrective Action was initiated to revise the design basis analysis for all affected pipe segments.

For piping systems which will experience plant modifications (e.g., MSR piping and relief valve modification) to address EPU conditions, the piping and support evaluations will be performed as part of the overall design change package associated with the specific plant modification.

Table 2.2.2.2-1 provides a summary of existing stress levels (i.e., pre-uprate), revised pipe stress levels for EPU conditions, and the resulting design margin for each piping analysis that required detailed evaluation to reconcile EPU conditions. Piping systems not specifically listed in Table 2.2.2.2-1 did not require detailed evaluation (i.e., no significant operating parameter increases due to EPU) to reconcile EPU conditions. The stress results reported have incorporated thermal expansion and fluid transient increases, as applicable, that were reconciled as part of the EPU evaluations.

Table 2.2.2.2-1 Stress Summary at EPU Conditions					
Piping Analysis Description	**Loading Condition**	**Existing Stress (psi)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin** (Note 1)
Main Steam Inside Containment Loop A	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	16,444	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	24,665	[ ]a,c
Main Steam Inside Containment Loop B	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	16,444	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	24,665	[ ]a,c
Main Steam Outside Containment	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	16,444	[ ]a,c
(Containment Penetrations 401 & 402 to Anchor MSU-35)	Equation 12U (Occasional) (3" Bypass)	[ ]a,c	[ ]a,c	18,000	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	24,665	[ ]a,c
	Equation 12F (Occasional) (3" Bypass)	[ ]a,c	[ ]a,c	27,000	[ ]a,c
Main Steam Outside Containment (Anchor MSU-35 to Turbine Stop Valve Nozzle Connections)	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	16,440	[ ]a,c

Table 2.2.2.2-1 Stress Summary at EPU Conditions					
Piping Analysis Description	**Loading Condition**	**Existing Stress (psi)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin** (Note 1)
Feedwater Inside Containment Loop A	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	21,000	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	31,500	[ ]a,c
Feedwater Inside Containment Loop B	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	21,000	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	31,500	[ ]a,c
Feedwater Outside Containment (Anchor FWU-28 to Containment Penetration 404)	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	21,000	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	31,500	[ ]a,c
Feedwater Outside Containment (Anchor FWU-28 to Containment Penetration 403)	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	21,000	[ ]a,c
	Equation 12F (Occasional)	[ ]a,c	[ ]a,c	31,500	[ ]a,c
Feedwater Outside Containment (Feedwater Pumps Discharge to Anchor FWU-28)	Equation 12U (Occasional)	[ ]a,c	[ ]a,c	21,000	[ ]a,c
Condensate (Heater 2A to Heater 3A)	Equation 13 (Thermal)	[ ]a,c	[ ]a,c	22,500	[ ]a,c

Table 2.2.2.2-1 Stress Summary at EPU Conditions					
Piping Analysis Description	**Loading Condition**	**Existing Stress (psi)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin** (Note 1)
Condensate (Heaters 3A/B to Heaters 4A/B)	Equation 13 (Thermal)	[ ]a,c	[ ]a,c	22,500	[ ]a,c
Extraction Steam to Feedwater Heaters 4A/B	Equation 13 (Thermal)	[ ]a,c	[ ]a,c	22,500	[ ]a,c
Feedwater Heater Drain Piping From Heaters 5A to 4A	Equation 14 (Thermal + Sustained)	[ ]a,c	[ ]a,c	37,500	[ ]a,c
Moisture Pre-separator Drains	Equation 13 (Thermal)	[ ]a,c	[ ]a,c	22,500	[ ]a,c
Moisture Separator Drains (normal drains)	Equation 13 (Thermal)	[ ]a,c	[ ]a,c	22,500	[ ]a,c

NOTES:

(1) Design Margin reported is based on the ratio of EPU stress divided by the Allowable stress.

The piping stress evaluations performed conclude that all piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from EPU conditions. The piping evaluations concluded that the main steam system can withstand the steam hammer loads associated with EPU conditions (resulting from a turbine stop valve closure event) and the feedwater system can withstand the water hammer loads associated with EPU conditions (resulting from a feedwater regulatory valve closure/pump trip event).
The results of the pipe support evaluations for systems impacted by EPU concluded that all supports remain acceptable, except for certain main steam and feedwater system pipe supports that require modification to accommodate the revised loads related to EPU conditions. The main steam and feedwater pipe support modifications are required to mitigate the larger flow induced fluid transient loads that resulted due to EPU conditions. The majority of these support modifications are required to mitigate the larger loads resulting from a turbine stop valve closure transient event. Also, one new snubber will also be installed on the main steam piping system. These pipe support modifications will be installed before the implementation of the EPU.

The results of the equipment nozzle and containment penetration evaluations concluded that these components remain within acceptable limits for EPU conditions.

In summary, the BOP and Non-class 1 piping and support systems will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-1, 2, 4, 14 and 15.

Additionally, the implementation of EPU will result in higher flow rates for several piping systems. Piping systems experiencing these higher flow rates will be reviewed for potential vibration issues. Potentially affected piping will be included as part of the start-up testing program related to the overall implementation of EPU. Refer to LR section 2.12 for discussion of the Power Ascension and Testing Plan.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

With respect to the pipe support modifications for the main steam and feedwater piping systems, these modifications will not impact the License Renewal system evaluation boundaries. The required pipe support modifications do not add any new or previously unevaluated materials to the BOP and Non-class 1 piping and support systems. No new aging effects requiring management are identified.

Piping system internal and external environments remain within the parameters previously evaluated. The Time-Limited Aging Analyses evaluations remain bounding. The existing fatigue evaluation for the main steam containment penetration remains acceptable, since the input parameters used in the evaluation remain bounding.

Conclusion

The EPU will result in pipe support modifications for the main steam and feedwater piping systems. These pipe support modifications will not impact the License Renewal system evaluation boundaries. The required pipe support modifications do not add any new or previously unevaluated materials to the BOP and Non-class 1 piping and support systems. No new aging effects requiring management are identified.

The evaluations have addressed the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the BOP and Non-class 1 piping evaluations have adequately addressed the effects of the proposed EPU on BOP and Non-class 1 piping components and their supports. Based on the above, it is concluded that the evaluations have demonstrated that pressure-retaining components and their supports will continue to meet the Ginna Station current licensing basis with respect to the requirements of 10CFR50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15. Therefore, the proposed EPU is acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.2.3 Reactor Vessel and Supports

Introduction

The Ginna Station reactor pressure vessel (RPV), as the principal component of the reactor coolant system (RCS), contains the heat-generating core and associated supports, controls, and instrumentation, and coolant circulating channels. Primary outlet and inlet nozzles provide for the exit of heated coolant and its return to the RPV for recirculation through the core.

The Ginna Station RPV consists of a cylindrical shell with a hemispherical bottom head and a flanged and gasketed removable upper head. The RPV shell is fabricated from integral ring forgings joined by circumferential welds. The RPV contains the core, core support structures, rod control clusters, thermal shield, and other parts directly associated with the core. Inlet and outlet nozzles are located at an elevation between the head flange and the core. The body of the RPV is low-alloy carbon steel, and the inside surfaces in contact with coolant are clad with austenitic stainless steel to minimize corrosion. The RPV is supported by steel pads integral with the four coolant nozzles and by two external support brackets. The pads and brackets rest on steel base plates attached to a concrete support structure. All six reactor vessel supports are equipped with cooling to minimize the transmission of heat to the concrete. A structural report on the reactor vessel and vessel supports, WCAP-16411 (Reference 1), was issued in support of the Ginna Station EPU Program. The report presents revisions to the Babcock & Wilcox (B&W) Company Original Reactor Vessel Stress Reports, performed under B&W Contract number 610-0110, for the reactor vessel components/regions below the vessel flange and it presents revisions to the BWC Design and Service Loading Condition Report No. BWC-083N-SR-01, Revision 1, for the Ginna Station replacement reactor vessel closure head (RRVCH). The report also presents the evaluation of the reactor vessel support loads. The structural report is necessary to evaluate the stress and fatigue effects of the revised operating parameters and RCS transients associated with the EPU. The report evaluates the maximum primary-plus-secondary stress intensity ranges, the maximum cumulative fatigue usage factors, and the vessel support loads resulting from the revised operating parameters and transients of the EPU.

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to NSSS piping, components, and supports, with the following amplifications. The Ginna Station reactor vessel was designed and fabricated by Babcock & Wilcox (B&W) Company in accordance with Westinghouse specifications and ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition requirements. The reactor vessel is described in the Ginna UFSAR, Section 5.3; the governing specifications are listed in Table 5.3-1

A replacement reactor vessel closure head was installed at Ginna Station during the Fall 2003 refueling outage. The replacement closure head was procured by PCR 2001-0042 through

B&W of Canada, Ltd. (BWC) in accordance with technical specification BWC-TS-2915. The replacement head was designed and fabricated in accordance with ASME Code, Section III, 1995 Edition, with 1996 Addenda, Class 1 requirements.

In addition to the basis described in the regulatory evaluation section above, the reactor vessel and vessel supports were evaluated for plant license renewal. The results of the NRC review of the reactor vessel are documented in section 3.1.2 of NUREG-1786, *License Renewal Safety Evaluation Report (SER) for the R. E. Ginna Nuclear Plant*, May 2004. The results of the NRC review of the vessel supports are documented in section 3.5 of the SER. Time Limited Aging Analysis (TLAA) and the programs used to manage aging were evaluated in those sections.

Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

Presented in the structural report are the analyses and evaluations necessary per Section III of the *ASME Boiler and Pressure Vessel Code* to substantiate the structural adequacy of the Ginna Station reactor vessel for operation under EPU conditions.

The analyses and evaluations performed in the structural report for the replacement reactor vessel closure head incorporate into the BWC replacement head design and service loading condition report revised operating temperatures, revised RCS transients, and revised seismic and LOCA reactor vessel/internals interface loads associated with the EPU.

The analyses and evaluations performed in the structural report for the vessel components below the vessel flange and for the vessel supports incorporate into the B&W Company original reactor vessel stress reports the revised operating temperatures, RCS transients, and revised seismic and LOCA reactor vessel/internals interface loads associated with the Ginna Station EPU and Replacement Steam Generator (RSG) Programs.

Analysis of flow induced vibration is not included in the licensing basis for Ginna. However, it was considered for more susceptible components that would experience a significant flow increase under EPU conditions. Reactor vessel components were evaluated and deemed unaffected by EPU conditions due to their heavy construction and small increase in flow, if any.

Assumptions

The following assumptions were made in performing the evaluations of the reactor vessel components:

- The Ginna Station reactor vessel components/regions below the vessel flange can be evaluated by using the stress report for a similar plant since the vessel and components are, in most cases, nearly identical. For those components that are not identical, the differences are reconciled or a different methodology is used to update stresses.

- The thermal stresses due to rapid transients can be calculated by using the methods outlined in Document PB-151987, *Tentative Structural Design Basis for Reactor Pressure Vessels*. Section A.3.5, since the 1965 ASME Code, Section III, did not have a satisfactory analytical method for determining thermal stresses due to rapid transients. The stress formulas and curves from PB-151987 were the best available analytical data for determining those stresses and were eventually adopted by the ASME Code.

- The changes in pressure during the design transients can be determined by scaling the known transient pressure stresses proportional to the pressure changes. The relationship between pressure and stress due to pressure is linear.

Acceptance Criteria

Revised maximum stress intensity ranges and cumulative fatigue usage factors were calculated and compared to the following acceptance criteria:

- The maximum range of primary-plus-secondary stress intensity resulting from mechanical and thermal loads shall not exceed $3S_m$ at operating temperature. In lieu of satisfying $3S_m$, the design shall be considered acceptable if the criteria specified for a plastic analysis per paragraph N-417.6(a)(2) of the ASME B&PV Code, Section III, Division 1, 1995 Edition, or the criteria specified for a simplified elastic-plastic analysis per Section NB-3228.5 of the ASME B&PV Code, Section III, Division 1, 1995 Edition through 1996 Addenda can be met.

- The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads cannot exceed 1.0 in accordance with the procedure outlined in Paragraph N-415.2 of the ASME B&PV Code, Section III, Division 1, 1965 Edition for the vessel components below the vessel flange and, in Paragraph NB-3222.4 of the ASME B&PV Code, Section III, Division 1, 1995 Edition through 1996 Addenda for the components of the replacement closure head and main closure region.

Description of Analyses and Evaluations

The Ginna Station reactor vessel recently underwent a closure head replacement program in which the closure head and the control rod drive mechanism (CRDM) and vent nozzles were replaced. As a result, for the Ginna Station closure head and main closure region components, the BWC replacement head stress report was used as the baseline for updating stresses and fatigue usage factors. The BWC report analyzed the components to the operating conditions and design transients of the Ginna Station Replacement Steam Generator (RSG) Program. The structural report (Reference 1) updated the stress intensities and fatigue usage factors of the closure head and main closure region components based on the changes that will occur in going from the Ginna Station RSG Program to the Ginna Station EPU.

For the evaluation of the Ginna Station closure head, CRDM and vent nozzles, and main closure region components including the vessel flange and closure studs, a graphical transient comparison as well as a transient comparison using one-dimensional heat transfer analyses were first performed in which the revised RCS design transients for the Ginna Station EPU were compared to the RSG design transients. This review determined which EPU transients are more severe than their design basis (RSG) counterparts by comparing rates, magnitudes and durations of the transient temperature variations as well as the magnitudes of the pressure variations. Scaling factors were developed based on the results of the graphical comparisons and analyses and those scaling factors were then applied in the stress and fatigue evaluations of the closure head, CRDM and vent nozzles, and main closure region components. There are several T_{hot} and pressure variations that were evaluated to be more severe for the Ginna EPU transients than for the Ginna RSG transients. The T_{hot} variations that were determined by the one-dimensional heat transfer analyses to be more severe for the Ginna EPU transients than for their counterparts of the RSG program and the component(s) to which that variation applies are listed below.

- Unit Loading at 5% per minute (all components)
- Unit Unloading at 5% per minute (all components)
- 10% Step Load Increase (vent nozzles)
- 10% Step Load Decrease (vent nozzles)
- Large Step Load Decrease (i.e. Step Reduction) (all components)
- Loss of Load (CRDM and vent nozzles)
- Reactor Trip (vessel flange, studs, CRDM nozzles, vent nozzles)

The pressure variations that were determined by the graphical comparisons to be more severe for the Ginna EPU transients than for their counterparts of the RSG program, and the component(s) to which that variation applies, are listed below.

- 10% Step Load Increase (all components)
- Large Step Load Decrease (i.e. Step Reduction) (all components)

For the Ginna Station reactor vessel and related components below the vessel flange, a stress report for a similar plant was used as the baseline instead of the original Ginna stress reports.

There are several reasons this approach was used. First, the original Ginna stress reports analyzed the vessel and related components to "lumped" or enveloping transient information, thereby making it difficult to use them as a baseline for updating stresses and fatigue usage factors on a transient by transient basis. Second, the approach of using a few enveloping transients is generally very conservative and it was believed that compounding the conservatism would eventually lead to little or no stress or fatigue usage margins for future uprate programs. Finally, it was easy to justify the use of the components of the similar plant as a baseline since the geometry and the material properties for its vessel components are essentially identical, with three exceptions, to those of the Ginna Station vessel. The three exceptions are the geometries of the safety injection nozzle, the bottom head instrumentation tubes, and the core support guides. The evaluation of the safety injection nozzle concludes that the dimensional differences will have no effect on the critical locations since they are remote from those locations. Due to the difference in the outside diameters of the bottom head instrumentation tubes between the similar plant and Ginna and the importance of the outside diameter and the penetration size to the J-groove weld stresses, the B&W original stress report methodology, not the "similar plant" approach, was used to calculate the stress intensity range and fatigue usage factor for the bottom mounted instrumentation nozzles. With regard to the core support guides, the equations and methodology for the similar plant were used but the geometry of the Ginna guides was used to determine the stresses and fatigue usage factors.

Prior to using the "similar plant" approach for the evaluation of the vessel components below the vessel flange, a transient review was performed in which the RCS design transients for the Ginna Station EPU, the Ginna Station RSG Program, and the Ginna Station original design basis (ODB) were compared to the design-basis transients for a similar plant. This transient review determined which Ginna Station transients are more severe than their baseline design transient counterparts by comparing rates, magnitudes, and durations of the transient temperature variations as well as the magnitudes of the pressure variations. Based on this review, a determination of which Ginna transients must be considered in the stress and fatigue evaluations was made. There are several Thot, Tcold, and pressure variations that were evaluated to be more severe for the Ginna ODB, RSG, and EPU transients than for the transients of the similar plant. These variations were considered in the reactor vessel structural evaluation for the EPU. However, a variation may not have been used in the evaluation if the variation does not govern at the time at which the critical stress occurs.

The following seven (7) Thot variations were determined by the transient review to be more severe for the Ginna transients than for their counterparts of the similar plant. The Ginna operating basis for the specific transient listed below is indicated in parentheses.

- Unit Loading at 5% per minute (ODB)
- Unit Unloading at 5% per minute (ODB)
- 10% Step Load Increase (ODB and RSG)
- 10% Step Load Decrease (RSG)

- Large Step Load Decrease (i.e. Step Reduction) (EPU)
- Loss of Flow (ODB)
- Loss of Load (RSG and EPU)

The following seven (7) Tcold variations were determined by the transient review to be more severe for the Ginna transients than for their counterparts of the similar plant. The Ginna operating basis for the specific transient listed below is indicated in parentheses.

- Unit Loading at 5% per minute (EPU)
- Unit Unloading at 5% per minute (EPU)
- 10% Step Load Increase (RSG)
- Large Step Load Decrease (EPU)
- Loss of Load (RSG and EPU)
- Reactor Trip (RSG)

The following four (4) pressure variations were determined by the transient review to be more severe for the Ginna transients than for their counterparts of the similar plant. The Ginna operating basis for the specific transient listed below is indicated in parentheses.

- 10% Step Load Increase (ODB)
- 10% Step Load Decrease (EPU)
- Large Step Load Decrease (RSG)
- Loss of Load (RSG)

Seismic and LOCA reactor vessel/internals interface loads for the EPU were reviewed for the barrel outlet nozzle and lower radial keys. A comparison of these loads and the allowable loads defined was performed in the reactor vessel evaluation.

The stress intensities for those transients that were deemed more severe than their baseline counterparts were examined to determine their effect on the maximum ranges of stress intensity for all the regions of the reactor vessel. The changes in the thermal and pressure stresses, due to adverse changes in temperature and/or pressure variations from the baseline transients, were evaluated using standard engineering approaches. The incremental thermal and pressure stress changes were then factored into stress intensities reported in the baseline stress report(s) and the effects of the changes on the maximum ranges of stress intensity were observed.

The peak stress intensity ranges for the fatigue evaluation were also adjusted to account for the incremental thermal and pressure stress changes caused by adverse changes from the baseline transients. The peak thermal and pressure stresses were multiplied by the appropriate scaling factor, where necessary, before determining a new peak stress intensity range and finally an alternating stress. The allowable number of cycles of alternating stress was found

from the applicable fatigue curve in either the ASME B&PV Code, Section III, Division 1, 1965 Edition, or the ASME B&PV Code, Section III, Division 1, 1995 Edition through 1996 Addenda, and the cumulative fatigue usage factors were revised accordingly.

Where applicable, the maximum and minimum stress intensity ranges and fatigue usage factors were revised to reflect the adverse changes to the baseline transients. In other cases, the baseline stress analysis in the baseline stress report remained conservative with regard to the design transients and new calculations were not necessary. For those cases, the maximum stress intensity ranges and fatigue usage factors reported in the baseline reactor vessel stress report were not changed.

In the end, if the "similar plant" approach used for the vessel components below the vessel flange yielded less conservative results than those reported in the UFSAR, then the original analyses for those components were still considered as bounding.

In addition to the evaluations described above for the reactor vessel components, the reactor vessel support loads for the EPU were also evaluated by comparing them to the maximum acceptable loads determined by the Gilbert Associates, Inc. analysis which qualified the vessel supports.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor vessel and supports. There are several TLAAs associated with the reactor vessel. From Table 3.1-1 of NUREG-1786, the TLAAs listed below were reviewed and evaluated for the reactor vessel.

- Cumulative fatigue damage of all reactor vessel components
- Loss of fracture toughness due to neutron irradiation embrittlement of RV materials
- Crack initiation and growth due to PWSCC of CRD nozzles
- Crack initiation and growth due to SCC and PWSCC of RV nozzle safe ends and CRD housing
- Loss of material due to wear of RV closure studs and stud assembly, and core support pads

There are several TLAAs associated with the reactor vessel supports. The particular age-related degradation mechanisms associated with the reactor vessel supports are:

- Stress Corrosion Cracking
- Corrosion
- Neutron Embrittlement

- Thermal Aging Embrittlement
- Mechanical Wear
- Fatigue
- Creep and Stress Relaxation
- Concrete Degradation
- Gamma Heating

This LR section addresses the maximum stress intensity ranges and cumulative fatigue damage for all reactor vessel components considering the impact of EPU conditions on license renewal and evaluates those ranges and fatigue damage against the ASME code limits. This section also addresses the vessel support loads considering the impact of EPU conditions. The remainder of the TLAAs listed above is addressed in other LR section and in various WCAP reports as follows:

- The loss of fracture toughness due to neutron irradiation embrittlement is addressed in LR section 2.1.1, *Reactor Vessel Material Surveillance Program*

- Stress corrosion cracking (SCC) of RV components is addressed in LR section 2.1.5, *Reactor Coolant Pressure Boundary Materials*

- Under clad cracking is addressed in WCAP-15338, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants*, 3/21/2000

- Aging management for reactor coolant system supports is addressed in WCAP-14422, Rev. 2-A, *License Renewal Evaluation: Aging Management for Reactor Coolant System Supports*, July 1995.

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor vessel and vessel supports. The aging evaluations approved by the NRC in NUREG-1786 for the reactor vessel and supports remain valid for EPU conditions.

In addition, the evaluations (summarized in this section) of maximum stress intensity ranges and cumulative fatigue usage factors for the components of the reactor vessel, considering EPU conditions, show that the reactor vessel components continue to meet the ASME acceptable limits. Since the original 40-year design transient set has been shown to be bounding for 60 years of operation based on the finding that the number of original design cycles bounds the actual plant cycles, and the number of design cycles for the EPU has not changed from the original 40-year transient set, the fatigue evaluations of the reactor vessel components are valid for 60 years of operation.

Finally, for the reactor vessel components determined to be potentially impacted by environmental fatigue, the environmental effects on fatigue were evaluated based on the

updated fatigue usage factors determined from the EPU evaluations. The cumulative fatigue usage factors for the inlet nozzle, outlet nozzle, and bottom-head-to-shell juncture, with environmentally-assisted fatigue factors applied, are still below the ASME code limit of 1.0. Environmental effects on fatigue is generally discussed in LR section 12.14.3, Impact of EPU on Environmentally Assisted Fatigue Evaluations.

Reactor Vessel and Vessel Supports Results

Summary of Results

Based upon the reactor vessel evaluations outlined in this report, all of the maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors for the following Ginna Station reactor vessel components continue to satisfy the applicable limits of ASME B&PV Code, Section III, Division 1, 1995 Edition through 1996 Addenda:

- Closure head
- CRDM housings
- Vent nozzles
- Main closure region (head and vessel flanges and closure studs)

In addition, all of the maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors for the balance of the Ginna Station reactor vessel components listed below continue to satisfy the applicable limits of the 1965 Edition of the ASME B&PV Code, Section III, Division 1:

- Outlet nozzle and support pad
- Inlet nozzle and support pad
- Safety injection nozzle
- Vessel wall transition
- Bottom head-to-shell juncture
- Bottom head instrumentation tubes
- Core support guides
- External support brackets

The reactor vessel/internals interface loads are also below the allowable limits.

The environmental effects on fatigue were evaluated and found to be below the ASME Code limit under EPU conditions.

Discussion of Results

The maximum ranges of stress intensity and maximum cumulative fatigue usage factors from the reactor vessel evaluation are shown in Table 2.2.2.3-1. The regions that had increases in

the maximum ranges of stress intensity are the CRDM nozzle, the vent nozzle, the outlet nozzle, the safety injection nozzle, the bottom head instrumentation tubes, the core support guides, and the external support brackets. Maximum stress intensity ranges were not calculated for the inlet nozzle and the vessel wall transition as part of the original stress report set. All regions except the outlet nozzle, the safety injection nozzle, the inlet nozzle, and the vessel wall transition had increases in the maximum fatigue usage factors. Usage factors were not calculated for the inlet nozzle and the vessel wall transition in the original stress reports. The seismic and LOCA loads are shown in Table 2.2.2.3-2. All of the loads due to the EPU are less than the allowable or limiting loads. The vessel support load comparisons are shown in Tables 2.2.2.3-3 and 2.2.2.3-4. All of the loads due to the EPU and Snubber Reduction Programs are less than the limiting loads as determined by the Gilbert Associates, Inc. report entitled "Ginna Reactor Vessel Supports".

Table 2.2.2.3-1
Maximum Range of Stress Intensity and Cumulative Fatigue Usage Factors

Location	Maximum Range of Stress Intensity (ksi)			Cumulative Fatigue Usage Factor		
	Pre-EPU	Post-EPU	Limiting	Pre-EPU	Post-EPU	Limiting
Closure Head at Flange	45.8	45.8	45.8 < $3S_m$ = 80.1 ksi	0.302	0.338	0.338 < 1.0
Vessel at Flange	51.8	51.8	51.8 < $3S_m$ = 80.1 ks	0.147	0.223	0.223 < 1.0
Closure Studs	94.8	94.8	94.8 < $3S_m$ = 103.4 ksi	0.810	0.963	0.963 < 1.0
CRDM Nozzle	72.2	79.0	79.0 > $3S_m$ = 69.9 ksi(1)	0.323	0.580	0.580 < 1.0
CRDM Nozzle J-weld	59.0	63.8	63.8 < $3S_m$ = 69.9 ksi	0.416	0.742	0.742 < 1.0
Vent Nozzle	29.0	31.9	31.9 < $3S_m$ = 41.1 ksi	0.002	0.009	0.009 < 1.0
Vent Nozzle J-weld	37.7	41.9	41.9 < $3S_m$ = 69.9 ksi	0.114	0.494	0.494 < 1.0
Outlet Nozzle						
Safe End	not reported(2)	39.9	39.9 < $3S_m$ = 49.2 ksi	not reported	n/a(3)	n/a
Nozzle	35.8	49.2	49.2 < $3S_m$ = 80.1 ksi	0.155	0.0440	0.155 < 1.0
Support Pad	n/a	n/a	n/a	not reported	0.3862	0.386 < 1.0
Inlet Nozzle						
Safe End	not reported	35.8	35.8 < $3S_m$ = 49.2 ksi	not reported	n/a	n/a
Nozzle	not reported	38.8	38.8 < $3S_m$ = 80.1 ksi	not reported	0.0329	0.033 < 1.0
Support Pad	n/a	n/a	n/a	not reported	0.0607	0.061 < 1.0
Safety Injection Nozzles	50.9	55.4	55.4 < $3S_m$ = 80.1 ksi	0.470	0.219	0.470 < 1.0
Vessel Wall Transition	not reported	32.2	32.2 < $3S_m$ = 80.1 ksi	not reported	0.0026	0.003 < 1.0
Bottom Head to Shell Juncture	30.8	28.6	28.6 < $3S_m$ = 80.1 ksi	0.000	0.0024	0.002 < 1.0
Bottom Head Instrumentation Nozzle	10.9	37.1	37.1 < $3S_m$ = 69.9 ksi	0.000	0.228	0.228 < 1.0

Table 2.2.2.3-1 (cont.)
Maximum Range of Stress Intensity and Cumulative Fatigue Usage Factors

Location	Maximum Range of Stress Intensity (ksi)			Cumulative Fatigue Usage Factor		
	Pre-EPU	Post-EPU	Limiting	Pre-EPU	Post-EPU	Limiting
Core Support Guides	22.1	52.5	$52.5 < 3S_m = 69.9$ ksi	0.000	0.132	0.132 < 1.0
External Support Brackets	24.9	41.2	41.2 < 3Sm = 80.1 ksi	0.020	0.979	0.979 < 1.0

Notes:
1. A simplified elastic-plastic analysis per section NB 3228.5 of the ASME Code was performed for this location and showed that it fulfills all the requirements.
2. "Not reported" indicates that values were not reported in the Ginna UFSAR, Revision 18.
3. Cumulative usage factors were not reported for the safe ends of the outlet and inlet nozzles in Table 2.2.2.3-1 because the nozzle-to-shell junction, not the safe end, was found to be the worst fatigue location.

Table 2.2.2.3-2
Reactor Vessel/Internals Interface Loads (lbs)

Location	EPU Interface Load	Allowable Load
Barrel Outlet Nozzle	22,204	75,000
Lower Radial Keys (Core Support Guides)	1,270,000	3,470,000

Table 2.2.2.3-3

Reactor Vessel Supports Faulted Loads Comparison (kips)

Load Direction	Gilbert Associates Support Load Cases			Snubber Reduction Program Support Loads (1)	EPU Support Loads (1)
	Case 1	Case 2	Case 3		
Horizontal	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c
Vertical	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c

Notes:

1. The support loads from the Snubber Reduction and EPU Programs are considered acceptable because they are less than at least one of the Load Cases 1, 2, or 3.

Table 2.2.2.3-4

Reactor Vessel Supports Normal/Operating Loads Comparison (kips)

Load Direction	Gilbert Associates Support Load Case	Snubber Reduction Program Support Loads	EPU Support Loads
Horizontal	[ ]a,c	[ ]a,c	[ ]a,c
Vertical	[ ]a,c	[ ]a,c	[ ]a,c

Notes:

[1.

]a,c

Reactor Vessel and Supports References

1. WCAP-16411, Rev. 0, Addendum to Babcock & Wilcox Company Original Analytical Reports (performed under Contract No. 610-0110) and Babcock & Wilcox Canada Report No. BWC-083N-SR-01, Rev. 1, Design and Service Loading Condition Report for Ginna Station Nuclear Power Plant Reactor Vessel.

Reactor Vessel and Supports Conclusions

- Ginna has reviewed the evaluations related to the structural integrity of the reactor vessel and vessel supports and concludes that the evaluations have adequately addressed the effects of the proposed EPU on the reactor vessel and vessel supports. Ginna further concludes that the evaluations have demonstrated that the reactor vessel and vessel supports continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR50.55a, GDC-1, GDC-2, GDC-4, GDC-14 and GDC-15, and the ASME Code, Section III, Division 1, following implementation of the proposed EPU. Flow-induced vibration was considered for the reactor vessel and is not a concern. Therefore, Ginna finds the proposed EPU acceptable with respect to the design of the reactor vessel and vessel supports.

- Since the evaluation of the reactor vessel and its supports considered the most severe conditions of all Ginna original, present (if different than original), and future operating bases (i.e., original design basis, replacement steam generator (RSG) program, and extended power uprate (EPU) program), and the 40-year design transient sets have been shown to be bounding for 60 years of operation, the fatigue evaluations performed for the EPU program demonstrate that the current design is acceptable to support EPU conditions for 60 years of plant operation.

- Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor vessel and vessel supports. The aging evaluations approved by the NRC in NUREG-1786 for the reactor vessel and supports remain valid for EPU conditions.

- Finally, the environmental effects on fatigue were evaluated for the reactor vessel components based on the updated fatigue usage factors determined from the EPU evaluations. The cumulative fatigue usage factors for the inlet nozzle, outlet nozzle, and bottom-head-to-shell juncture, with environmentally-assisted fatigue factors applied, are still below the ASME code limit of 1.0.

2.2.2.4 Control Rod Drive Mechanism

Introduction

The evaluation of the Ginna Station control rod drive mechanisms (CRDM) is an assessment of the impact on the structural integrity of the assemblies from the thermal transients and maximum operating temperatures and pressures that result from the proposed EPU operating conditions. The results of the structural analysis of the CRDM pressure boundary show that the analyzed stresses do not exceed the stress allowable of the ASME Code, and that the cumulative fatigue usage factors from the code fatigue analysis remain less than 1.0.

The CRDMs are mounted onto the reactor vessel (RV) head by means of head adaptors welded to RV head penetrations. The CRDM consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack. Main coolant fills the pressure containing parts of the drive mechanism. Thus, the pressure vessel component of the CRDM assembly constitutes a portion of the reactor coolant pressure boundary. The pressure boundary of the CRDMs and all the components of the control rod drive system are designed as Seismic Category I equipment.

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the full-length model L-106 control rod drive mechanisms (CRDMs). The CRDMs were evaluated using the NSSS operating parameters of LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1, and the NSSS design transients of LR section 2.2.6, NSSS Design Transients developed for the Ginna Station EPU program. The original Westinghouse model L-106 CRDMs supplied to the Ginna Station were replaced during the 2003 refueling outage with equivalent units (model L-106A) as part of the RV head replacement program (work order number PCR 2001-0042). The L-106 and L-106A models have been compared and were found equivalent for the purposes of evaluation for EPU conditions. Therefore the following discussions that address the L-106 Model are applicable to the new L-106A Model currently installed.

Control Rod Drive Mechanisms are described in UFSAR sections 3.1.1.4.1, Protection System Reliability; 3.1.1.5.6, Maximum Reactivity Worth of Control Rods; 3.7.3.1.1.3, (Seismic Analysis Methods) Control Rod Drive Mechanisms; 3.9.4., Control Rod Drive System; 4.2.1.3.4.4, (Mechanical Limits) Control Rod Drive Assembly; 4.5.1, Control Rod Drive System Structural Materials; 5.3.1.1, Reactor Vessel Description; 7.2.1.1.2, Reliability and Testing; and 9.4.1.2.3, Control Rod Drive Mechanism Cooling System.

As described in the UFSAR, each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant water. All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels for Class 1 Vessel appurtenances. Control rod drive assemblies are mounted on the top of the reactor vessel and are considered an extension of the reactor vessel head. All parts exposed to reactor coolant, such as the pressure vessel, latch assembly, and drive rod, are made of metals which resist the corrosive action of the water. Three types of metals are used exclusively: stainless steels, Inconel X-750, and cobalt-based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, stainless steel is used. Cobalt-based alloys are used for the pins and latch tips. Inconel X-750 is used for the springs of both latch assemblies and type 304 stainless steel is used for all pressure containment.

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to control rod drive mechanisms with the following amplifications:

- The adequacy of the Ginna design of CRDMs relative to the general design criteria is discussed in Ginna UFSAR section 3.1.1.4.1, 3.7.3.1.1.3, 3.9.2.2.4.11 and 3.9.4. Regarding GDC-1, discussions of applicable codes and standards, quality assurance programs, test provisions, etc., for the control rod drive system components are given in USFAR sections 3, 4 and 7 wherein the CRDM system is described.

As part of the Systematic Evaluation Program (SEP) the NRC evaluated the stresses in reactor coolant system components under normal and accident conditions. In the NRC Safety Evaluation Report (NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983) it was concluded that the control rod drive mechanisms were acceptably designed, with the stress analysis results within established limits.

The CRDM components evaluated herein for the proposed EPU are discussed in NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant, May 2004 in sections 3.0 and 3.1, primarily as they relate to the industry operating experience of incidents of primary pressure boundary degradation due to primary water stress corrosion cracking (PWSCC). Regarding the programs used to manage the aging effects associated with these components (AMPs), a discussion of Ginna RV head replacement program is provided in LR section 2.1.5 Reactor Coolant Pressure Boundary Materials. The head replacement project included replacing CRDMs with RV head penetrations fabricated with materials that provide enhanced resistance to PWSCC.

Analysis of flow induced vibration with respect to CRDMs is not included in the licensing basis for Ginna. However, it was considered for more susceptible components that would experience a significant flow increase under EPU conditions. CRDMs were evaluated and deemed unaffected by any increased flows due to EPU conditions and have not been analyzed for FIV.

Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The model L-106 CRDMs were originally designed and analyzed to meet the ASME Code. Those generic analyses for model L-106 CRDMs were the basis for this evaluation. As explained above, the NSSS operating parameters and NSSS design transients developed for the Ginna EPU were used as the new inputs for this evaluation. The seismic loading has not been changed for the Ginna EPU program.

The Ginna Station CRDMs are of the hot head type, defined by the vessel outlet reactor coolant temperature, T_{HOT}, in the NSSS operating parameters, Section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. Thus, the analysis assumes the NSSS design transients defined for the hot leg.

The acceptance criteria for the ASME Code structural analysis of the CRDM reactor coolant pressure boundary are that the analyzed stresses do not exceed the stress allowable of the ASME Code, and that the cumulative fatigue usage factors from the code fatigue analysis remain less than 1.0. For those cases for which changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease. The NSSS design transients developed for the Ginna EPU are shown to be bounded by the generic model L-106 CRDMs analysis, so the stresses and cumulative usage factors calculated for the CRDMs in the generic analysis of model L-106 CRDMs remain bounding and applicable for the Ginna Station EPU program.

Description of Analyses and Evaluations

Operating Pressure and Temperature

The NSSS temperatures and pressures developed for the Ginna EPU program as given in LR section 1.1, Nuclear Steam Supply System Paramaters, Table 1-1 were compared to those used for the generic model L-106 CRDMs design and analysis. There is no change in the reactor coolant pressure of 2250 psia for any of the EPU cases. The hot-leg temperature (T_{HOT}) defined by the vessel outlet temperature is a maximum of 611.8°F, which is less than the 650.0°F temperature used in the generic analysis for

model L-106 CRDMs. Since none of the temperatures exceed the previously analyzed temperature, and the pressure does not change, the NSSS parameters developed for the EPU program and used for this evaluation are bounded by the generic analyses model L-106 CRDMs.

Transient Discussion

The NSSS design transients, discussed in LR section 2.2.6, NSSS Design Transients, were not significantly changed from those used to analyze the generic model L-106 CRDMs . Thus, the generic transients analyzed for model L-106 CRDMs were shown to remain bounding and acceptable for the Ginna EPU program. There are no changes in the pressure transients.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of EPU on the conclusions reached in the Ginna License Renewal Application for the CRDMs. In regard to license renewal [NUREG-1786], Section 4.3.1 of the SER projects that the original 40 year design transient will remain bounding for 60 years of plant operation and in SER Section 4.3.2 the NRC Staff concluded the same is acceptable. Therefore, the 60 year life extension remains valid for the CRDMs including accounting for the effects of EPU. No specific environmentally assisted fatigue affects are addressed in the SER [NUREG-1786] concerning the Control Rod Drive Mechanisms, hence it is concluded that there is no impact with regard to environmentally assisted fatigue for the CRDMs.

CRDM Results

A summary of the results of the evaluations performed for the EPU is presented in Table 2.2.2.4-1. As noted above, the results of the model L-106 generic analysis remain bounding and are acceptable for the Ginna Station EPU operating conditions and the calculated stresses in various parts of CRDMs meet the allowable ASME stress limits. The cumulative usage factors that were calculated are also given in Table 2.2.2.4-1, and remain bounded for the EPU program. The highest cumulative usage factor was calculated [0.858] at the upper joint canopy. The usage factor was calculated in a conservative manner, where the applied transients were grouped and the allowable number of cycles considered for each group was based on the most severe transient in the group.

CRDM Conclusions

The Ginna staff has reviewed the evaluation related to the structural integrity of pressure-retaining components of the CRDM. For the reasons presented above, the Ginna staff concludes that the effects of the proposed EPU on these components have been adequately addressed. The Ginna staff further concludes that these pressure-retaining components will continue to meet the requirements of 10CFR50.55a, GDC-1, GDC-2, GDC-4, and GDC-14 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components.

2.2.2.4-1 Cumulative Fatigue Usage Factors for CRDM Joints, Applicable for Ginna Station EPU		
CRDM Joint and Component	**Cumulative Usage Factor**	
	Values Applicable for the EPU Program *	**Allowable Value**
Upper Joint Canopy	[ ]a,c	1.00
Upper Joint Canopy Weld	[ ]a,c	1.00
Upper Joint Threaded Area	[ ]a,c	1.00
Middle Joint Canopy Weld	[ ]a,c	1.00
Lower Joint Canopy Weld	[ ]a,c	1.00

* - The values from the generic analysis performed for model L-106 CRDM remain applicable and bounding for the Ginna EPU program.

2.2.2.5 Steam Generators and Supports

Introduction

The specification of Ginna Station replacement steam generators supplied by the Babcock & Wilcox Company (B&W) in 1996 anticipated power uprate and the steam generators were specified accordingly. The replacement steam generators have been evaluated for operation at the EPU conditions specified in LR section 1.1, Table 1-1, and contained in the steam generator certified design specification (B&W Specification No. TS-3270, Rev. 1, *Constellation Energy R.E. Ginna Station Certified Design Specification for Replacement Steam Generator*). The steam generators are described in the *Ginna Updated Final Safety Analysis Report* (UFSAR) section 5.4.2. Evaluation of the steam generators has demonstrated continued compliance with applicable regulatory and industry structural integrity and thermal-hydraulic performance requirements following the implementation of EPU. The steam generators were also evaluated to demonstrate that failure due to tube vibration and wear does not occur. The evaluations considered an EPU full-power core thermal power level of 1811 MWt (nuclear steam supply system [NSSS] power level of 1817 MWt), steam generator tube plugging (SGTP) over the range from 0 to 10%, with a primary average temperature (T_{avg}) window from 564.5° to 576°F, a full-load steam generator outlet pressure ranging from 700 to 781 psia, and a feedwater temperature window from 435° to 390°F. Postulated steam generator design transients that are affected by the EPU were addressed in the evaluations.

The steam generator and supports evaluation was performed in four separate, but coordinated, portions:

- Supports
- Structural Integrity
- Thermal-Hydraulic
- Tube Vibration and Wear

The following separate sub-subsections within this Licensing Report subsection describe each analysis; its input parameters, assumptions, and acceptance criteria; the impact of EPU on each topic within the License renewal application/approval; and provide results and a conclusion regarding the topic. A collective finding regarding the adequacy of the steam generators and supports under EPU conditions concludes this LR subsection.

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to steam generators and supports, with the following amplifications.

The steam generators were designed, fabricated, tested, and inspected in accordance with the Replacement Steam Generator Certified Design Specification (BWNT Doc. 18-1224785-05, Rev. 5, Design Specification for Replacement Steam Generator for Rochester Gas and Electric Corporation Ginna Station Unit 1). Both the primary and secondary side pressure boundary and integral attachments were designed in accordance with the ASME B&PV Section III, Division 1, Class 1 (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III, Subsection NB, NF and Appendix F, 1974 and 1986 Editions

This satisfied NUREG-0800 Standard Review Plan, Section 3.9.3, which provides guidance for the NRC staff's review of ASME Class 1 components and component supports to ensure they are in compliance with the applicable portions of 10CFR50.55a and General Design Criteria (GDC) -1, -2, -4, -14, and -15 in 10CFR50, Appendix A as described in the generic CLB (LR section 2.2.2).

Except for the U-tubes, the steam generator internal components are not governed by the ASME B&PV Code, however, the ASME B&PV Code Class 1 requirements were used as a guide for their design. The internal components were required to withstand all specified loadings to maintain heat transfer capabilities during and following a design basis earthquake (DBE). In addition, it was shown that the U-tubes do not deform as a result of a DBE.

The steam generators were also designed to maintain tube integrity following an instantaneous full rupture of the steam line downstream of the steam outlet nozzle during full-power operation. This evaluation was in accordance with criteria from the ASME B&PV Code, Section III for Level D conditions. Tube integrity was also demonstrated following a small steamline break in accordance with criteria from the ASME B&PV Code, Section III for Level C conditions.

An analysis of the steam generators was performed to ensure that the U-tubes are adequately supported to avoid significant levels of tube vibration causing wear and fatigue. The analysis was performed to Ginna Station specification requirements and industry accepted criteria.

By meeting the acceptance criteria discussed in the various sections below, it is concluded that the portion of the RCPB formed by the steam generators and tubing will continue to meet all the specified criteria at the EPU conditions.

In addition to the evaluations described in the Introduction section above, the steam generators and supports were evaluated for plant license renewal. The results of the NRC review of the steam generators are documented in section 3.1.2 of NUREG-1786, *License Renewal Safety Evaluation Report (SER) for the R. E. Ginna Nuclear Plant*, May 2004. Time Limited Aging Analysis (TLAA) and the programs used to manage aging

were evaluated in those sections. The Steam Generator Integrity Program is discussed in SER sections 3.1.2.3 and 3.1.2.4.5.

2.2.2.5.1 Steam Generator Supports

Technical Evaluation

Introduction

The primary equipment steam generator supports of the nuclear steam supply system (NSSS) as described in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in sections 3.7.3.1.1.2, 3.9.2.2.4.8 and 3.9.3 are also evaluated for the EPU program. The reactor coolant loop (RCL) piping loads on the primary equipment steam generator supports due to the parameters associated with the EPU as discussed in LR section 2.2.2.1, NSSS Piping, Component, and Supports, were reviewed for impact on the existing RCL primary equipment steam generator supports design-basis in Reference 1. The RCL piping loads on the steam generator supports due to deadweight, thermal expansion, operational basis earthquake (OBE), and safe shutdown earthquake (SSE) loading cases are obtained from the evaluation for the EPU program as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports. The loss-of-coolant (LOCA) and the pipe break analyses from the current design basis in Reference 1 remain valid for the EPU program.

The steam generator supports stress margin values are evaluated for the EPU program based on the stress margin data from the current design basis in Reference 1 and the steam generator support loads obtained from the evaluation for the EPU program from the RCL piping system analyses as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports.

Input Parameters, Assumptions, and Acceptance Criteria

The RCL piping loads on the steam generator supports due to deadweight, thermal expansion, seismic OBE, and seismic SSE loading cases are obtained from the piping system analyses for the EPU program as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports. The LOCA and the pipe break analyses from the current design basis in Reference 1 remain valid for the EPU program.

The acceptance criteria for the Ginna Station RCL piping primary equipment steam generator supports indicated in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in section 3.9, are based upon the *ASME Boiler and Pressure Vessel Code* (B&PV Code), Section III, Subsection NF and Appendix F, 1974 Edition (Reference 2) and as used in the current design basis in Reference 1.

Description of Analyses and Evaluations

The steam generator support loads from the RCL piping system analyses as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports, and the current design basis steam generator support loads and stress margins from Reference 1 are used to calculate the stress margins available for the EPU program for the steam generator supports. The current design basis for the RCL piping system analysis in Reference1 was updated to consider the replacement steam generators and EPU PCWG design parameters. The steam generator upper support bumpers, steam generator upper support snubbers, steam generator lower lateral supports, and the steam generator lower support columns are evaluated for the stress margin values for the EPU program. The stress margin values are summarized and are tabulated in Table 2.2.2.5.1-1 for the loading combinations as specified in the acceptance criteria in the Code of Record in Reference 2, in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in section 3.9, and as evaluated in the current design basis in Reference 1.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the steam generator supports. The aging evaluations approved by the NRC in Section 3.1.2 of the License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, NUREG-1786, for the steam generator supports remain valid for the EPU conditions. As approved by the NRC in Section 3.1.2 of the SER, and as in *WCAP-14422 "License Renewal Evaluation: Aging Management Evaluation for Reactor Coolant System Supports" Revision 2-A, dated December 2000,* the evaluations for the aging management performed for the steam generator supports remain valid for the EPU conditions.

Steam Generator Supports Results

The stress margins available for the EPU program for the steam generator upper support bumpers, steam generator upper support snubbers, steam generator lower lateral supports, and the steam generator lower support columns are evaluated and summarized in Table 2.2.2.5.1-1 for the EPU program. In all cases, the calculated stress is less than the allowable and the stress margin values in Table 2.2.2.5.1-1 for the different loading combinations for the EPU program are either equal to or less than the stress margin values in the current design basis in Reference 1.

Steam Generator Supports References

1. Westinghouse Letter: SE&PT-CSE-3041, *Final Report for the Robert Emmett Ginna Nuclear Generating Station Steam Generator Hydraulic Snubber*

Replacement Program, October 1, 1992 (Final Report attached to letter, *Evaluation Of The Reactor Coolant System For The Steam Generator Hydraulic Snubber Replacement Program*, September 1992.).

2. *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code* Section III, Subsection NF and Appendix F, American Society of Mechanical Engineers 1974 Edition.

Table 2.2.2.5.1-1
RCL Primary Equipment Support Loads Stress Margin Summary
(Stress Margin = Allowable/Actual) (Based on $K_{average}$) (See Note 3)

Service Level	Normal	Upset	Emergency	SSE	Faulted (See Note 1)
Load Combination	Deadweight + Thermal Normal	Deadweight + Thermal (Normal + Overtemperature) + OBE	Deadweight + Thermal Normal + DBPB	Deadweight + Thermal Normal + SSE	Deadweight + Thermal Normal + $[(SSE)^2 + (PIBK^2)]^{1/2}$
Steam Generator Upper Supports Bumpers	(See Note 2)	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c
Steam Generator Upper Supports Snubbers	(See Note 2)	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c
Steam Generator Lower Supports Lateral	(See Note 2)	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c
Steam Generator Lower Supports Columns	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c

Notes
1. PIBK (pipe break) includes DBPB (design basis pipe break).
2. Under normal conditions no significant loads are imposed on theses lateral support elements.
3. $K_{average}$ is the average of the steam generator upper support stiffness.

Steam Generator Supports Conclusions

Since stress allowables are not exceeded, all steam generator supports are adequate and acceptable for the EPU program.

2.2.2.5.2 Steam Generator Structural Integrity

Technical Evaluation

Introduction

The structural integrity evaluation of the steam generator was performed for an EPU core power level of 1811 MWt (nuclear steam supply system [NSSS] power level of 1817 MWt), steam generator tube plugging (SGTP) over the range from 0 to 10%, with a primary average temperature (T_{avg}) window from 564.5° to 576°F, a full load steam generator outlet pressure ranging from 700 to 781 psia, and a feedwater temperature window from 435° to 390°F. The stresses, stress intensity ranges, and fatigue usage factors in the steam generator for the EPU range of conditions were determined by reconciling the original design basis analyses against the new EPU conditions provided in the certified design specification (Reference 1).

To quantify the range of stress occurring during each postulated transient for the EPU conditions, ratios were determined between the EPU and design basis pressure, and temperature variations and the design basis range of stresses were prorated by these ratios. Acceptance of the results for EPU conditions was based on demonstrating continued compliance with the structural criteria in the *American Society of Mechanical Engineers Boiler Pressure Vessel* (ASME B&PV) Code Section III, Subsection NB (Reference 3). These acceptance criteria are the same as those used for the original design basis analyses of the steam generators.

The internal components, which are not part of the pressure boundary, are not governed by the ASME B&PV Code (Reference 3). However, ASME Code, Section III, Subsections NB and NF were adopted as guidelines for performing the structural analysis of these components.

The scope of the reconciliation was the entire steam generator pressure boundary, internal and external pressure boundary attachments, and all internal components. Formal reconciliations were performed for the tubesheet, U-tubes, primary divider plate, primary head and external attachments, secondary shell and internal/external attachments, primary and secondary nozzles, primary and secondary manways, handholes, inspection ports, studs and covers on all bolted openings, lower shell internals, and steam drum internals.

It was determined that the steam generator pressure boundary and internals continue to remain in compliance with the structural criteria of the ASME Code, Section III, Subsection NB (Reference 3) and, therefore, are adequate for operation at the EPU conditions.

Input Parameters, Assumptions, and Acceptance Criteria

The structural evaluation was performed for an EPU core power level of 1811 MWt (NSSS power level of 1817 MWt), steam generator tube plugging (SGTP) over the range from 0 to 10%, with a primary average temperature (T_{avg}) window from 564.5° to 576°F, a full-load steam generator outlet pressure ranging from 700 to 781 psia, and a feedwater temperature window from 435° to 390°F. The design transients for the EPU conditions are provided in the certified design specification (Reference 1).

The stresses, stress intensity ranges, and fatigue usage factors in the steam generators for operation at the EPU conditions were calculated by reconciling the original design basis analyses. To quantify the change in the range of stress occurring during each postulated transient for the EPU relative to the design basis, an approximate method was used by determining the ratios between the pressure and temperature variations and prorating the range of stresses by these ratios. A detailed comparison was performed between the EPU and design basis transients to determine these temperature and pressure variation ratios. Variations were compared for the primary side inlet and outlet temperatures, secondary side temperatures, feedwater temperatures, primary side pressures, and secondary side pressures. The bounding ratios were used to prorate the stress results. Where the temperature histories were considered to be too dissimilar for a meaningful comparison, finite element analysis was used.

The replacement steam generators began operation in 1996 and are qualified for the equivalent of 40 years of the design basis transients. Operation at the EPU conditions will begin no earlier than 2006. For simplicity, all components, except for the studs on the pressure boundary bolted openings, were considered to operate at the EPU conditions for the full 40-year design life. Where this assumption was overly conservative, the original design basis transients were considered for 9 years of operation and the EPU transients for the remaining 31 years. Reductions in the fatigue life due to the EPU were considered for the pressure boundary studs.

No revisions were specified to the external nozzle and attachment loads in the EPU certified design specification (Reference 1). The original design basis loads were used in the reconciliation analyses. No revisions were specified in the certified design specification (Reference 1) for the burst pipe loads for operation at the EPU conditions. The original design basis loads were used in the reconciliation analyses. The seismic loading of the internals was unaffected by the EPU and the original design basis loads

remain bounding. The bolt preloads for bolted pressure boundary openings were established on the basis of "leak proofing" the joints. The original design basis preloads were used in the reconciliation analyses.

The design basis analysis demonstrating protection against non-ductile fracture is unaffected by the EPU since the lower temperature operation steam pressure remained unchanged. Only pressures at less limiting elevated temperatures were changed. As a result, the current reconciliation of the design basis analysis remains bounding for the EPU condition.

Continued compliance with the current steam generator design basis analysis is the acceptance criteria for the structural analysis for EPU conditions. For the structural evaluation of the pressure boundary components, the acceptance criteria from ASME, Section III, Subsection NB for Class 1 (Reference 3) components continued to remain applicable. Excessive plastic deformation is prevented by limits on the acceptable primary stresses. Plastic instability and incremental collapse are prevented by limits on the acceptable primary-plus-secondary stresses. High-strain, low-cycle fatigue is prevented by limits on the total stresses and their cycles. Satisfaction of these limits demonstrates continued compliance with the current design acceptance criteria and, therefore, the adequacy of the steam generator design for operation at the EPU conditions for the remainder of the 40-year design life.

The steam generator internal components, other than the U-tubes, are not part of the pressure boundary and, therefore, are not governed by the ASME Code. However, ASME Code Section III Subsections NB and NF (Reference 3) were adopted as guidelines for performing the structural analysis of these components. These components were reviewed and it was determined that they satisfy the ASME Code requirements for components not requiring an analysis for cyclic operation. As a result, a fatigue analysis was not performed for the internals. The feedwater header was analyzed for fatigue since it is the most highly loaded of all the internals due to rapid feedwater flow and temperature changes.

Description of Analyses and Evaluations

From a structural standpoint, the increased pressure and temperature variations specified in the certified design specification (Reference 1) during EPU normal- and upset-operating conditions impact the steam generator. Both the primary and secondary side steam generator components are affected to differing degrees by these increased pressure and temperature variations, resulting in increased stress intensity ranges and fatigue usage factors. The steam generator ASME code design conditions are unaffected by the EPU conditions except for those components affected by primary-to-secondary-side pressure differentials. The steam generator differential design pressure

for the tube sheet was increased from 1550 to 1575 psi as a result of the EPU steam pressure operating range. This was necessary due to a review of EPU Level "A" transient pressures which showed that during several transients, the pressure differentials between primary and secondary sides exceeded the previously qualified value of 1550 psi.

The steam generator EPU structural evaluation was performed by reconciling the existing steam generator design basis analyses relative to the conditions specified in the certified design specification (Reference 1). This reconciliation was performed by B&W in Report BWC-143O-SR-01 (Reference 2). The scope of the reconciliation included all of the steam generator pressure boundary and the internal components. Formal reconciliations were performed for the tubesheet, U-tubes, primary divider plate, primary head and external attachments, secondary shell and internal/external attachments, primary and secondary nozzles, primary and secondary manways, handholes, inspection ports, studs and covers on all bolted openings, lower shell internals, and steam drum internals.

Both high and low T_{avg} operation were considered in the reconciliation. High T_{avg} operation corresponded to primary-and-secondary-side operating temperatures of 611.8° and 515.4°F, respectively. Low T_{avg} operation corresponded to primary-and-secondary-side operating temperatures of 601.0° and 503.1°F, respectively. The more bounding of the conditions was considered, and depended on the location of the component.

The reconciliation analyses presented in Reference 2 used both classical and finite element methods to determine the stresses, stress intensity ranges, and fatigue usage factors for the EPU conditions. Classical methods included the use of prorating factors to multiply the results from the original design basis analyses to approximate the corresponding stresses for EPU conditions. To quantify the increase in the range of stress occurring during each transient for EPU conditions with respect to the corresponding design basis transient, an approximate method was to determine the ratios between the pressure and temperature variations, and to prorate the design basis range of stresses by these ratios.

prorating factor = $\Delta T'/\Delta T$ or $\Delta P'/\Delta P$

where: $\Delta T'$ is the temperature variation during the EPU transient
ΔT is the temperature variation during the design basis transient
$\Delta P'$ is the pressure (or pressure difference) variation during the EPU transient
ΔP is the pressure (or pressure difference) variation during the design basis transient

Variations were compared for the primary side inlet and outlet temperatures, secondary side temperatures, feedwater temperatures, primary side pressures, secondary side pressures, and primary-to-secondary-side pressure differentials. The bounding ratios were used as appropriate. Factors less than 1.0 were ignored since they are bounded by the existing analyses.

Where the temperature histories were considered too dissimilar for a meaningful comparison, finite element analysis was used to either requalify a component or perform a benchmark study. Benchmark studies were performed to compare the design basis and EPU transients from a stress intensity range and fatigue standpoint. The results from these benchmark studies were used to quantify the increase in the severity of the EPU condition based on transients for other components with a similar geometry.

Where finite element analysis was used, the calculated heat transfer coefficients and fluid temperature ramps for the Levels A and B transients were used as input to the finite element program to determine the metal temperature variation with time. The thermal analysis results were reviewed to determine the appropriate times during the transients for subsequent analysis of total and linearized stress levels. The critical times were chosen by monitoring component-through-thickness-temperature differences, between components temperature differences, and skin temperature differences. The time-temperature histories of these differences were plotted, and the times at which they reached extremes were tabulated. These times formed the basis upon which subsequent analysis of total and linearized stress levels were performed. The temperature distribution and the corresponding pressure at each critical time were used to determine the pressure-plus-thermal loading stresses in static analyses.

Linearized and total stress intensities were obtained through various sections of the models. A post-processing program was used to calculate the linearized stress intensity range and fatigue usage factor at each selected location.

All components, except for the pressure boundary bolted opening studs, were considered to operate at the EPU conditions for the original 40-year design life. If this assumption was too conservative, the current operating condition based design basis transients were considered for 9 years of operation, and the EPU condition based transients were considered for the remaining 31 years. A reduction in the fatigue life associated with the EPU conditions was necessary only for the studs.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

As an integral part of the License Renewal Program the Ginna Station steam generator integrity program manages aging effects such as cracking due to primary water stress-

corrosion cracking (PWSCC), outside diameter stress-corrosion cracking (ODSCC), IGA, pitting, wastage, wear fouling due to corrosion product buildup, mechanical degradation due to denting and impingement damage, and fatigue. The steam generator integrity program manages these aging effects/mechanisms through a balance of prevention, inspection, examination, assessment, evaluation, repair, and leakage monitoring measures. The program is administered through a series of plant directives and interface procedures, as well as the plant Technical Specifications. Key program attributes include nondestructive examination (NDE), sludge lancing, primary and secondary water chemistry control, and primary-to-secondary leakage trending and monitoring.

The current Ginna Station steam generators installed in 1996 incorporated design and manufacturing improvements to reduce and/or prevent many of the problems that the industry experienced with the original design. The structural integrity evaluation of the steam generators under the License Renewal program did not require time-limiting aging analysis evaluation for fatigue because an analysis was performed according to the requirements of the ASME Section III, Subsection NB, for a 40-year design life. As a result, the steam generators were concluded to require no fatigue monitoring.

The structural integrity evaluation for the EPU operating conditions demonstrated that, with the exception of the studs on the pressure boundary bolted openings, the steam generators continue to comply with the ASME fatigue requirements for a 40-year life. Therefore, except for the studs, no fatigue monitoring is required for operation at the EPU conditions.

Changes in the water film velocities and operating temperatures for the EPU operating conditions may affect the flow-assisted corrosion rates. However, due to the uncertainty in the calculation of these rates and the large scatter in the experimental data, the corrosion rates used in the original design basis analysis were used.

Steam Generator Structural Integrity Results

The most critical results from the structural evaluation are presented in Table 2.2.2.5.2-1. The internals are largely unaffected by the EPU and are not included in Table 2.2.2.5.2-1.

Steam Generator Structural Integrity References

1. B&W Specification No. TS-3270, Rev. 1, *Constellation Energy R.E. Ginna Station Certified Design Specification for Replacement Steam Generator.*

2. B&W Report, BWC-143O-SR-01, Rev. 1, *Constellation Energy R.E. Ginna Station Replacement Steam Generators - Qualification Report for Power Uprate Operation with Core Power of 1811 MWt.*

3. *ASME Boiler and Pressure Vessel Code*, Section III, Subsection NB and NF and Appendices, (no addenda), American Society of Mechanical Engineers, New York, 1986.

Table 2.2.2.5.2-1
Ginna Station EPU Evaluation Summary
Critical Locations of Primary and Secondary Side Pressure Boundary Components

Component	Load Condition	Stress Category	Pre-EPU Stress (ksi)	EPU Stress (ksi)	Allow. Stress	Comments
			Fatigue	Fatigue[a]	Allow. Fatigue	
Primary Side Components						
Primary-Nozzle-to-Shell Juncture	Normal/Upset	Pm + Pb + Q Fatigue	80.6[b] 0.94	80.6[b] 0.93	80.1 1.0	
Tubesheet-Blowdown-and-Shell-Drain Holes	Normal/Upset	Pm + Pb + Q Fatigue	29.8 0.90	37.3 0.99	80.1 1.0	A conservative Stress Concentration Factor (SCF) of 4 was used in fatigue. A more accurate SCF will produce a lower fatigue usage factor.
Tubesheet Center	Design	Pm Pm + Pb	4.2 32.8	4.3 33.3	26.7 40.0	
	Normal/Upset	Pm + Pb + Q Fatigue	23.1 0.31	23.7 0.79	80.1 1.0	
Tubesheet at Tube-Free Zone	Normal/Upset	Pm + Pb + Q Fatigue	23.0 0.28	23.6 0.74	80.1 1.0	
Tubesheet at Innermost Tube Hole Edge	Normal/Upset	Pm + Pb + Q Fatigue	80.0 0.996	80.0 0.36	87.6 1.0	Since one extreme of the range is a pressure test, the limit on Pm+Pb+Q is 2Sy.

Table 2.2.2.5.2-1 (cont.)
Ginna Station EPU Evaluation Summary
Critical Locations of Primary and Secondary Side Pressure Boundary Components

Component	Load Condition	Stress Category	Pre-EPU Stress (ksi)	EPU Stress (ksi)	Allow. Stress	Comments
			Fatigue	Fatigue[a]	Allow. Fatigue	
Tubesheet Ligament	Normal/Upset	Pm + Pb + Q Fatigue	76.3 0.10	76.3 0.30	80.1 1.0	
Tubesheet Gutter	Normal/Upset	Pm + Pb + Q Fatigue	80.3 0.30	80.3 0.82	87.6 1.0	Since one extreme of the range is a pressure test, the limit on Pm+Pb+Q is 2Sy.
	Normal/Upset	Pm + Pb + Q Fatigue	44.2 0.11	57.5 0.37	80.1 1.0	
Primary Head at Support Pads	Normal/Upset	Pm + Pb + Q Fatigue	80.1 0.38	80.1 0.998	80.1 1.0	
Primary Manway at Shell	Normal/Upset	Pm + Pb + Q Fatigue	76.8 0.12	77.0 0.18	80.1 1.0	
Primary Manway Studs	Normal/Upset	σ_{ave} σ_{max} Fatigue	54.1 72.8 0.84	54.1 72.8 0.97	54.9 74.1 1.0	The fatigue limit of 1.0 corresponds to a 14-year design life.
	Emergency	σ_{ave} σ_{max}	45.5 73.4	46.4 74.9	56.6 84.9	
Divider Plate	Normal/Upset	Pm + Pb + Q Fatigue	66.9 0.09	66.9 0.44	69.9 1.0	

Table 2.2.2.5.2-1 (cont.)
Ginna Station EPU Evaluation Summary
Critical Locations of Primary and Secondary Side Pressure Boundary Components

Component	Load Condition	Stress Category	Pre-EPU Stress (ksi)	EPU Stress (ksi)	Allow. Stress	Comments
			Fatigue	Fatigue[a]	Allow. Fatigue	
Secondary Side Components						
Cone-to-Lower-Shell Juncture	Normal/Upset	Pm + Pb + Q Fatigue	99.7[b] 0.20	128.4[b] 0.98	80.1 1.0	
Lower Shell	Normal/Upset	Pm + Pb + Q Fatigue	73.7 0.03	95.8[b] 0.13	80.1 1.0	
Lower-Shell-at-Ring Girder	Normal/Upset	Pm + Pb + Q Fatigue	92.3[b] 0.20	117.1[b] 0.98	80.1 1.0	
Steam Outlet Nozzle	Normal/Upset	Pm + Pb + Q Fatigue	77.3 exempt	86.4[b] 0.94	80.1 1.0	
Shell Attachments	Normal/Upset	Pm + Pb + Q Fatigue	56.0 --	77.9 0.30	80.1 1.0	
Secondary Manway at Shell	Normal/Upset	Pm + Pb + Q Fatigue	70.3 0.55	77.9 0.77	80.1 1.0	
Secondary Manway Studs	Normal/Upset	σ_{ave} σ_{max} Fatigue	40.7 82.4 0.67	42.3 85.7 0.98	62.8 94.2 1.0	The fatigue limit of 1.0 corresponds to a 31-year design life.
Recirculation Nozzle	Normal/Upset	Pm + Pb + Q Fatigue	77.2 --	74.8 0.67	80.1 1.0	
Feedwater Nozzle Transition	Normal/Upset	Pm + Pb + Q Fatigue	74.3[b] 0.24	74.3[b] 0.24	69.9 1.0	

Table 2.2.2.5.2-1 (cont.)
Ginna Station EPU Evaluation Summary
Critical Locations of Primary and Secondary Side Pressure Boundary Components

Component	Load Condition	Stress Category	Pre-EPU Stress (ksi)	EPU Stress (ksi)	Allow. Stress	Comments
			Fatigue	Fatigue[a]	Allow. Fatigue	
Feedwater Nozzle	Normal/Upset	Pm + Pb + Q Fatigue	71.5 0.53	71.5 0.9998	80.1 1.0	
Lower Shell Handholes	Normal/Upset	Pm + Pb + Q Fatigue	60.0 0.91	61.0 0.36	80.1 1.0	
Lower Shell Handhole Studs	Normal/Upset	σ_{ave} σ_{max} Fatigue	39.8 71.7 0.86	40.0 74.3 0.98	57.6 77.8 1.0	The fatigue limit of 1.0 corresponds to a 27-year design life.
Lower Shell at Tubesheet	Normal/Upset	Pm + Pb + Q Fatigue	85.3 0.52	85.3 0.97	90.2 1.0	Since one extreme of the range is a pressure test, the limit on Pm+Pb+Q is 2Sy.
Upper Shell Handholes	Normal/Upset	Pm + Pb + Q Fatigue	76.6 0.59	67.2 0.79	80.1 1.0	
Upper Shell Handhole Studs	Normal/Upset	σ_{ave} σ_{max} Fatigue	53.2 74.3 0.85	55.3 77.3 0.97	57.6 78.1 1.0	The fatigue limit of 1.0 corresponds to a 28-year design life.
Inspection Ports	Normal/Upset	Pm + Pb + Q Fatigue	147.5[b] 0.97	117.1[b] 0.76	80.1 1.0	
Inspection Port Studs	Normal/Upset	σave σmax Pm + Pb + Q Fatigue	39.9 68.4 84.0 0.36	40.1 68.9 84.5 0.45	57.6 77.8 86.4 1.0	

Table 2.2.2.5.2-1 (cont.)
Ginna Station EPU Evaluation Summary
Critical Locations of Primary and Secondary Side Pressure Boundary Components

Component	Load Condition	Stress Category	Pre-EPU Stress (ksi)	EPU Stress (ksi)	Allow. Stress	Comments
			Fatigue	Fatigue[a]	Allow. Fatigue	
¾", 1" and 3" Small Nozzles	Normal/Upset	Pm + Pb + Q Fatigue	44.5 0.58	51.5 0.93	53.4 1.0	
Earthquake Brackets	Normal/Upset	Pm + Pb + Q Fatigue	84.1[b] 0.66	85.3[b] 0.93	80.1 1.0	
Seal Skirt	Normal/Upset	Pm + Pb + Q Fatigue	95.0[b] 0.94	93.2[b] 0.62	80.1 1.0	
U-Tubes	Design	Pm Pm/PL + Pb	14.1 34.1	14.7 34.8	26.6 40.0	
	Normal/Upset	Pm + Pb + Q Fatigue	73.6 0.00	73.6 0.00	79.8 1.0	
Feedwater Distribution System	Normal/Upset	Pm + Pb + Q Fatigue	69.89 0.80	69.89 0.93	69.9 1.0	

Notes:

a. The stress and/or fatigue results for the EPU conditions might be lower than those for non-EPU due to the removal of conservative assumptions during the EPU analytical reconciliation.

b. Simplified elastic-plastic analysis was used and the usage factor includes the Ke factor. The Ke factor is a multiplier for the alternating stress in a fatigue analysis and it accounts for the increased fatigue damage due to plastic cycling when the linearized stress intensity range exceeds the allowable (see ASME B&PV Code, Section III, NB-3228.5 [Reference 3]).

Steam Generator Structural Integrity Conclusion

The results of the evaluation demonstrated that the steam generator pressure boundary and internal components continue to comply with the structural criteria of the ASME Code Section III Subsection NB and NF (Reference 3) for operation at the EPU conditions. The EPU associated changes in operating conditions will require the following:

- The primary manway studs are replaced after 14 years of equivalent design cycles of actual operation rather than 16 years for current non-EPU condition operation.
- The lower shell handhole studs are replaced after 27 years of equivalent design cycles of actual operation rather than 32 years for current non-EPU condition operation.
- The upper shell handhole studs are replaced after 28 years of equivalent design cycles of actual operation rather than 40 years for current non-EPU condition operation,
- The secondary manway studs are replaced after 31 years of equivalent design cycles of actual operation rather than 40 years for current non-EPU condition operation.

The reductions in the operating life of the studs are caused by increases in the temperature and pressure variations of the design transients. This produces higher alternating stresses and a greater rate of fatigue over time. The transients that produce the most fatigue damage are plant heatup/cooldown and plant loading/unloading. The fatigue analysis is conservative from an analytical standpoint and extensions to the operating life of the studs can be achieved either by monitoring the number of actual cycles for these transients, or by performing fatigue tests in accordance with the ASME B&PV Code.

Table 2.2.2.5.2-1 lists the stresses, stress intensities (SI) and fatigue usage factors for both pre-EPU conditions and EPU conditions. For design conditions, only the primary-to-secondary-side pressure differential is affected by the EPU and it increases from 1550 psi to 1575 psi. The only components exposed to differentials in pressure between the primary and secondary sides are the tubesheet and U-tubes. The stresses in these components increase by 1.6% (i.e., 1575 psi/1550 psi) and these stresses are listed in Table 2.2.2.5.2-1.

The emergency and faulted conditions are unchanged for the EPU conditions except that the higher primary side temperature at full power results in a greater temperature range on the primary side during the loss-of-secondary-pressure transient (emergency). The ASME B&PV code only requires the evaluation of thermal stresses in studs for emergency conditions. For this reason, the primary manway studs are the only

components affected by the EPU emergency conditions. The stress in the studs increases by 2% and these stresses are listed in Table 2.2.2.5.2-1.

Relative to normal and upset conditions for the EPU, the SI ranges in the majority of the steam generator pressure boundary were found to remain unchanged since they were most often produced by the stress extremes occurring during the plant heatup/cooldown transient and the hydrostatic pressure test, neither of which is affected by the EPU. At some locations, where the more severe EPU conditions did produce a higher stress, slight increases in the SI range of approximately 5% to 10% occurred when this new maximum stress was combined with the previous maximum stress occurring during plant heatup/cooldown or the hydrostatic pressure test. At a few locations, larger increases in the SI range of 25% to 40% were found. These increases are conservative since they represent locations where prorating factors were liberally applied to the entire SI range even when this range was produced mainly by the plant heatup/cooldown or hydrostatic pressure test. The SI ranges are listed in Table 2.2.2.5.2-1.

Relative to normal and upset conditions for the EPU, the fatigue usage factors increase throughout the steam generator pressure boundary. Greater temperature and pressure variations result in greater alternating stresses and higher fatigue rates. Several components on the secondary side that were previously exempt from a fatigue evaluation no longer qualified for this exemption due to the increased pressure fluctuation occurring during the loss-of-load transient (479 psi versus 369 psi). Fatigue evaluations were performed for these components and the new usage factors were determined to be acceptable. For other components, usage factors were recalculated and, given that the allowable cycles on a fatigue curve are related to alternating stresses on a log basis, they increased by a larger factor than for the SI ranges discussed previously. Usage factors increased by between 10% and 400%.

Where these increases could not be accommodated, the original design basis methodology was altered to identify assumptions that, while simplifying the original analysis, were overly conservative for the analysis for EPU conditions. For this reason, some usage factors associated with the EPU conditions were lower than those previously calculated for the original design basis transients. For example, the tubesheet at the innermost tube hole edge previously had a usage factor of 0.996; a review of the analysis revealed that the support of the divider plate had not been included in the analysis. By accounting for this component, significant margin was created and the resulting usage factor for EPU conditions was 0.36.

Similar reductions were achieved elsewhere by correcting the amount of stress that was added to account for adjacent gross structural discontinuities that were omitted from the modeling (reflected stress). The full range of reflected stress occurring for all transients was often added to small stress amplitude fatigue passes, greatly increasing their alternating stress and fatigue usage factor. Conservative assumptions such as these

were removed until the usage factors for the EPU conditions were reduced to values below 1.0.

Fatigue usage factors close to 1.0 are indicative of more limiting areas within the steam generator pressure boundary. These include: primary-nozzles-at-shell juncture, tubesheet at shell blowdown and drain holes, primary head at support pads, cone-to-lower-shell juncture and at ring girder, steam outlet nozzle forging, feedwater nozzle forging, lower shell handhole forgings, lower-shell-to-tubesheet juncture, small nozzles, earthquake brackets, and the feedwater distribution system. Although all of these locations have usage factors greater than 0.90, these values are the result of simple, conservative analyses and could be reduced through further, more sophisticated analysis.

Changes in the water film velocities and operating temperatures for the EPU operating conditions may affect the flow-assisted corrosion rates. However, due to the uncertainty in the calculation of these rates and the large scatter in the experimental data, the corrosion rates used in the original design basis analysis were used. For the primary and secondary side pressure boundary, changes in the corrosion rates will be small since materials are clad and/or consist of materials resistant to corrosion. Internal components made from stainless steel are also resistant to corrosion while those made from carbon steel, such as the steam separators and U-bend support components, are more susceptible. This will be managed through regular inspection under the steam generator integrity program.

2.2.2.5.3 Steam Generator Thermal-Hydraulic Evaluation

Technical Evaluation

Introduction

Thermal-hydraulic analyses were performed for the Ginna station steam generators at the EPU conditions with a nuclear steam supply system (NSSS) power level of 1817 MWt. The analyses determined the steam generator thermal-hydraulic characteristics and inventories, and provided input used to evaluate the potential for tube wear and flow-induced vibration (FIV) failures. The results of this effort concluded that the steam generators will continue to have satisfactory thermal-hydraulic performance for the EPU operating conditions provided in the Steam Generator Certified Design Specification (Reference 1). Detailed discussions regarding the evaluations and the conclusions reached are provided below.

Input Parameters, Assumptions, and Acceptance Criteria

The operating conditions for the EPU were provided in the Steam Generator Certified Design Specification (Reference 1). The uprated NSSS power level of 1817 MWt, corresponding to a reactor core power of 1811 MWt, was used in the analyses. All of the significant thermal-hydraulic input parameters are listed below. Performance was determined for both start-up and end-of-life (EOL) conditions of the steam generator.

The following describes the start-up and EOL scenarios. The flow-induced vibration and tube wear analysis is addressed in LR section 2.2.2.5.4, Steam Generator Tube Vibration and Wear Evaluation.

	Start-Up Conditions	End-of-Life Conditions
Tube Plugging	0%	10%
Average Primary Temperature	576°F	564.5°F
Steam Nozzle Pressure	823.2 psia	700 psia
Feedwater Temperature	390-435°F	390-435°F
Primary Flow Rate	85,100 gpm	85,100 gpm

A blow-down flow rate corresponding to approximately 1% of steam flow was assumed in the analysis.

Steam carry-under, which is a linear function of the actual water level (AWL), was accounted for in the thermal-hydraulic calculations.

Appropriate values of tube fouling resistances were considered for both start-of-life and end-of-life conditions.

The following thermal-hydraulic acceptance criteria were adopted for the EPU conditions:

- moisture carry-over < 0.1% of steam flow.
- two-phase stability ratio > 0.2

Description of Analyses and Evaluations

The thermal-hydraulic performance evaluation consisted of a steady-state heat balance using the Babcock & Wilcox Company (B&W) one-dimensional nuclear code CIRC (B&W Document, Rev. 2, *CIRC Users Guide – Thermal Hydraulic Analysis Program*) as well as a full 3-D flow field analysis using the thermal hydraulic program ATHOSBWI (B&W Document, Rev. 1, *ATHOSGPPBWI & ATHOSBWI Users Guide – Three-dimensional Thermal Hydraulic Analysis of Nuclear Steam Generators*). The thermal-hydraulic models developed for the original analysis of the Ginna station steam generators were utilized. The input conditions corresponded to the EPU conditions as

described above. Water level measurement uncertainty analysis for the uprated conditions was performed to determine the AWLs from the measured water levels.

As described in Reference 1, start-up, EOL, and FIV cases were considered in the analysis. Secondary side inventory was determined for normal water level. All of the analyses were documented in the *Thermal-hydraulic Performance of Replacement Steam Generators at Power Uprate Conditions* report (Reference 2).

FIV aspects are discussed in LR section 2.2.2.5.4, Steam Generator Tube Vibration and Wear Evaluation.

- **Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs**

Ginna has evaluated the impact of the EPU on the conclusions reached for the evaluation of the Steam Generator Integrity Program in the Ginna License Renewal Application. Ginna has concluded that the Steam Generator Integrity Program remains adequate, following implementation of the EPU.

Steam Generator Thermal Hydraulic Evaluation Results

Table 2.2.2.5.3-1 compares thermal-hydraulic attributes for start-up at original conditions (B&W Report: BWI-222-7705-PR-01, Rev. 1, *Thermal-hydraulic Performance Report*), start-up at EPU conditions, and end-of-life at EPU conditions. All thermal-hydraulic attributes were demonstrated to be acceptable for operation at the EPU conditions.

The secondary side pressure loss between feed-water inlet and steam outlet nozzles was determined to increase by approximately 7 psi for the EPU conditions. However, this is considered negligible when compared with the feedwater system pressure enveloped in the original design analysis. The primary side pressure drop is essentially unchanged, other than coolant density effects, since the primary coolant flow remained unchanged for the EPU conditions. The secondary side fluid inventory was approximately 9% lower than that for the original design basis analysis due to the increase in void fraction and lower uncertainty for the EPU conditions.

All the thermal-hydraulic parameters were within expected ranges for the EPU for tube plugging up to 10%.

The two-phase stability ratio generally decreases with power. Although the EPU start-up conditions stability ratio was determined to be reduced by 30% from the current non-EPU full power conditions value, it remained well above the 0.2 value considered a conservative minimum for stable operations free of water level oscillations (Reference 2).

From the moisture separator test results, the moisture carry-over of a separator increases with the amount of steam flow through it (B&W Report: BWI-222-7705-PR-07, Rev. 0, *Moisture Separation Equipment Report*). From the separator performance curves, it was determined that with the increased steam flow associated with the EPU conditions, the moisture carry-over will remain well below the original design limit of 0.1% of steam flow (Reference 2) for operation at the EPU conditions.

Steam Generator Thermal-Hydraulic Evaluation References

1. B&W Specification No. TS-3270, Rev. 1, *Constellation Energy R.E. Ginna Station Certified Design Specification for Replacement Steam Generator.*

2. B&W Report BWC-143O-PR-01, Rev. 1, *Thermal-hydraulic Performance of Replacement Steam Generators at Power Uprate Conditions.*

3. B&W Report BWI-222-7705-PR-01, Rev. 1, *Thermal-hydraulic Performance Report*

Table 2.2.2.5.3-1
Ginna Station EPU Evaluation Summary
Thermal-Hydraulic Results for Bounding Cases

Thermal-hydraulic Attribute		Start-Up for original Conditions (Ref. 3)	Start-Up for EPU Conditions (Ref. 2)	End-of-Life for EPU Conditions (Ref. 2)	Acceptance Limits
NSSS Power [MWt]		1524.6	1817	1817	-
Plugging [%]		0	0	10	-
Temperatures [°F]	Feedwater	425	435	435	-
	Primary Inlet	601.7	611.9	601.0	-
	Avg. Primary	573.5	576.0	564.6	-
	Primary Outlet	545.4	540.1	528.1	-
Pressures [psia]	Steam Nozzle	877	823.2	700	-
Flow Rates [10^6 lbm/hr] per steam generator	Steam	3.26	3.944	3.929	-
	Primary Fluid	34.95	32.41	32.90	-

Steam Generator Thermal Hydraulic Evaluation Conclusions

Based on the results of the thermal-hydraulic analyses performed and the associated results tabulated in Table 2.2.2.5.3-1, the Ginna station steam generators will continue to comply with the currently applicable design criteria for operation at the EPU conditions.

2.2.2.5.4 Steam Generator Tube Vibration and Wear Evaluation

Technical Evaluation

Introduction

Flow-induced vibration (FIV) analysis and tube wear calculations were performed to demonstrate that the steam generator tubes continue to be adequately supported to prevent detrimental FIV and fretting wear at the EPU conditions. It was demonstrated that the tubes continue not to be susceptible to fluid-elastic instability and that the accumulated tube wear over the life of the steam generators remains below the established allowable limits.

Input Parameters, Assumptions and Acceptance Criteria

The assessment of FIV and tube wear was performed for the conditions provided in the Steam Generator Certified Design Specification (B&W Specification No. TS-3270, Rev. 1, *Constellation Energy R.E. Ginna Station Certified Design Specification for Replacement Steam Generator*). The bounding case scenario corresponds to the end-of-life (EOL) conditions at full power (1817 MWt) with 10% tubes plugged and an end-of-life tube fouling resistance. The input steam nozzle pressure of 750 psia and the high feedwater temperature of 435°F were conservatively bounding for the FIV assessment as these conditions produced the highest secondary side velocities. The EOL conditions effectively bounded the start-up conditions for FIV responses. The thermal-hydraulic analysis methodology for FIV conditions was similar to the analysis in LR section 2.2.2.5.3, Steam Generator Thermal-Hydraulic Evaluation.

The FIV analysis of the RSG tubes was performed with a SG nozzle pressure of 750 psia. Although the 750 psia RSG pressure is higher than the minimum SG pressures used by Westinghouse in their PCWG (LR section 1.1) sheet for the low Tavg condition, it represents a conservative estimate of the actual full power RSG pressures that can be expected due to uprate. The uprate will replace the existing High Pressure Turbine (HPT) with a new unit. The full power HPT inlet pressure being used to procure the new HPT is 730 psia. Since the pressure drop in the Main Steam piping system at uprate is ~70 psi, the minimum possible SG pressure for full power operation is ~800 psia. Therefore, using 750 psia is conservative.

The applicable EPU condition acceptance criteria, as described in Reference 1, were:

- critical velocity ratio < 1.0 precluding fluid-elastic instability.
- accumulated tube wear over 40-year life < 40% nominal tube wall thickness.

Description of Analyses and Evaluations

FIV and wear analysis of the steam generator tubes at the EPU conditions was performed for the potential FIV mechanisms of fluid-elastic instability, vortex shedding resonance (VS), and random turbulence excitation (RT). Critical regions in the tube bundle were analyzed, taking into account crudded support conditions.

For the FIV analysis the tubes with the largest spans and maximum gap velocities were chosen from the thermal-hydraulic analysis performed at bounding EPU conditions (Reference 1) because of their potential for the largest vibration responses. Homogenous flow was considered since this would produce higher, conservative mixture velocities.

In the tube wear analysis, a work rate was determined from the integral average of the normal contact force multiplied by the sliding distance over the tube-to-support interface for each mode shape. This work rate was then converted into a wear volume and an equivalent wear depth based on wear coefficients for the tube and support materials.

Both the FIV and wear analyses were documented in the *Flow-induced Vibration and Wear Analysis of Replacement Steam Generators at Power Uprate Conditions* report (Reference 1).

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached for the evaluation of the Steam Generator Integrity Program in the Ginna License Renewal Application. Ginna has concluded that the Steam Generator Integrity Program remains adequate, following implementation of the EPU.

Steam Generator Tube Vibration and Wear Evaluation Results

For the EPU conditions, the maximum calculated critical velocity ratio for fluid-elastic instability in the U-Tubes is 0.874 which satisfies the < 1.0 acceptance limit (Reference 1) used for assessing flow induced vibration (FIV). The EPU calculated value compares to a maximum calculated ratio of 0.82 for the original design basis (Reference 2) prior to EPU. The FIV calculated results are conservative since they are based on homogenous flow models which calculate higher flow velocities across the tubes. Additionally, the velocity ratio is calculated by conservatively assuming fully "crudded" supports in the U-

tube region. Fully crudded, locked supports are not expected to occur with the lattice grid and U-bend flat bar support design features utilized by the replacement steam generators. The combination of a conservative full power SG pressure, homogenous flow and locked crudded supports ensures that the calculated velocity ratio is bounding for the EPU conditions.

The maximum expected tube wear at EPU conditions was also assessed. The results of this assessment demonstrated that over a 40 year operating life the EPU did not increase the maximum calculated tube wear as compared to the original design basis operating condition. The maximum calculated tube was approximately 50% of the replacement steam generator design limit for tube wear over a 40 year operating life.

Steam Generator Tube Vibration and Wear Evaluation References

1. The B&W Report, BWC-143O-FIV-01, Rev. 1, *Flow-induced Vibration and Wear Analysis of Replacement Steam Generators at Power Uprate Conditions.*

2. The B&W Report, BWI-222-7705-FIV-01, Rev. 00, *Flow-Induced Vibration Analysis Report.*

Steam Generator Tube Vibration and Wear Evaluation Conclusions

The above results demonstrate that the steam generator tube bundles are adequately designed and supported for the FIV and fretting wear that might occur over the 40-year design life with operation at the EPU conditions.

2.2.2.5.5 Steam Generators and Supports Overall Conclusion

Each of the preceeding sub-subsections which presented the evaluation results for steam generator supports, structural integrity, thermal-hydraulic performance, and tube vibration and wear concluded that the pertinent acceptance criteria are met. Therefore, steam generators and supports will continue to perform acceptably under the proposed EPU conditions.

2.2.2.6 Reactor Coolant Pumps and Supports

Introduction

The reactor coolant pumps (RCPs) are described in the *Ginna Updated Final Safety Analysis Report* (UFSAR), Sections 3.9.2.2.4.9 and 5.4.1. The RCP supports are described in Ginna UFSAR Section 3.9.3.2.3. Each reactor coolant loop contains a vertical single-stage centrifugal type pump that employs a controlled leakage seal assembly. The functions of the RCPs are

- to maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the reactor coolant system (RCS).
- to provide adequate flow coastdown to prevent core damage in the event of a simultaneous loss of power to both pumps.
- to provide a portion of the reactor coolant pressure boundary (the pressure boundary parts of the RCP).

The Reactor Coolant Pump Supports (RCP columns and lateral tierods) were reviewed for the impact of Reactor Coolant Loop piping loads under EPU conditions. Supports for the Reactor Coolant Pumps and for the RCL piping are discussed in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in Section 3.9.3. The Reactor Coolant Loop (RCL) piping loads under EPU conditions, are discussed in LR section 2.2.2.1, "NSSS Piping, Component, and Supports", The RCL piping loads on the RCP supports due to the deadweight, thermal expansion, seismic operational basis earthquake (OBE), and seismic safe shutdown earthquake (SSE) loading cases are obtained from LR section 2.2.2.1. The LOCA and the pipe break analyses from the current design basis in Reference 1 remains valid for the EPU program as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports.

The RCP supports stress margin values are evaluated for the EPU program based on the stress margin data from the current design basis in Reference 1 and the RCP support loads obtained from the evaluation from the RCL piping system analyses for the EPU program as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports.

Ginna Current Licensing Basis

The generic Current Licensing Basis in section 2.2.2, above, applies to the reactor coolant pumps and supports. In addition to the general discussion regarding review for plant license renewal in section 2.2.2, the specific review for reactor coolant pumps and

supports is documented in the license renewal Safety Evaluation Report (SER), Sections 2.3.1.1, 3.1.2.3, 4.3, 4.7.6, and 4.7.7.

Section 2.3.1.1 of the SER identifies that the reactor coolant pump casing, main flange, thermal barrier flange, the bolting for the reactor coolant pump main closure, and the reactor coolant pump lugs are among the component groups of the reactor coolant system that require an aging management review. Section 3 of the SER identifies the aging management programs that are applicable to reactor coolant system components.

On the basis of its review, the staff concluded that RG&E had adequately identified the aging effects and the aging management programs credited for managing the effects of the reactor coolant (Class 1) components, such that there is reasonable assurance that the component intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

Time-limiting aging analyses (TLAAs) are discussed in Section 4.0 of the SER. In particular, Section 4.3 of the SER discusses metal fatigue. The reactor coolant pump is included in the discussion of metal fatigue as it relates to ASME Boiler and Pressure Vessel Code, Section III, Class 1 equipment. To address this issue, estimates were made that the expected number of transients to which the reactor coolant pump would be subject over the 60 year life of the plant with license renewal would be bounded by the number of transients considered in the analyses performed for the original 40 year life of the plant. A commitment was also made to implement a Fatigue Monitoring Program to provide assurance that the number of design cycles would not be exceeded during the period of extended operation. The staff agreed that the number of design cycles given in UFSAR Table 5.1-4 for the listed transients is conservative. The staff also found that renewal applicant action item 8 in WCAP-14575-P, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components", was adequately addressed by the Fatigue Monitoring Program for those components labeled I-M and I-RA in Tables 3-2 through 3-16 in WCAP-14575-P. It is noted that there are no RCP components labeled I-M and I-RA in these tables.

It is also noted that the reactor coolant pumps are not among the fatigue sensitive component locations which are included in the evaluation of environmentally assisted fatigue discussed in section 4.3.17 of the SER.

Section 4.7.6 of the SER discusses the reactor coolant pump flywheel TLAA. Westinghouse Topical Report WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination", presents an evaluation of the probability of failure over an extended operating period of 60 years. Based on WCAP-14535A and in accordance with NRC recommendations, Rochester Gas & Electric requested and received a relief request from the NRC allowing it to revise the ISI frequency of flywheel examination to once every 10 years. The staff concluded that the Inservice Inspection

Program requirements for the reactor coolant pump flywheels at RG&E will continue to ensure that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

Section 4.7.7 of the SER discusses the TLAA on thermal aging of cast austenitic stainless steel. The TLAA on this topic that is applicable to the reactor coolant pumps was documented in WCAP-15873, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R. E. Ginna Nuclear Power Plant for the License Renewal Program". In WCAP-15873, a fracture mechanics analysis (flaw tolerance) was performed for the cast austenitic stainless steel reactor coolant pump casings according to the requirements of ASME Code Case N-481 for the period of the extended operation. The results of the analysis showed that the stability criteria will be met with the fracture toughness of the pump casing materials in a fully aged condition. Therefore, for the Inservice Inspection of cast austenitic stainless steel pump casings at the Ginna Station, the alternative visual examinations as delineated in Code Case N-481 can be performed in lieu of the volumetric examinations required by ASME Code, Section XI. The staff noted that RG&E's ASME Code, Section XI, Subsections IWB, IWC, and IWB Inservice Inspection Program is required to be updated by RG&E and reviewed by the staff every 10-year interval, and that the acceptability of using Code Case N-481 as an alternative requirement for the Inservice Inspection of the pump casing would be evaluated by the staff during each 10-year review.

Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The major inputs used in the RCP evaluation are the EPU parameters provided in LR section 1.1, Nuclear Steam Supply System Parameters, and the EPU nuclear steam supply system (NSSS) design transients provided in LR section 2.2.6, NSSS Design Transients. These LR section provide the operating and transient conditions for the EPU conditions. The RCPs are installed in the RCS cold leg, between the steam generator outlet and the reactor vessel inlet. Therefore, the cold-leg temperatures and the cold-leg transients are applicable to the RCPs. These operating and transient conditions differ in some cases from those specified in the RCP equipment specification, to which the Ginna Station RCPs were already designed and analyzed.

The EPU parameters and EPU NSSS system design transient parameters were considered in the EPU evaluations. There are no other changes to the pressure or thermal-hydraulic design parameters due to the EPU that would affect the reactor coolant pumps or their supports. For design load under EPU conditions, there were no changes to the design loads, load application points, or number of occurrences.

The inputs for seismic analysis of the reactor coolant pump, including seismic accelerations and pump component mass and stiffness, have not changed due to the EPU conditions. The power required to operate the pump under the EPU conditions remains within the nameplate rating of the motor. Therefore, seismic analyses and non-pressure boundary component evaluations were considered to be unaffected by the EPU. The evaluation of the RCPs for the EPU compared the operating temperatures and pressures defined in the EPU parameters to the pressures and temperatures considered in previous analyses of the RCPs. In addition, the NSSS design transients for the EPU were compared to the transients considered in previous evaluations. Where temperatures, pressures, and NSSS transients considered in previous analyses enveloped the temperatures, pressures, and NSSS transients defined for the EPU, no additional analysis was required. For the inputs that were not enveloped by the previously analyzed parameters, RCP structural analyses and evaluations were performed as necessary to incorporate the revised design inputs. Where these analyses and evaluations were required, the acceptance criteria were that the Ginna Station RCP pressure boundary components meet the stress limits and fatigue usage requirements of the American Society of Mechanical Engineers (ASME) Code, Section III for plant operation with the EPU conditions. While the RCP is not an ASME Code pressure vessel, the pressure retaining parts of the RCPs were designed, fabricated, inspected, and tested in conformance with the ASME Code.

The reactor coolant pump motors were evaluated for the Ginna Station EPU parameters provided in LR section 1.1, Nuclear Steam Supply System Parameters, and best-estimate flows at an assumed core power of 1811 MWt. The input parameters considered in the evaluation of the reactor coolant pump motors for the Ginna Station EPU Program are for a range of steam generator outlet temperatures from 528.0° to 539.9°F. The range of best-estimate flows considered is from 96,800 to 94,800 gpm/loop for a range of steam generator tube plugging (SGTP) from 0 to 10% SGTP at full power operation. For the cold condition (70°F), the range of best-estimate flows considered is from 91,100 to 89,000 gpm/loop for 0 to 10% SGTP.

The steam generator outlet temperatures and best-estimate flows were considered in a hydraulic analysis using the operating characteristics of the Ginna Station reactor coolant pumps. This hydraulic analysis calculates the power requirements for the impeller that operates at the highest power for both hot and cold operation. The RCP motors were evaluated to confirm that they continue to meet their design requirements.

The RCL piping loads on the RCP supports due to deadweight, thermal expansion, seismic OBE, and seismic SSE loading cases are obtained from the evaluation for the RCL piping system analyses for the EPU program as described in LR section 2.2.2.1, NSSS Piping, Components, and Supports. The loss-of-coolant accident (LOCA) analyses from the current design basis in Reference 1 remain valid for the EPU program.

The acceptance criteria for the Ginna Station RCL piping RCP supports are as used in the current design basis in Reference 1 and as indicated in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in Section 3.9.3 and are based upon the *ASME Boiler and Pressure Vessel Code* (B&PV Code), Section III, Subsection NF and Appendix F, 1974 Edition.

Description of Analyses and Evaluations

Operating Temperature and Pressure

The EPU parameters (see LR section 1.1, Nuclear Steam Supply System Parameters) for Ginna Station were used to evaluate the acceptability of the RCPs. In the EPU parameters for Ginna Station, there are no changes from the current reactor coolant pressure of 2250 psia for any of the EPU cases. For EPU, the reactor coolant system cold-leg temperature (T_{cold}), defined by the vessel inlet (RCP outlet) temperature, is a maximum of 540.2°F and a minimum of 528.3°F. Since lower temperatures result in higher allowable stresses for the pressure boundary materials, a decrease in operating temperature is conservative. The maximum EPU RCS T_{cold} is less than the equipment specification operating temperature of 556°F. Since none of the EPU temperatures exceed the previously considered temperature and the pressure does not change, the EPU NSSS parameters are bounded by those defined in the equipment specification. No further evaluation of the Reactor Coolant Pump pressure boundary integrity was required for the operating temperature and pressure associated with the Ginna Station EPU.

Transient Discussion

The NSSS design transients were recalculated for the Ginna Station EPU Program. The recalculated transients had some temperature and pressure changes that were different than those in the design transients given in the equipment specification or used in the original analyses.

The EPU NSSS design transients are provided in LR section 2.2.6, NSSS Design Transients. The cold-leg transients are applicable to the RCP evaluation. Since there was some variation in the transients considered in the original analyses, the comparison of the revised transients to the analyzed transients was approached on the basis of the various original stress analyses.

Main Flange Bolted Joint Stress Analysis

This analysis showed that the only transients with a potential for affecting the fatigue usage of the main flange bolted joint are the heatup and cooldown transients. Because the heatup and cooldown transients remained unchanged for EPU, the cumulative usage factor for the main flange studs remained as calculated. In the original analysis the highest stress in the studs occurred for the loss-of-load transient, which had a maximum pressure increase of 500 psi. For the EPU the maximum pressure increase is 388 psi, and thus the original analysis results envelope the maximum stress levels for the EPU conditions. The stresses remain within the ASME Code allowable values. The values of the stresses and the cumulative usage factor are shown in Table 2.2.2.6-1.

Main Flange Stress Analysis

The main flange is protected from the high reactor coolant temperatures by the thermal barrier, and thus sees much lower temperatures than those in the parts of the pump directly exposed to the primary coolant. This protection also means that the effect of the primary system cold-leg transients on the main flange is small, except for the heatup and cooldown transients. As was the case for the main flange bolted joint analysis, the original analysis results envelope the maximum stress levels for the EPU conditions. The stresses remain within the ASME Code allowable values. The values of the stresses and the cumulative usage factor are shown in Table 2.2.2.6-1.

Pump Casing Stress Analysis

The transients considered in the original, non-plant specific, analysis of the casing differed from the transients specified for the Ginna Station RCPs, for both the present and for the proposed EPU operating conditions, and in most cases the non-plant specific transients were more severe. The exception to this was the temperature range spanned by the heatup and cooldown transients. In the original stress analysis, the temperature range spanned 433°F, while the Ginna Station temperature range for EPU spans 447°F, from 100°F to 547°F (547°F is the hot zero power temperature). The maximum values of the stress intensity occur at the suction nozzle area of the casing. Adjusting the calculated thermal stresses for this difference resulted in small increases in the primary-plus-secondary-stress intensity and the maximum thermal-plus-pressure-plus-mechanical-stress intensity. The stress intensities remain less than the ASME Code allowable values. The values of these stress intensities are shown in Table 2.2.2.6-1. It is noted that the hot zero power temperature has not changed as a result of the EPU. This change in stresses is a result of updating the values originally calculated to more accurately reflect the Ginna Station hot zero power temperature that has always existed.

The original calculations resulted in a cumulative usage factor of zero since the calculated stresses were below the lowest alternating stress value on the fatigue curve. This is still true for the Ginna EPU conditions.

Support Foot Analysis

The support foot was considered a structural member in the original stress analysis and was analyzed only for mechanical loads. There was no transient analysis. Thus, changes to the NSSS design transients did not affect the support foot analysis.

Flow Induced Vibration Analysis

Analysis of flow induced vibration is not included in the licensing basis for Ginna. However, it was considered for more susceptible components that would experience a

significant flow increase under EPU conditions. Reactor coolant pumps were considered and deemed unaffected by EPU conditions due to their heavy construction and small increase in flow.

RCP Motors

For the RCP motors, a hydraulic analysis was performed using best estimate flows and modeling the characteristics of the Ginna RCPs. The hydraulic analysis is used to calculate the required brake horsepower for the RCP motors and the loading on the thrust bearings.

RCP Supports

The RCP support loads from the RCL piping system analyses as described in LR section 2.2.2.1, NSSS Piping, Component, and Supports and the current design basis RCP support loads and stress margins from Reference 1 are used to calculate the stress margins available for the EPU program for the RCP supports. The RCP lateral tierods, and the RCP columns are evaluated for the stress margin values for the EPU program. The stress margin values are summarized and are tabulated in Table 2.2.2.6-2, RCL Primary Equipment Support Loads Stress Margin Summary, for the loading combinations as specified in the acceptance criteria in the Code of Record as evaluated in the current design basis in Reference 1, and as in the *Ginna Updated Final Safety Analysis Report* (UFSAR) in Section 3.9.3.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

There is no change to the number of design transient cycles, and there is no change to the cumulative usage factors for the reactor coolant pump pressure boundary components based on those cycles. While a small increase in stress was calculated for the casing, as described above, stresses remain within ASME Code allowables. The operating temperature of the reactor coolant pump is not increasing. There is thus no impact on the License Renewal programs and renewed plant operating license evaluations as described in the License Renewal SER.

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the RCP supports. The aging evaluations approved by the NRC in the License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant NUREG-1786 for the RCP supports remain valid for the EPU conditions.

Reactor Coolant Pumps and Supports Results

The results of the evaluations of the RCP pressure retaining components are given in Table 2.2.2.6-1 for the major RCP pressure retaining components. Some of these components required the recalculation of stresses or cumulative usage factors for the EPU conditions, while for other major components, no changes to the stress and usage factor were necessary. All of the stresses and cumulative usage factors given in Table 2.2.2.6-1 are within the allowable values given in the ASME Code.

The reactor coolant pump motors were evaluated in the following three areas for the Ginna Station EPU Program conditions under loadings of 5625 hp for worst-case hot-loop operation and 7207 hp for worst-case cold-loop operation:

- Continuous operation at hot-loop (100% power) conditions
- Continuous operation at cold-loop (70°F) conditions
- Thrust bearing loading

The RCP motor brake horsepower results are as given in Table 2.2.2.6-3, Reactor Coolant Pump Motor Brake Horsepower. The worst-case hot-loop load under the EPU operating conditions is 5625 hp. The worst-case cold-loop load under the EPU operating conditions is 7207 hp. These loadings are less than the motor nameplate ratings of 6000 hp for hot-loop operation and 7500 hp for cold-loop operation. The motors have been shown by test and analysis to operate within the equipment specification limits at the nameplate ratings. Per design, motor operation is acceptable for any load up to the hot-loop nameplate rating of 6000 horsepower (hp) and the cold-loop nameplate rating of 7500 hp. Thus, the revised motor loadings are acceptable based on the loadings being within the nameplate ratings of the motors.

The thrust-bearing loading used for the motor design is given in the equipment specifications for the motor. Performance of the thrust bearings in a reactor coolant pump motor can be adversely affected by excessive or inadequate loading. The analysis for the Ginna Station EPU Program conditions indicates an increase in the downward impeller thrust from 55,000 lbs. to 60,632 lbs. for hot-loop operation, and an increase from 75,000 lbs. to 79,503 lbs. for cold-loop operation. For hot-loop operation, this increase in impeller down thrust results in a net decrease in the up thrust loading on the thrust bearing. For cold-loop operation, the increase in impeller down thrust loading results in a net increase in the down thrust loading on the thrust bearing of 4.7%. In comparison to the normal operating thrust-bearing load of 101,200 lbs given in the equipment specifications, these changes are not considered significant and the thrust bearings are acceptable for the EPU Program loads.

Since the new reactor coolant pump motor loads are within the nameplate ratings of the motors, the motor temperature rise for hot and cold operating conditions will be within the NEMA requirements and the first two areas continue to meet these requirements. In

comparison to the normal operating thrust-bearing load of 101,200 lbs, the thrust bearing load changes due to the EPU are not considered significant and the thrust bearings remain acceptable for the EPU loads. Therefore, the reactor coolant pump motors at Ginna Station are acceptable for the EPU Program conditions.

There are no changes required as a result of the EPU for the reactor coolant pumps and motors supporting systems such as cooling water, seal injection flow, or lubricating oil/lube oil spillage collection.

The stress margins available for the EPU conditions for the RCP lateral tie rods, as well as the RCP columns are evaluated and summarized in Table 2.2.2.6-2, RCL Primary Equipment Support Loads Stress Margin Summary. The stress margin values in Table 2.2.2.6-2 for the different loading combinations for the EPU program are either equal to or less than the stress margin values in the current design basis in Reference 1. Based on the review of the stress margin values for the EPU program in Table 2.2.2.6-2 for the RCP supports, the RCP supports are all acceptable for the EPU program.

References

1. Final Report for the Robert Emmett Ginna Nuclear Generating Station Steam Generator Hydraulic Snubber Replacement Program, October 1, 1992. Evaluation of the Reactor Coolant System for the Steam Generator Hydraulic Snubber Replacement Program, September 1992.).

Table 2.2.2.6-1
RCP Pressure Retaining Component Stresses and Usage Factors

Component	EPU Stresses and Usage Factors		Allowable	Comments
Casing	Primary Membrane Stress Intensity	[ ]a,c	16,700 psi (S_m)	The original calculations had a zero usage factor since the calculated stresses were below the lowest S_a value on the fatigue curve. This is still true for the Ginna Station EPU conditions.
	Primary + Secondary Pressure and Mechanical Loads	[ ]a,c	25,000 psi (1.5 S_m)	
	Maximum Steady State Thermal + Pressure and Mechanical Stresses	[ ]a,c	50,100 psi (3 S_m)	Original stress intensities were 16,359 psi, 22,924 psi, and 41,898 psi, respectively.
Main Flange	General Primary Membrane Stress Intensity	[ ]a,c	20,000 psi (S_m)	No changes from the original calculation.
	Local Primary Membrane Stress Intensity	[ ]a,c	30,000 psi (1.5 S_m)	
	Primary + Secondary Stress Intensity	[ ]a,c	60,000 psi (3 S_m)	
	Usage Factor	[ ]a,c	1.0	
Main Flange Studs	Maximum Service Stress, Averaged Across Section	[ ]a,c	55,800 psi (2 S_m)	No changes from the original calculation. Allowable stresses are based on SA-193, Grade B7 material. Current drawings show the stud material is SA-540, Grade B24, Class 4, or Grade B23, Class 4. Both of these have higher allowable stresses than the SA-193, Grade B7 material originally considered.
	Maximum Service Stress at Periphery of Cross Section from Tension + Bending	[ ]a,c	83,700 psi (3 S_m)	
	Usage Factor	[ ]a,c	1.0	

Table 2.2.2.6-2
RCL Primary Equipment Support Loads Stress Margin Summary
(Stress Margin = Allowable/Actual) (Based on $K_{average}$)

Service Level	Normal	Upset	Emergency	SSE	Faulted (See Note 1)
Load Combination	Deadweight + Thermal Normal	Deadweight + Thermal (Normal +Overtemperature) + OBE	Deadweight + Thermal Normal + DBPB	Deadweight + Thermal Normal + SSE	Deadweight + Thermal Normal + $[(SSE)^2 + (PIBK^2)]^{1/2}$
Reactor Coolant Pump Lateral Tierods	See Note 2	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c
Reactor Coolant Pump Columns	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c

Note:
1. PIBK (pipe break) includes DBPB (design basis pipe break).
2. Under normal conditions no significant loads are imposed on these lateral support elements.

Table 2.2.2.6-3 Reactor Coolant Pump Motor Brake Horsepower				
	Case 1	**Case 2**	**Case 3**	**Case 4**
Hot Condition Inlet Temperature, °F	528.0	528.0	539.9	539.9
Hot Condition Flow Rate, gpm/loop	96,800	94,800	96,800	94,800
Hot Condition Brake Horsepower	5553	5625	5476	5547
Cold Condition Flow Rate, gpm/loop	91,100	89,000	91,100	89,000
Cold (70°F) Condition Brake Horsepower	7137	7207	7137	7207

Reactor Coolant Pumps and Supports Conclusions

Because the calculated stresses and cumulative usage factors remain below the allowable values given in the ASME Code, the structural integrity of the RCPs and supports remains adequate under the proposed EPU conditions. The EPU conditions, and the analyses and evaluations performed in support of the EPU, do not impact the aging management reviews, aging management programs, and TLAAs associated with the reactor coolant pumps and supports for the Ginna Station License Renewal. The reactor coolant pumps and supports continue to meet the current licensing basis with respect to GDC-1, 2, 4, 14, and 15.

As shown by the stress margin values summarized in Table 2.2.2.6-2, RCL Primary Equipment Support Loads Stress Margin Summary, the RCP supports (RCP columns and lateral tierods) are acceptable for the EPU program.

The RCP motors under EPU conditions continue to have acceptable motor loadings, thrust bearing loadings, and temperature rise and therefore are acceptable for EPU.

2.2.2.7 Pressurizer and Supports

Introduction

The Ginna Station pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads, constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The pressurizer is insulated to minimize heat loss from the pressurizer vessel. The insulation consists of reflective panels that are removable to permit visual examination of the pressurizer as required by the Inservice Inspection (ISI) Program document.

The pressurizer vessel contains replaceable direct immersion heaters, multiple safety and pressurizer power operated relief valves (PORVs), a spray nozzle, and interconnected piping, valves and instrumentation. The heaters are sheathed in austenitic stainless steel.

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure and, in conjunction with the pressure control system components, keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow to, or outflow from, the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{sat}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer, and can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (such as the surge nozzle, lower head/heater well, and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (such as the spray nozzle, safety and relief nozzle, upper head/upper shell, manway, and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg and cold leg temperatures are low. This maximizes the ΔT that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS hot leg at T_{hot}. If the RCS pressure is high (which means, correspondingly, that T_{sat} is high) and T_{hot} is low, then the surge nozzle will see maximum thermal gradients, and thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at T_{sat} and spray water, which, for many transients, is at T_{cold}. Therefore, if RCS pressure is high (T_{sat} is high) and T_{cold} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

An evaluation was performed to support the Ginna Station EPU to address the impact on the pressurizer. This evaluation was based on the range of nuclear steam supply system (NSSS) operating parameters to support an NSSS power level of 1817 MWt.

Ginna Current Licensing Basis

The generic Current Licensing Basis in LR section 2.2.2, above, applies to the pressurizer and supports, with the following amplifications.

The Ginna Station pressurizer is designed, fabricated, inspected and tested in accordance with the 1965 Edition of Section III of the ASME Boiler and Pressure Vessel Code. Quality control techniques used in the fabrication of the reactor coolant system were equivalent to those used in the manufacture of the reactor vessel which conforms to Section III of the ASME Code. The pressurizer design pressure is 2485 psig. This design pressure allows for operating transient pressure changes. The pressurizer is designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes is shown in Table 5.1.4 of the UFSAR. For the pressurizer, the heatup and cooldown rates do not exceed 100°F per hour and 200°F per hour, respectively. An additional limitation is that spray cannot be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F.

NRC Bulletin 88-11 requested licensees to take certain actions to monitor thermal stratification in the pressurizer surge line because measurements indicate that top-to-bottom temperature in the surge line can reach 250°F to 300°F in certain modes of operation, particularly during heatup and cooldown. The generic evaluation of surge line stratification in Westinghouse PWRs was reported in WCAP-12639 submitted to the NRC in June of 1990. Westinghouse performed a plant specific analysis of the Ginna pressurizer surge line to demonstrate compliance with NRC Bulletin 88-11, and the results are reported in WCAP-12968. The results indicate that the surge line meets the stress limits and usage factor requirements, and the pressurizer surge nozzle meets the code stress allowables under thermal stratification loading and fatigue usage requirements of ASME Section III, 1986 edition. Plant operating strategies as discussed in WCAP-13588, "Operating Strategies for Mitigating Pressurizer Insurge and Outsurge Transients", dated March 1993, have an effect on the frequency and severity of pressurizer flow surges. Work performed by Westinghouse for the Westinghouse Owners Group (WOG) during the past decade has shown that modified plant operational strategies such as those that have been implemented at the Ginna Station can provide an effective means of mitigating pressurizer insurge-outsurge transients.

The pressurizer is classified as Seismic Category I, requiring that there be no loss of function in the event of the assumed maximum potential ground acceleration in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses. As part of the Systematic Evaluation Program (SEP), the NRC evaluated, in part, the stresses in the

pressurizer and it was concluded that the vessel and supports were acceptably designed, with the stress results within established limits. Based on analyses of a heavier, 1800 ft^3, model (but with support skirts identical to the Ginna 800 ft^3 model) and using a finite element model, it is concluded that the Ginna pressurizer is adequately supported for the 0.2g safe shutdown earthquake.

In addition to the basis described in the regulatory evaluation, section 2.2.2 above, the Ginna Station pressurizer and supports have been evaluated for plant license renewal. The WOG issued Generic Topical Report WCAP-14754-A, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers" to address the aging management of pressurizers. WCAP-12928, "Structural Evaluation of the Robert E. Ginna Surge Line, Considering the Effect of Thermal Stratification, " issued in May 1991, addresses the potential for environmentally assisted fatigue during the period of extended operation. Also, the NRC staff issued License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated March 2004. Also see LR section 2.14.3, Impact of EPU on Environmentally Assisted Fatigue Evaluations.

During the review of the Ginna license renewal application by the NRC staff, the following commitments were made for the pressurizer component in order to provide aging management programs to manage the aging effects of structures and components prior to the extended period of operation (see Appendix A of NUREG-1786):

- Implement a fatigue monitoring program to confirm that the number of operating cycles (causing fatigue) are fewer than the plant design cycles.

- Provide a baseline nondestructive examination for the pressurizer surge line by inspecting all circumferential welds, and develop a methodology to employ NRC-approved augmented inservice inspection for the pressurizer surge line, or recalculate to determine an acceptable cumulative fatigue usage factor, or repair/replace surge line, as necessary.

- Prior to September 2009, the pressurizer manway stainless steel insert will receive a visual and surface examination as part of the applicant's Inservice Inspection Program to detect the potential for stress corrosion cracking.

The subcomponents of the pressurizer that require Aging Management Review (AMR) include the lower head, surge nozzle, surge nozzle safe end, heater well and heater sheath, shell, instrument nozzle thermal wells, upper head, spray nozzle and safe end, relief nozzle and safe end, manway cover, support skirt and flange, and manway cover bolts.

The intended functions are pressure boundary, structural support and closure integrity.

Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The major inputs used in this evaluation were the Performance Capability Working Group (PCWG) uprate parameters provided in LR section 1.1, Nuclear Steam Supply System Parameters and the NSSS uprate design transients, LR section 2.2.6, NSSS Design Transients. These provided the operating and transient conditions that differ from those addressed in Equipment Specification 676248, Revision 1 , to which the Ginna Station pressurizer was already designed and analyzed.

The PCWG uprate parameters and NSSS uprate design transient parameters were considered in the EPU evaluations. There are no other changes to the pressure or thermal-hydraulic design parameters due to the EPU that would affect the pressurizer or its supports.

Analysis of flow induced vibration is not included in the licensing basis for Ginna. However, it was considered for more susceptible components that would experience a significant flow increase under EPU conditions. This degradation mechanism is not considered to be a concern for the pressurizer because the components are subject to relatively low fluid flow velocities.

Seismic analyses and non-pressure boundary component evaluations were considered to be unaffected by the EPU as the conditions in the original design specification remain bounding. The load combinations considered in the original design (i.e., normal + design earthquake, normal +maximum potential earthquake) do not change.

The initial set of acceptance criteria for evaluating design inputs affecting the pressurizer stress reports by comparison with the design inputs considered in LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients were as follows:

- T_{hot} and T_{cold} remained within the ranges of the operating temperatures that had previously been considered and justified in the pressurizer stress reports.

- NSSS design transients were less than or equal to the design transients previously considered in the pressurizer stress reports with regard to both severity and number of occurrences. Additionally, no new NSSS design transients that had not previously been considered were identified. The pressurizer temperature and pressure variations for each transient were considered in this comparison review to determine the relative severity of the revised design transients compared to the existing design transients.

- Design loads were less than or equal in magnitude to the loads that were previously considered in the pressurizer stress reports, with no changes to the load application points or number of occurrences.

Design inputs for the Ginna Station EPU were compared with the design inputs in LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients. If differences were found in T_{hot} and T_{cold}, design transients, or design loads, then pressurizer structural analyses and evaluations were performed as necessary to incorporate the revised design inputs. The acceptance criterion was that the Ginna Station pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III for the plant operation in accordance with the EPU.

Description of Analyses and Evaluations

A review of the temperature parameters in LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients showed that the changes in reactor coolant system (RCS) hot leg (T_{hot}) and cold leg (T_{cold}) are enveloped by the original equipment specification (Equipment Specification 676248, Rev. 1, *800 Cu. Ft. Pressurizer – Ginna Plant Unit No. 1*, W. Steffers, February 23, 1966). The changes made to the design transients did not impact the pressurizer, since the primary side transients were either unaffected or not significantly affected. For this reason, it was concluded that the revised parameters would not impact the existing pressurizer stress and fatigue analyses.

The reactor vessel outlet (T_{hot}) and the reactor vessel inlet (T_{cold}) temperatures from the PCWG parameters in LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients define the normal operating temperatures for the surge and spray lines to the pressurizer. The reactor coolant pressure from LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients defines the pressurizer normal operating pressure (2250 psia) and saturated temperature (653°F). The minimum values of T_{hot} and T_{cold} from all cases in LR section 1.1, Nuclear Steam Supply System Parameters and LR section 2.2.6, NSSS Design Transients were used in the evaluation of the pressurizer. The NSSS design transients in LR section 2.2.6, NSSS Design Transients were also applicable to the pressurizer and were considered in this analysis.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The pressurizer performs the intended function of ensuring the integrity of the reactor coolant pressure boundary. The aging effects identified and evaluated in WCAP-14574-A include:

- Fatigue
- Corrosion/SCC/PWSCC
- Irradiation Embrittlement
- Thermal Aging
- Erosion and Erosion/Corrosion
- Wear
- Creep and Stress Relaxation

WCAP-14574-A concludes that the only effect that will require utility action is fatigue. The aging effects of irradiation embrittlement, thermal aging, erosion and erosion/corrosion, wear, and creep and stress relaxation have been determined to not be detrimental to the pressurizer at the current operating or at the EPU conditions. There are no pressure boundary materials in the pressurizer subject to thermal aging embrittlement, erosion, and mechanical wear. For creep and stress relaxation, the temperature during all plant conditions is well below the 1000° F limit of concern reported in Table 3-2 of WCAP-14574-A. The potential effects of SCC/PWSCC and boric acid corrosion can be managed by current industry programs. The only Time-Limited Aging Analysis for the pressurizer is fatigue.

As noted in section 4.3.1.6, Pressurizer Fracture Mechanics Analyses, of the LR SER, pre-service UT examination of the pressurizer detected a "Defect-like" indication in the lower shell-to-head circumferential weld. This indication was reported as a linear reflector with the approximate dimensions of 11.5 inches in length by 0.5 inches in width and embedded partially in the circumference weld and the base metal of the pressurizer shell. Fracture mechanics analysis has been performed by Westinghouse and concluded that the "Defect" would not cause failure of the pressurizer during the design life (40 years) of the component. The most recent UT examination (as well as two other examinations) characterized the indication as consisting of several intermittent, low amplitude indications located in the center third of the weld thickness. These indications were also evaluated and found to meet the acceptance criteria by examination in the ASME Code, Section XI, 1995 Edition with 1996 Addenda. Because it has been demonstrated that the initial indication is actually a number of small, discrete indications, the fracture mechanics analysis completed by Westinghouse is no longer applicable or relevant. Moreover, the NRC staff has concluded in section 4.3.2.6 of the LR SER that fracture

mechanics analysis is no longer required, the analysis does not meet the criteria of a TLAA and no additional analysis is required in support of extended plant operation. The effects of EPU do not change this conclusion.

As discussed in the License Renewal SER, action item 3.3.1-1 requests that Ginna demonstrate that the pressurizer subcomponents cumulative fatigue usage values remain below 1.0 for the period of extended operation, including the insurge/outsurge transients discussed in WCAP-14754-A, while also considering the effects of the reactor coolant environment. On the basis of the projection of the number of design transients, Ginna concluded that the existing fatigue analyses of the pressurizer component will remain valid for the period of extended operation. In addition, Ginna will utilize a fatigue monitoring program to provide assurance that the number of design cycles will not be exceeded during the period of extended operation.

The staff has concluded that Ginna has adequately addressed renewal applicant action item 3.3.1.1-1 in WCAP-14754-A by evaluating the fatigue-sensitive pressurizer components for insurge/outsurge transients, including the effects of reactor coolant environment, and by assuring that the thermal transients that are significant contributors to the design fatigue usage of the pressurizer will be monitored by a fatigue monitoring program. The NRC staff's review of Ginna's evaluation of the pressurizer surge nozzle and the lower head is contained in section 4.3.2.7 of the SER. The Ginna UFSAR supplement for metal fatigue of the pressurizer (and Class 1 components) is provided in Section A3.3.1 of the License Renewal Application. The NRC staff has concluded that the UFSAR supplement provides an adequate description of the fatigue TLAA of ASME Class 1 components to satisfy 10CFR54.21 (d).

The LR SER conclusions on fatigue remain valid for the EPU. The following transients remain unchanged and are not affected by the power uprate:

Heatup and Cooldown Transients
Pressurizer Surge Transients

The pressurizer surge rate is a function of the volumetric expansion of the RCS coolant during power transients. The heatup/cooldown surge transients and the load/unload transients remain unchanged for the EPU conditions because the RCS pressure and pressurizer pressure remains unchanged for these transients. For the remainder of the design transients, the changes in pressures and temperatures have been determined to be bounded by the original equipment specification. Therefore, the effects of EPU do not impact the conclusions of the License Renewal SER.

Pressurizer and Supports Results

For the components at a normal operating pressure of 2250 psia, affected by T_{hot} (e.g., the surge nozzle), the temperature difference of 52.0°F was bounded by the original Equipment Specification. The same was true for those components at a normal operating pressure of 2250

psia, affected by T_{cold} (e.g., the spray nozzle); the temperature difference of 124.7°F was bounded by the original Equipment Specification.

Comparison of Change in Temperature Values

Parameter	ΔT (°F) Original Equip. Spec.	ΔT (°F) LR section 1.1, Nuclear Steam Supply System Parameters
T_{hot}	110	52.0
T_{cold}	125	124.7

Pressurizer and Supports Conclusions

The changes made to the design transients were either non-existent or insignificant. Based on a comparative study, it is concluded that the revised parameters associated with the EPU would not have any impact on the existing pressurizer stress and fatigue analyses. Also, the effects of EPU would not affect the surge line stratification analysis performed for the surge nozzle. Therefore, this evaluation concludes that the pressurizer will maintain its ability to function as part of the primary pressure boundary and that the pressurizer components will continue to meet the requirements of GDC-1, GDC-2, GDC- 4, GDC-14, GDC-15 and 10 CFR50.55a under the EPU conditions.

Additionally, portions of the pressurizer components are within the scope of License Renewal. The EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal boundaries. The changes associated with operating the pressurizer at EPU conditions (i.e., the pressurizer will be operated with a higher full power level and a lower hot zero power level) do not adversely affect the reactor coolant pressure boundary integrity. System component internal environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified a need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus, no new aging management effects are identified and no new commitments are required for Ginna for the pressurizer beyond those described in this report.

Based on the above, Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the pressurizer and supports. The aging evaluations

approved by the NRC in NUREG-1786 for the pressurizer and supports remain valid for the EPU conditions.

2.2.2.8 Conclusions for Pressure-Retaining Components and Component Supports

Ginna has reviewed the evaluations related to the structural integrity of pressure-retaining components and their supports and concludes that the effects of the proposed EPU on these components and their supports have been adequately addressed. Ginna further concludes that the pressure-retaining components and component supports will continue to meet the requirements of the Ginna Station's Current Licensing Basis with respect to 10CFR50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15 following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

2.2.3.1 Regulatory Evaluation

Reactor pressure (RPV) vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. Ginna Nuclear Power Plant, LLC (Ginna) reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. Ginna's review covered the analyses of flow-induced vibration (FIV) for safety-related and nonsafety-related reactor internal components, as well as the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. Ginna's review also included a comparison of the resulting stresses and cumulative fatigue usage factors against the corresponding Code-allowable limits. NRC's acceptance criteria were based on:

- 10CFR50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed

- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences

Specific review criteria are contained in the SRP, Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5 and guidance provided in Matrix 2 of RS-001.

Ginna Current Licensing Basis

As noted in *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predate those provided in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their

safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to:

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1. General Design Criterion 1 - Quality Standards and Records, wherein is noted that all SSCs of the facility were classified according to their importance. The classification of structures and equipment is discussed in UFSAR section 3.2. SSCs were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. Discussions of applicable codes and standards, quality assurance programs, test provisions, etc., are given in UFSAR Chapters 3, 4 and 7. The quality control and quality assurance program for Ginna Station construction, and the current quality assurance program, under which this system was installed and maintained, are described in UFSAR sections 17.1, and 17.2, respectively.

- GDC-2 is described in Ginna UFSAR section 3.1.2.1.2, General Design Criterion 2 - Design Bases for Protection Against Natural Phenomena. As described therein, Ginna Station, as participant in the Seismic Qualification Utility Group (SQUG), received NRC acceptance of its response to Generic Letter 87-02, including approval of the methodology in the SQUG Generic Implementation Procedure for use in verification of equipment seismic adequacy including equipment involved in future modifications and replacement equipment.

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. SEP review of this topic is described in this UFSAR section., Conformance to the requirements of GDC-4 is also described in the following:
    - Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
    - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)

- GDC-10 is described in Ginna UFSAR section 3.1.1.2.5, Reactor Containment, which states that containment structure shall be designed to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without the loss of required integrity.

The reactor pressure vessel internals and core supports structure analyses are discussed in detail in Ginna UFSAR sections 3.9.2.3.6, 3.9.2.5.1, 3.9.5 and 4.2.1.3.4.1, including replacement steam generator (RSG) and reduced T_{avg} Operation.

In addition to the evaluations described in the UFSAR, the Ginna Station's reactor internals components were evaluated for plant license renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant, May 2004. The reactor internals and core support structural components evaluated herein for the proposed EPU are discussed in the License Renewal SER in sections 3.0 and 3.1.

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor internals and core supports. The aging evaluations approved by the NRC in Section 4.3 of the License Renewal SER for the reactor internals and core supports remain valid for the EPU conditions. As approved by the NRC per Section 4.3 in the License Renewal SER, and as in *WCAP-14577 "License Renewal Evaluation: Aging Management Evaluation for Reactor Internals" Revision 1-A, dated March 2001,* the evaluations for the aging management performed for the reactor internals and core supports remain valid for the EPU conditions.

2.2.3.2 Technical Evaluation

2.2.3.2.1 Introduction

The RPV internal system consists of the reactor vessel, reactor internals, fuel, and control rod drive mechanisms (CRDMs). The reactor internals functional description is provided in the following text. The reactor internals are designed to withstand forces due to normal, upset, emergency, and faulted conditions.

Changes in the primary coolant system operating conditions (e.g., increase in power) also produce changes in the boundary conditions; this includes loads and temperatures experienced by the reactor internal components. Ultimately, this results in changes in the stress levels in these components and changes in the relative displacement between the reactor vessel and the reactor internals. To ensure that the reactor internal components still maintain their design functions, and to ensure safety questions have been reviewed, a systematic evaluation of the

reactor components has been performed to assess the impact of increased core power on the reactor internal components. The reactor internal core support structure is defined as follows:

Upper Core Support Assembly (consisting of the following components)

- Upper support plate/deep beam structure
- Upper core plate
- Upper core plate fuel pins
- Upper support columns

Lower Core Support Assembly (consisting of the following components)

- Lower support plate
- Lower core plate
- Lower core plate fuel pins
- Lower support columns
- Core barrel assembly
- Baffle-former assembly
- Radial keys and clevis insert assembly
- Upper core plate alignment pin

The internal structures are defined as all those other structures within the reactor vessel that are not core support structure, fuel assemblies, control assemblies, and instrumentation. These structures are attached to and supported by the core support structure.

Reactor Internals Functional Description

The reactor internals core support structure is within the confines of the reactor vessel. The function of the structure is to provide the direct support and restraint of the core, i.e., fuel assemblies. The core support, together with the internal structures, provides:

- The orientation of the reactor core
- The orientation, guidance, and protection of the reactor control rod assemblies
- A passageway for directional and metered control of the reactor coolant flow through the reactor core
- A passageway, support, and protection for any in-vessel/core instrumentation
- A secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core-barrel assembly

- Reactor vessel neutron shielding

Function of Core Support Structure

Upper Core Support Assembly

The upper core support assembly provides the vertical and lateral restraint and lateral alignment of the top of the core through its primary components (upper support plate/deep beam structure, support columns, and the upper core plate) and its interface with the reactor vessel. The assembly also provides the support for the internals structures, such as the instrumentation conduit and supports, and reactor control rod guide tubes.

The upper support subassembly, which is supported on its outer edges, transfers the loading of the upper core support assembly to the reactor vessel. Keyways, with customized inserts to maintain required gaps, are located in the outer edges of the subassembly to provide the upper-core-support-assembly to reactor vessel to lower-core-support assembly alignment, and to limit any transverse or rotational movement of the subassembly. There are penetrations through the subassembly for spray nozzles that allow limited flow into the reactor vessel upper head region.

The upper support columns transfer vertical and lateral loads to the upper support subassembly and support the upper core plate vertically. Guides are provided at the lower end of the upper columns for coolant flow.

The upper core plate, which is attached to the bottom of the upper support columns, forms the upper periphery of the core, transfers core loading to the support columns, and, when in place within the reactor vessel, rests on the fuel assembly springs causing the core preload. The plate is perforated to allow coolant flow while maintaining a uniform velocity profile. The underside of the upper core plate contains the upper fuel pins, which engage the top of the fuel assemblies. The upper-core-periphery to lower-core-periphery alignment is provided through keyways in the outer edges of the upper core plate that contain customized inserts that provide the required pin engagement gaps. In addition, the keyway/insert system limits any rotation or translation of the upper core plate.

Lower Core Support Assembly

The lower core support assembly is the major supporting assembly of the total structure. The assembly functions to:

- Support the core and the attached internal structures
- Transfer these and other design loadings to the reactor vessel
- Provide the restraint and alignment of the core
- Provide the directional and metered control of the reactor coolant flow through the core
- Provide neutron shielding for the reactor vessel

Fuel assemblies are placed into the core-barrel subassembly and rest on the lower core plate. The lower core plate, is supported on the lower core barrel ledge and by the lower support columns, and contains the lower fuel pins that provide location and alignment for the bottom of the fuel assemblies. The lower core plate is perforated to allow directional and metered control of flow of the reactor coolant and is attached to the core barrel and the flange, forming the core barrel subassembly. The function of the core barrel subassembly is to transmit the loading to the reactor vessel. This is accomplished at the top by the core-barrel flange, which rests on a ledge provided on the reactor vessel and limited loading is transmitted at the bottom by the radial support system.

The radial support system consists of keys that are attached to the lower end of the core-barrel subassembly on the lower support plate and that engage clevises provided in the reactor vessel. This system restricts the lower end of the core-barrel subassembly from rotational or tangential movement, but allows for radial thermal growth and axial displacement.

Inside the core barrel, above the lower support plate, is the baffle-former assembly. This assembly forms a radial periphery of the core and, through the dimensional control of the cavity, i.e., the gap between the fuel assemblies and baffle plates, provides directional and metered control of the reactor coolant through the core.

2.2.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The principal input parameters utilized in the analysis of the reactor internal components and core supports are the reactor coolant system (RCS) design parameters provided in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. For structural analysis/evaluations, the nuclear steam supply system (NSSS) design transients discussed in LR section 2.2.6, NSSS Design Transients, were considered. The fuel considered is a full core of Westinghouse 14x14 Vantage + fuel without intermediate flow mixing (IFM) grids with the thimble plugging devices removed.

Ginna has performed evaluations/analyses to assess the effect of operating at 1817 Mwt EPU on the reactor pressure vessel/internals system of the Ginna Station. The description of these analyses and evaluations are provided in the following sections.

Acceptance Criteria

- The design core bypass flow limit with the thimble plugging devices removed is 6.5% of the total vessel flow rate.
- The RCCA drop time Technical Specification of 1.8 seconds is to be maintained.

- For the structural and fatigue evaluations of core support components, the components stresses meet the allowable stress limits and the cumulative fatigue usage factors must be less than 1.0.

2.2.3.2.3 Description of Analyses and Evaluations

The reactor vessel internals have been analyzed for the Ginna Station EPU revised design parameters and the design basis load combinations. The analysis of the components was performed for the normal, upset, emergency and faulted conditions (LOCA/Seismic). The results of these analyses confirm that there is no adverse impact on the structural adequacy of the reactor internals components for the EPU conditions.

In addition to the evaluations described above, the Ginna Station's reactor internals components were evaluated for plant license renewal. System and system component materials of construction, operating history and programs used to manage aging effects are described and documented in NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant, May 2004.

Thermal-Hydraulic System Evaluations

System Pressure Losses

The principal RCS flow route through the RPV system at Ginna Station begins at the inlet nozzles. At this point, flow turns downward through the reactor vessel/core-barrel annulus. After passing through this downcomer region, the flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core support plate, and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the outlet nozzles. Note that the upper plenum region is occupied by support columns and rod cluster control assembly (RCCA) guide columns.

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, i.e., vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces. The pressure loss data is necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations. The hydraulic forces are considered in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold-down spring, and the stresses in the reactor vessel closure studs.

Thermal hydraulic evaluations were performed by solving the mass and energy balances for the reactor internals fluid system. These analyses determined the distribution of pressure and flow within the reactor vessel, internals, and the reactor core. Results were obtained with a full core

of Westinghouse 14x14 Vantage + fuel without IFM grids with the thimble plugging devices removed, and at RCS conditions, as given in LR section 1.1 Nuclear Steam Supply System Parameters, Table 1-1.

Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Variations in the size of some of the bypass flow paths, such as gaps at the outlet nozzles and the core cavity, occur during manufacturing or change due to fuel assembly changes. Plant-specific, as-built dimensions are used in order to demonstrate that the bypass flow limits are not violated. Therefore, analyses are performed to estimate core bypass flow values to either show that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow.

Fuel assembly hydraulic characteristics and system parameters, such as inlet temperature, reactor coolant pressure, and flow were used to determine the impact of EPU RCS conditions on the total core bypass flow. The results of this analysis calculated a core bypass flow value of 5.59% with the thimble plugging devices removed. Therefore, the design core bypass flow value of 6.5% with thimble plugging devices removed remains acceptable.

Hydraulic Lift Forces

An evaluation was performed to estimate hydraulic lift forces on the various reactor internal components for the EPU parameters shown in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. This is done to show that the reactor internals assembly would remain seated and stable for all conditions. Based on the evaluation performed for Ginna Station EPU, the reactor internals will remain seated and stable for the EPU RCS conditions. There is no significant impact on the currently analyzed core barrel ledge contact load of 2.498E+05 lbf compared to an allowable load of 1.0E+05 lbf for the reactor internals to remain seated.

Upper Head Fluid Temperatures

The average temperature of the primary coolant fluid that occupies the reactor vessel closure head volume is an important initial condition for certain dynamic LOCA analyses. Therefore, it was necessary to determine the upper head temperature when changes in the RCS conditions take place in the plant. Determination of upper head temperature stemmed from the THRIVE evaluations used to assess the core bypass flow. The THRIVE code models the interaction between all different flow paths into and out of the closure head region. Based on this interaction, it calculates the core bypass flow into the head region and average head fluid temperature for different flow path conditions. Ginna Station is configured such that the upper head region is at slightly below T_{hot}. These upper head temperatures were provided as inputs and were used in subsequent LOCA analyses.

RCCA Scram Performance Evaluation

The RCCAs represent perhaps the most critical interface between the fuel assemblies and the other internal components. It is imperative to show that the EPU RCS conditions will not adversely impact the operation of the RCCAs, either during accident conditions or normal operation.

The analysis performed determined the potential impact of the conditions shown in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1 on the limiting RCCA drop time. The maximum estimated RCCA drop time was calculated to be 1.42 seconds to the top of dashpot, which is still less than the current Technical Specification limit of 1.8 seconds.

Mechanical System Evaluations

LOCA Loads

To perform the RPV LOCA analyses of the Ginna Station, a finite element model of the RPV system was developed. The mathematical model of the RPV is a three-dimensional, nonlinear finite element model that represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN code, a general purpose, finite element computer code that has been used for this analysis since the original plant design. For the EPU at the Ginna Station, LOCA analyses were performed to generate core plate motions and the reactor vessel/internals interface loads. Because larger lines are to be excluded by leak-before-break methodology (see LR section 2.1.6), the largest branch lines considered in this EPU analysis were the 3-inch pressurizer spray line connected to the cold leg and the 4-inch upper plenum injection (UPI) line. Of these two breaks, the 3-inch pressurizer spray line break was found to be more limiting.

The results of LOCA reactor vessel displacements and the impact forces calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. The core plate motions for both breaks were used in fuel grid crush analysis and to confirm the structural integrity of the fuel as discussed in detail in LR section 2.8.1.2.3, Seismic/LOCA.

Seismic Analyses

The EPU does not impact the seismic response of the reactor internals; however, due to the changes in fuel assembly design, a nonlinear time history seismic analyses of the RPV system was performed. This analysis included the development of the system finite element model and the synthesized time history accelerations.

The results of the system seismic analysis include time history displacements and impact forces for all the major components. The reactor vessel seismic displacements and the impact forces

calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. The core plate motions were used in fuel grid crush analysis and to confirm the structural integrity of the fuel as discussed in detail in LR section 2.8.1.2.3, Seismic/LOCA.

Flow-Induced Vibrations

Flow-induced vibrations (FIV) of pressurized water reactor internals have been studied for a number of years. The objective of these studies is to show the structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale-model tests, as well as tests in fabricators' shops and bench tests of components, along with various analytical investigations. The results of these scale-model and in-plant tests indicate that the vibrational behavior of two-, three-, and four-loop plants is essentially similar, and the results obtained from each of the tests complement one another and make possible a better understanding of the FIV phenomena. Based on the analysis performed for the Ginna Station, reactor internals response due to FIV is extremely small and well within the allowable based on the high cycle endurance limit for the material. The results of FIV analyses due to EPU at Ginna Station are provided in Table 2.2.3-1 and Table 2.2.3-2.

Evaluation of Reactor Internal and Core Support Structure Components

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove heat generated within the structural components due to radiation (for example, gamma heating). The configuration of the internals provides adequate cooling capability. Also, the thermal gradients resulting from gamma heating and core coolant temperature changes are maintained below acceptable limits within and between the various structural components.

The Ginna Station reactor internals were designed and built prior to the implementation of Subsection NG of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and therefore, a plant-specific stress report on the reactor internals was not required. The structural integrity of the Ginna Station reactor internals design has been ensured by analyses performed on both generic and plant-specific bases to meet the intent of the ASME Code. These analyses were used as the basis for evaluating critical Ginna Station reactor internal components for EPU RCS conditions and revised NSSS design transients.

Structural evaluations demonstrate that the structural integrity of reactor internal components is not adversely affected either directly by the EPU RCS conditions and NSSS design transients, or by secondary effects on reactor thermal-hydraulic or structural performance. Heat generated

in reactor internal components, along with the various fluid temperature changes, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of the various components.

Component Analyses/Assessments

A series of evaluations/assessments for the Ginna Station were performed on reactor internal components for the EPU conditions. The most critical components that were evaluated are:

- Upper support plate/deep beam structure
- Upper core plate
- Upper core plate fuel pins
- Upper support column

- Lower support plate
- Lower core plate
- Lower support column
- Core barrel
- Thermal shield and flexures
- Radial keys and clevis insert assembly
- Baffle-former assembly

The results of these evaluations/assessments demonstrate that the above listed components are structurally adequate for the EPU conditions and the fatigue usage factors were less than 1.0. A summary of stresses versus allowable and corresponding fatigue usage factors is given in Table 2.2.3-3

Baffle-Barrel Bolt Evaluation

The bolts are evaluated for loads resulting from hydraulic pressure, seismic and LOCA loads, preload, and thermal conditions. The temperature difference between baffle and barrel produces the dominant loads on the baffle-former bolts and the loads that are most directly affected by the uprating. The EPU RCS conditions do not affect deadweight or preload forces and do not increase the seismic or LOCA loads on the bolts.

As noted in the Ginna UFSAR, the baffle-former bolts have been inspected in response to concerns raised in the Nuclear Regulatory Commission issued Information Notice (IN) 98-11. Based on the results of these inspections some bolts were replaced. The replaced bolts, which are considered less susceptible to the flaw initiating mechanism, were considered equivalent to the original bolts in the evaluation for the EPU RCS conditions. The number of bolts verified to be acceptable, together with those replaced, resulted in a current baffle bolt pattern with margin over that required by a 10-inch break as documented in WCAP-15036. Furthermore, LBB

analyses have been performed for all 10-inch RCS lines at Ginna Station. The next largest postulated break sizes are in the 2-, 3-, and 4-inch lines, which provides additional margin (see LR section 2.2.1).

The basis of the bolt qualification is a fatigue test. The evaluation of the revised loads consisted of demonstrating that the loads associated with the EPU are acceptable for the plant design life. Therefore, it is concluded that the baffle-former bolts are structurally adequate for the EPU RCS conditions.

This evaluation has determined that baffle-former bolts remain acceptable for the renewal license extended period of operation. The following summarizes the basis for this determination:

- Extended operation for a 60 year design life is acceptable based on the determination that 7,960 cycles of the plant loading and unloading design transient were qualified with 7,570 of these cycles at the EPU RCS conditions. This loading controls the fatigue life of the baffle-former bolts. All other design transients, except plant loading and unloading, listed in the UFSAR, Table 5.1-4, or generated for the EPU RCS conditions were qualified for the specified number of cycles. Per NUREG-1786, Ginna uses its Fatigue Monitoring Program to provide assurance that the number of design cycles will not be exceeded during the period of extended operation. Ginna's Fatigue Monitoring Program provides an acceptable program for monitoring the fatigue usage of RCS components, in accordance with the requirements of 10CFR54.21(c)(1)(iii).
- Evaluation of environmental fatigue effects is not required for the baffle-former bolts as they are not included in the components which require such evaluations in NUREG-1786.
- Aging degradation issues for the Ginna Reactor Systems Components are listed in Table 3.1-1 of NUREG-1786. The baffle-former bolts are addressed as follows.
    - Per Section 3.1.2.2.3, loss of fracture toughness due to neutron embrittlement and void swelling could occur in baffle-former bolts.
    - Section 3.1.2.2.8 of NUREG-1786 discusses the crack initiation and growth due to stress corrosion cracking or irradiation-assisted stress-corrosion cracking could occur.
    - Section 3.1.2.2.9 of NUREG-1786 discusses loss of preload due to stress relaxation.

It is stated the programs described in NUREG-1768 provide reasonable assurance that the aging effect will be adequately managed during the period of extended operation. Those programs remain effective methods of managing this issue considering the EPU RCS conditions.

2.2.3.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor vessel internal components. This LR section addresses the maximum stress intensity ranges and cumulative fatigue damage for critical reactor vessel internal components considering the impact of EPU conditions on license renewal and evaluates those ranges and fatigue damage against the ASME code limits. Stress corrosion cracking (SCC) of RV internals components is addressed in LR section 2.1.5, Reactor Coolant Pressure Boundary Materials.

The evaluations (summarized in this section) of maximum stress intensity ranges and cumulative fatigue usage factors for the components of the reactor vessel internals, considering EPU conditions, show that the reactor vessel components continue to meet the ASME acceptable limits. Since the original 40-year design transient set has been shown to be bounding for 60 years of operation based on the finding that the number of original design cycles bounds the actual plant cycles, and the number of design cycles for the EPU has not changed from the original 40-year transient set, the fatigue evaluations of the reactor vessel internals components are valid for 60 years of operation.

Finally, the current ASME Section XI Inservice Inspection Program is considered to provide reasonable assurance that aging effects will be managed such that the intended functions of reactor vessel internal (RVI) components will be maintained during the license renewal period. The NRC staff concluded that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of structure and components subject to an aging management review such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the current licensing basis, as required by 10CFR54.29 (a).

In addition, as part of the License Renewal efforts, Ginna Station has committed to submit a reactor vessel internals inspection program by September, 2007. The reactor internals inspection program is anticipated to be modeled in accordance with current efforts now being developed by the industry through the EPRI Materials Reliability Program (MRP) Issues Task Group (ITG).

Ginna has evaluated the impact of the EPU on the conclusions reached in the Ginna License Renewal Application for the reactor vessel internals. The aging evaluations approved by the NRC in NUREG-1786 for the reactor vessel internal components remain valid for EPU conditions.

2.2.3.2.5 RPV Internals and Core Supports Results

Analyses have been performed to assess the effect of changes due to EPU at Ginna Station. The various results reached are as follows:

- The design core bypass flow value of 6.5% of the total vessel flow with thimble plugging devices removed is maintained for the EPU conditions.
- An RCCA performance evaluation was completed and the results indicated that the current 1.8-second RCCA drop-time-to-dashpot entry limit (from gripper release of the drive rod) is satisfied at the EPU conditions.
- Evaluations of the critical reactor internal components were performed, which indicated that the structural integrity of the reactor internals is maintained at the EPU conditions and the cumulative fatigue usage factors were all shown to be less than 1.0

The results of component structural analyses are summarized in Table 2.2.3-3.

2.2.3.3 Conclusion

Ginna has reviewed the evaluations related to the structural integrity of reactor internals and core supports and concludes that the evaluations have adequately addressed the effects of the proposed EPU on the reactor internals and core supports. Ginna further concludes that the evaluations have demonstrated that the reactor internals and core supports continue to meet the Ginna Station current licensing basis requirements with respect to GDC-1, GDC-2, GDC-4, and GDC-10 following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

Table 2.2.3-1
Lower Internal Critical Component Stresses Due to FIV

Component	Maximum Alternating Stress psi	ASME Code Endurance Limit(1) (high-cycle fatigue) psi
Top Support Bolts	[ ]a,c	23,700
Flexures	[ ]a,c	23,700

1. Basis is ASME Code section NB-3222 and Figure I-9.2.2, Curve A and Table I-9.2.2.

Table 2.2.3-2
Upper Internal Critical Component Strains Due to FIV

Component	Maximum Mean Strain in./in. x 10^{-6}	ACCEPTABLE(1) Mean Strain in./in. x 10^{-6}	Maximum Alternating Dynamic Strain in./in. x 10^{-6}	ACCEPTABLE(1) Alternating Dynamic Strain in./in. x 10^{-6}
Guide Tubes with Core	[ ]a,c	266.0	[ ]a,c	± 65.0
Guide Tubes without Core	[ ]a,c	266.0	[ ]a,c	± 65.0

1. Basis of acceptance is from measured strain data.

Table 2.2.3-3
Reactor Internal Components Stresses and Fatigue Usage Factors

Component	Stress Intensity (ksi) S.I. = ($P_m + P_b + Q$)	Allowable S.I. (3 S_m) ksi	Fatigue Usage
Upper Support Plate	[ ]a,c	49.2	[ ]a,c
Deep Beam Structure		49.2	[ ]a,c
Upper Core Plate	[ ]a,c	48.6	[ ]a,c
Upper Core Plate Alignment Pins	[ ]a,c	34.44(b)	[ ]a,c
Upper Support Columns	[ ]a,c	49.2	[ ]a,c
Lower Support Plate	[ ]a,c	49.2	[ ]a,c
Lower Core Plate	[ ]a,c	49.2	[ ]a,c
Lower Support Columns	[ ]a,c	49.2	[ ]a,c
Core Barrel Assembly:			
Upper Girth Weld	[ ]a,c	49.2	[ ]a,c
Lower Girth Weld	[ ]a,c	49.2	[ ]a,c
Outlet Nozzle	[ ]a,c)	34.44(b)	[ ]a,c
Thermal Shield & Flexures			
Thermal Shield Flexures	[ ]a,c	49.2	[ ]a,c
Thermal Shield Flexure Bolts	[ ]a,c	57.0	[ ]a,c
Radial Keys and Clevis Insert Assembly			
Lower Radial Inserts	[ ]a,c	69.0	[ ]a,c

Notes:
[a.
]a,c
b. Allowable based on weld quality factor

2.2.4 Safety-Related Valves and Pumps

2.2.4.1 Regulatory Evaluation

Ginna Nuclear Power Plant, LLC (Ginna) reviewed safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME Boiler & Pressure Vessel Code, within the scope of Section XI of the ASME Boiler & Pressure Vessel Code, and the ASME Operations and Maintenance Code, as applicable. The Ginna review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The Ginna review also covered any impacts that the proposed EPU may have on the licensee's motor-operated valve program related to Generic Letter (GL) 89-10, GL 96-05, and GL 95-07. Lessons learned from the motor-operated valve program and the application of those lessons learned to other safety-related power-operated valves were evaluated. The NRC's acceptance criteria are based on:

- General Design Criterion 1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed

- General Design Criterion 37, General Design Criterion 40, General Design Criterion 43, and General Design Criterion 46, insofar as they require that the emergency core cooling system, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components

- General Design Criterion 54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits

- 10CFR50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the in-service testing program requirements identified in that section

Specific review criteria are contained in Standard Review Plan, Sections 3.9.3 and 3.9.6, and other guidance provided in Matrix 2 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then

current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to:

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1, General Design Criterion 1 – Quality Standards and Records. As described in this UFSAR section, safety-related systems and components were designed, fabricated, inspected, and erected to the applicable provisions of the then-recognized codes, good nuclear practice, and the quality standards that reflected their importance. As part of the SEP, the original codes and standards used in the design of systems and components were compared with the later licensing criteria based on Regulatory Guide 1.26 and 10CFR50.55a. The results of this review are documented in Ginna UFSAR Table 3.2-1, Classification of Structures, Systems, and Components."

- GDC–37 is described in Ginna UFSAR section 3.1.2.4.8, General Design Criterion 37 – Testing of Emergency Core Cooling Systems. As described in this UFSAR section, components of the emergency core cooling system located outside containment are accessible for leak tightness inspection during periodic tests. All of the pumps in the emergency core cooling system are started at intervals as specified in the Inservice Testing Program. Valve operability as well as system operability tests are performed during refueling shutdowns to demonstrate proper automatic operation of the emergency core cooling system. The required surveillance tests are described in the Technical Specifications.

- GDC–40 is described in Ginna UFSAR section 3.1.2.4.11, General Design Criterion 40 - Testing of Containment Heat Removal System. As described in this UFSAR section, design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the containment spray system. Permanent test lines for the containment spray loops are located so that all components up to the isolation valves at the spray nozzles may be tested. These isolation valves are checked separately. The required periodic tests are described in the Technical Specifications.

- GDC-43 is described in Ginna UFSAR section 3.1.2.4.14, General Design Criterion 43 - Testing of Containment Atmosphere Cleanup Systems. As described in this UFSAR section, the containment atmosphere cleanup systems are tested as described under GDC-40, above.

- GDC-46 is described in Ginna UFSAR section 3.1.2.4.16, General Design Criterion 46 - Testing of Cooling Water Systems. As described in this UFSAR section, redundancy and isolation are provided to allow periodic pressure and functional testing of the systems as a whole. One of the redundant pumps in the component cooling water system is in service during MODES 1 and 2. During plant operation, three service water pumps are in operation.

- GDC – 54 is described in Ginna UFSAR section 3.1.2.5.5, General Design Criterion 54 – Piping Systems Penetrating Containment. As described in this UFSAR section, piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections as necessary to allow periodic leak detection to be performed.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's systems and components have been evaluated for plant license renewal. Plant system and component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The safety-related valves and pumps are addressed within the SER under the systems that contain them. In-service testing of safety-related valves and pumps was evaluated as part of the Periodic Surveillance and Preventive Maintenance Program, as described in section 3.0.3.8 of the above SER.

Performance / Inservice Testing of Safety-Related Pumps and Valves

As addressed in Ginna UFSAR section 3.9.6, the Inservice Pump and Valve Testing Program includes quality group A, B, and C pumps which are provided with an emergency power source, and those quality group A, B, and C valves which are required to shut down the reactor or to mitigate the consequences of an accident and maintain the reactor in a safe shutdown condition. Quality groups A, B, and C components correspond to those defined in Regulatory Guide 1.26, Rev. 3, February 1, 1976.

As addressed in Ginna UFSAR section 3.9.6, the program has been developed as required by 10CFR50.55a, following the guidance of ASME Boiler and Pressure Vessel Code, Section XI (herein referred to as "ASME Section XI"). The program follows the guidance of Generic Letter 89-04, with possible exceptions approved by the NRC. The Ginna Inservice Pump and Valve Testing Program is controlled by the Ginna Station Quality Assurance Program for Station Operation.

As addressed in Ginna UFSAR section 3.9.6, the Inservice Pump and Valve Testing Program is considered part of the pump and valve surveillance program required by the Technical Specifications. Technical Specification requirements associated with pump and valve surveillance will continue to be implemented as specified. When changes to Technical Specifications create conflicts with the program, the revised Technical Specifications will provide guidance until the program is revised to incorporate the changes. When a valve, pump, or its control system has been replaced or repaired, or has undergone maintenance that could affect its performance, and prior to the time it is returned to service, it will be tested as necessary to demonstrate that the performance parameters which could have been affected by the replacement, repair, or maintenance are within acceptable limits.

Ginna Station Technical Specification 5.5.7, "Inservice Testing Program," states that the Inservice Testing Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components and that the program shall include testing frequencies specified in ASME Section XI.

Motor-Operated Valve Program

Generic Letters 89-10 and 96-05

As addressed in Ginna UFSAR section 5.4.9.3, the Ginna Station motor-operated valve program was established in response to IE Bulletin 85-03. The program was later expanded to address the recommendations of GL 89-10 and Generic Letter 96-05 to include all motor-operated valves in safety-related systems that are not blocked from inadvertent operation from either the control room, motor control center, or the valve itself. Motor operated valves within the following safety-related systems / portions of systems are included in the program:

- High head safety injection – injection and recirculation modes
- Low head safety injection (Residual heat removal) – injection and recirculation modes
- Containment spray
- Chemical & volume control
- Reactor coolant
- Auxiliary feedwater
- Standby auxiliary feedwater
- Component cooling water – safety injection and residual heat removal pump cooling; sump recirculation cooling
- Service water – non-essential load isolation
- Main steam system
- Main feedwater system

As addressed in Ginna UFSAR section 5.4.9.3, the motor-operated valves in the above systems are tested at design pressure when practicable; otherwise, alternative methods are used to ensure motor-operated valve operability. The motor-operated valve program is described in the Ginna Station Motor-Operated Valve Program Plan. The motor-operated valve program is used to establish torque switch and limit switch settings for safety-related AC and DC motor-operated valves, and to demonstrate valve operability during normal and abnormal design-basis events. The program also includes periodic and post-maintenance and repair testing to verify continued valve operability. This program includes periodic verification of motor-operated valve capability and trending of motor-operated valve problems. The motor-operated valve program and station procedures are designed to ensure that the switch settings of the motor-operated valves in the program are selected, set, and maintained correctly to accommodate the maximum differential pressures expected across the valves during both normal and abnormal design-basis events throughout the life of the plant. The NRC closed out its review of Ginna's GL 89-10 Program in a letter dated June 12, 1998 (Reference: Letter from E. M. Kelly, NRC, to R. C. Mecredy, RG&E, "NRC Motor Operated Valve Inspection 50-244/98-06," June 12, 1998). In response to GL 96-05, the program was enhanced to include provisions for continually monitoring valve performance for degradation and periodic verification of program effectiveness. In a Safety

Evaluation dated December 27, 1999 (Reference: Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, "Safety Evaluation Regarding the Licensee's Response to Generic Letter 96-05 (TAC No. M97050)," December 27, 1999), the NRC stated that Ginna has established an acceptable program to verify periodically the design–basis capability of all safety-related motor-operated valves, and is adequately addressing the actions required in GL 96-05.

Generic Letter 95-07

As addressed in Ginna UFSAR section 5.4.9.3, in response to Generic Letter 95-07, Ginna considered the safety-related motor-operated gate valves, including all valves within the GL 89-10 Program that could be potentially susceptible to the phenomenon of pressure locking and thermal binding, and performed assessments, analyses, or identified previous valve modifications to justify continued operability of the valves. The assessments of each valve were based upon the operational configurations and conditions imposed.

As addressed in Ginna UFSAR section 5.4.9.3:

- Valves 852A and 852B (RHR loop inlet valves to the reactor vessel) were modified in 1999 with flexible wedges that have vent holes to eliminate the potential for pressure locking.
- Valves 515 and 516 (pressurizer PORV isolation valves) were modified in 1989 with upstream discs that have vent holes to eliminate the potential for pressure locking.
- Valves 850A and 850B (RHR pump suction valves from containment sump B) were modified in 1970 to include bonnet vents to the RHR suction side of the valves to eliminate the potential for pressure locking.
- Valves 857A, 857B, and 857C (RHR discharge valves to SI / CS pump suctions) were modified in 1996 to install a bonnet pressure relieving hole in the designated valve disc, relieving pressure to the RHR side of the valves.
- For applicable valves, an analysis was performed to show that the MOV actuator has sufficient thrust to overcome the pressure-locked bonnet condition.
- For all other valves determined to be susceptible to pressure locking / thermal binding, the valves were justified based on valve design or upon the operational configuration and conditions imposed.

In a Safety Evaluation dated July 19, 1999 (Reference: Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, "Safety Evaluation of Licensee's Response to Generic Letter 95-07 (TAC No. M93466)," July 19, 1999), the NRC stated that Ginna had adequately addressed the actions requested in GL 95-07.

2.2.4.2 Technical Evaluation

2.2.4.2.1 Introduction

Performance / Inservice Testing of Safety-Related Pumps and Valves

Paragraph (f) of 10CFR50.55a requires that Inservice Testing Programs be updated at ten year intervals to comply with the latest NRC approved edition and addenda of the ASME Code incorporated by reference in 10CFR50.55a, Paragraph (b), 12 months prior to the start of the interval. The fourth ten year test interval for Ginna Station commenced January 1, 2000. The Code of Record for the fourth ten year test interval is the 1989 Edition of ASME Section XI. This edition refers directly to the ASME/ANSI OMa-1988 standard (Reference: ASME/ANSI OMa-1988, "Operation and Maintenance of Nuclear Power Plants") for pump testing (Part 6) and for valve testing (Part 10). The ASME OM Code-1998 Appendix I (Reference: ASME OM Code-1998, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants") is utilized for relief device testing.

Ginna defines those pumps and valves which fall within the scope of ASME Section XI as those in systems or portions of systems which are required to accomplish the following functions:

- Shutdown the reactor to Hot Shutdown / maintain Hot Shutdown condition (reactivity control, core heat removal, steam generator heat removal, pressure inventory and control, reactor coolant system pressure and inventory control)

- Mitigate the consequences of an accident (reactivity control, core heat removal, reactor coolant system integrity, containment integrity)

- Provide overpressure protection or vacuum relief for systems or portions of systems required to perform the above-described functions

The Ginna Inservice Testing Program includes pumps and valves required to perform the above-identified safety functions for the following systems:

- Main steam system
- Auxiliary feedwater system
- Standby auxiliary feedwater system
- Main feedwater system
- Component cooling water system
- Service water system
- Chemical and volume control system
- Residual heat removal system
- Safety injection system
- Containment spray system
- Reactor coolant system
- Containment isolation system
- Spent fuel pool cooling system

Note: The spent fuel pool cooling system is not considered necessary to mitigate the consequences of an accident. However, Ginna considers this system important to safety during refueling operations and full core offload situations. Therefore, this system is included in the Inservice Testing Program.

Some pumps and valves that do not fall within the scope of ASME Section XI may be included in the Inservice Testing Program as "augmented components" to meet other Ginna commitments or from a good engineering practice standpoint. These components need not be tested to, nor comply with, specific ASME Section XI requirements.

A design analysis determines safeguards flow rates to prove open direction valve operability of safety-related check valves which, per ASME Section XI, require open position verification testing, as implemented by the Inservice Testing Program.

Motor-Operated Valve Program

The Ginna Motor-Operated Valve Program implements the actions required to comply with the recommendations of GL 89-10 for safety-related motor-operated valves which meet the GL 89-10 selection criteria. Ginna has performed a design analysis which screens motor-operated valves to identify which valves should be included within the scope of the Motor-Operated Valve Program and which documents the "design basis" operating conditions under which these valves must perform their safety-related functions. Of the total motor-operated valves in the plant, 60 safety-related motor-operated valves have been determined to meet the GL 89-10 screening criteria. For each GL 89-10 motor-operated valve, the design analysis contains an operating basis evaluation, in which the following parameters are determined / documented:

- Maximum opening / closing upstream pressure / line pressure
- Minimum opening / closing downstream pressure
- Maximum opening / closing differential pressure
- Fluid flow rate in the full open condition
- Fluid temperature

The resulting line pressures / differential pressures from the motor-operated valve operating basis evaluations are used as inputs in the calculations which determine motor-operated valve thrust and torque values. The flow rates documented in the evaluation Operating Basis Summary are not used as inputs in the calculations which determine motor-operated valve thrust and torque values. The fluid temperatures documented in the evaluation Operating Basis Summary are not used as direct inputs in the calculations which determine motor-operated valve thrust and torque values; however, in the SW system operating basis evaluations the documented fluid temperature is used as an input in the determination of motor-operated valve differential pressure.

Motor-operated valve AC / DC motor de-rate factors are a function of (1) degraded voltage loss, and (2) temperature loss. Design analyses have been performed which determine the worst case degraded voltage at terminals of GL 89-10 motor-operated valves that are powered from

motor control centers and from the 125V DC system. The motor-operated valve motor de-rate factors are evaluated in the motor-operated valve thrust / torque calculations.

The maximum ambient temperature for each GL 89-10 motor-operated valve used in the motor-derate evaluations are identified in a Ginna design analysis. These temperatures are based on temperatures documented in Ginna UFSAR Table 3.11-1, "Environmental Service Conditions for Equipment Designed to Mitigate Design-Basis Events.

In response to Generic Letter 95-07, evaluations which determine the susceptibility of gate valves to pressure locking and thermal binding at current plant conditions were performed and documented in a Technical Report (Ginna Station MOV Program Pressure Locking/Thermal Binding Screening Technical Report, 94108-TR-01, Rev. 2, June 1997). The Technical Report identifies those gate valves susceptible to pressure locking and for which analysis was required. Calculations were performed to show that for these valves the motor-operated valve actuator has sufficient thrust to overcome the pressure-locked bonnet condition.

Ginna has implemented a motor-operated valve periodic verification program in response to the requirements of Generic Letter 96-05. The schedule for periodic verification testing is a function of motor-operated valve actuator margin and the risk significance of the motor-operated valve.

2.2.4.2.2 Description of Analyses and Evaluations

Performance / Inservice Testing of Safety-Related Pumps and Valves

The following addresses the impact of the EPU on the performance requirements of Ginna safety-related pumps and valves in Nuclear Steam Supply System systems and applicable balance of plant systems, including impact on Inservice Testing Program / Technical Specification requirements:

Nuclear Steam Supply System (NSSS) Systems

The NSSS systems include the reactor coolant system, chemical & volume control system, safety injection system, residual heat removal system, and containment spray system. Evaluations show that the EPU has no / negligible impact on system operating pressures, flow rates, and pump head performance for NSSS systems under normal operating conditions (refer to LR section 2.1.11, Chemical and Volume Control System, 2.6.5, Containment Heat Removal, and 2.8.4.4, Residual Heat Removal System). Accident and transient analyses confirm that safety-related pumps and valves in NSSS systems will continue to meet their performance requirements at EPU conditions (refer to LR section 2.8.5, Accident and Transient Analyses). Accordingly, the EPU has no impact on the performance characteristics and Inservice Testing Program / Technical Specification requirements for safety-related pumps and valves in the NSSS systems.

Balance of Plant (BOP) Systems

Main Steam System

Safety-related valves in the main steam system include the main steam isolation valves, main steam non-return check valves, main steam safety valves, main steam atmospheric relief valves, the turbine driven auxiliary feedwater pump steam admission valves, and the check valves in the steam inlet lines to the turbine driven auxiliary feedwater pump (refer to LR section 2.5.5.1, Main Steam).

- The main steam isolation valves are reverse-mounted swing-disc check valves, with the disc held open against the flow of steam by an air cylinder. The EPU does not affect the Technical Specification requirement for these valves to close in less than or equal to 5 seconds under no flow and no load conditions. The increased steam flow under EPU conditions will enhance the closing of these valves, thus ensuring that the Inservice Testing Program / Technical Specification required closing time is met. There is no minimum closing time requirement for these valves.

- The main steam non-return check valves are free swinging gravity closing type check valves, which protect the main steam header against reverse flow from one steam generator to another in event of a steam line rupture. The EPU does not affect the Technical Specification requirement for these valves to close. Because these valves protect the main steam header against reverse flow, the ability of the valves to close is not affected by the increased steam flow at EPU conditions. The EPU does not create a condition whereby the valves must operate against different forces than are currently applied. Accordingly, the EPU does not affect the Inservice Testing Program / Technical Specification performance criteria associated with these valves.

- The setpoints of the main steam safety valves are based on the design pressure of the steam generators. Since the design pressure of the steam generators has not changed for the EPU, there is no need to change the setpoints of the safety valves. Evaluation shows that the existing main steam safety valve capacities are acceptable for operation under EPU conditions (refer to LR section 2.5.5.1, Main Steam). Accordingly, Inservice Testing Program / Technical Specification requirements for these valves are not affected by the EPU.

- To limit the frequency of main steam safety valve operation, the set pressure of the atmospheric relief valves is based on plant no-load conditions and the lowest set pressure of the main steam safety valves. Since neither the no-load steam pressure nor the lowest main steam safety valve setpoint pressure is changing for the EPU, there is no need to change the atmospheric relief valve setpoint. The existing capacity of the main steam atmospheric relief valves is adequate to support operation under EPU conditions (refer to LR section 2.5.5.1, Main Steam). The Inservice Testing Program / Technical Specification test / operability requirements for these valves, which include periodic cycling of the valves, are not affected by the EPU.

- The turbine driven auxiliary feedwater pump steam admission valves are motor-operated valves. The maximum opening / closing differential pressures for these valves, which form the basis for the valve thrust / torque values, are not affected by the EPU (refer to Table 2.2.4-1). The Inservice Testing Program includes testing of the open / closed stroke times of these valves; the parameters affecting MOV stroke time (e.g., motor rpm, overall gear ratio of the actuator) are not affected by the EPU. Accordingly, the EPU

does not affect the performance characteristics and Inservice Testing Program requirements for these valves.

- The check valves in the steam inlet lines to the turbine driven auxiliary feedwater pump function to isolate one steam generator from the other in the event of a main steam line break. This performance requirement is not changed by the EPU. The EPU does not create a condition whereby the valves must operate against different forces than are currently applied. Accordingly, the EPU does not affect the performance characteristics and Inservice Testing Program requirements for these valves.

Main Feedwater System

Safety-related valves in the main feedwater system include the main feedwater regulating valves, main feedwater bypass valves, and the main feedwater check valves (refer to LR section 2.5.5.4, Condensate and Feedwater). As discussed below, in support of the EPU, the isolation function of the existing main feedwater pump discharge valves will be replaced by safety-related main feedwater isolation valves.

- In support of the EPU, a modification to existing manual main feedwater isolation valves to provide faster-acting feedwater system isolation in the event of failure of the existing main feedwater regulating valves to close in the event of a steam line break will be implemented. A Ginna License Amendment Request, dated April 29, 2005, would revise the Technical Specification 3.7.3 to allow use of main feedwater isolation valves in lieu of the existing main feedwater pump discharge valves to provide the capability to isolate the steam generators in event of a steam line break. Use of the modified main feedwater isolation valves provides a reduction in valve closure time as well as a reduction in the water volume downstream of the valves. Inservice Testing Program requirements for the main feedwater isolation valves will be developed as part of the plant change process.

- Due to the increase in main feedwater system flow at EPU conditions, a modification to the main feedwater regulating valves to allow for proper flow control of these valves will be implemented (refer to LR section 2.5.5.4, Condensate and Feedwater). The EPU does not affect the Technical Specification requirement for these valves to close in less than or equal to 10 seconds. The design specification associated with the main feedwater regulating valve modification includes the requirement that the modified valves close in less than or equal to 10 seconds. Any required changes to Inservice Testing Program requirements for the modified main feedwater regulating valves will be developed as part of the plant change process.

- The main feedwater bypass valves are used at low power levels (e.g., plant startup) to prevent erosion damage to the main feedwater regulating valves. The EPU does not affect the Technical Specification requirement for these valves to close in less than or equal to 10 seconds. As addressed in LR section 2.5.5.4, Condensate and Feedwater, the main feedwater bypass valves will experience a lower differential pressure at EPU conditions versus current conditions. Therefore, the Technical Specification / Inservice Testing Program closure time requirements for these valves will continue to be met at EPU conditions.

- The main feedwater check valves function as main feedwater line containment isolation valves, and also function to prevent leakage of water from the auxiliary feedwater

system into the main feedwater system. The EPU does not create a condition whereby the valves must operate against different forces than are currently applied. Accordingly, the EPU does not affect the functionality of these valves.

Preferred Auxiliary Feedwater System

As addressed in LR section 2.5.4.5, Auxiliary Feedwater, the auxiliary feedwater pump flowrate requirements for normal, transient, and accident conditions do not change at EPU conditions, and therefore remain within the design capacities of the motor-driven and turbine-driven auxiliary feedwater pumps (200 gpm and 400 gpm, respectively). Accordingly, the EPU has no impact on Inservice Testing Program / Technical Specification requirements for safety-related pumps and valves in the auxiliary feedwater system.

Standby Auxiliary Feedwater System

Safety-related standby auxiliary feedwater system components included in the Inservice Testing Program are the standby auxiliary feedwater pumps, standby auxiliary feedwater pump suction motor-operated valves, standby auxiliary feedwater pump discharge motor-operated valves, standby auxiliary feedwater pump cross-tie motor-operated valves, standby auxiliary feedwater pump isolation motor-operated valves, standby auxiliary feedwater pump "D" emergency isolation motor-operated valve, and check valves in the standby auxiliary feedwater pump suction and discharge lines.

As addressed in LR section 2.5.4.5, Auxiliary Feedwater, the required standby auxiliary feedwater accident analysis flow rate will increase from 200 gpm at current conditions to 235 gpm at EPU conditions. The Inservice Testing Program analysis / procedures for the standby auxiliary feedwater pumps will be revised to address testing the standby auxiliary feedwater pumps at EPU flow / head conditions.

The maximum opening / closing differential pressures for the standby auxiliary feedwater system motor-operated valves, which form the basis for the motor-operated valve thrust / torque values, are not affected by the EPU (refer to Generic Letter 89-10 evaluations below). The Inservice Testing Program includes testing of the open / closed stroke times of these valves; the parameters affecting MOV stroke time (e.g., motor rpm, overall gear ratio of the actuator) are not affected by the EPU. Accordingly, the EPU does not affect the performance characteristics or Technical Specification / Inservice Testing Program requirements for these valves.

As addressed in the design analysis which determines check valve safeguards flow rates, the check valves in the standby auxiliary feedwater pump suction and discharge lines have an open position safety function to pass 200 gpm from the standby auxiliary feedwater pumps to the steam generators. The Inservice Testing Program analysis / procedures for the standby auxiliary feedwater pump suction and discharge check valves will be required to be revised to address testing the standby auxiliary feedwater pump suction and discharge check valves at EPU flow conditions.

Service Water System

As addressed in LR section 2.5.4.2, Service Water, the EPU does not affect the existing service water system flow rates or system operating pressures. The maximum opening / closing differential pressures for the service water system motor-operated valves, which form the basis for the motor-operated valve thrust / torque values, are not affected by the EPU (refer to

Generic Letter 89-10 evaluations below). Accordingly, the EPU has no impact on Technical Specification / Inservice Testing Program requirements for safety-related pumps and valves in the service water system.

Component Cooling Water System

As addressed in LR section 2.5.4.3, Component Cooling Water System, there are no changes in component cooling water system flow rates or operating pressures for normal operating, plant cooldown, or post-accident conditions for the EPU. The maximum opening / closing differential pressures for the component cooling water system motor-operated valves, which form the basis for the motor-operated valve thrust / torque values, are not affected by the EPU (refer to Generic Letter 89-10 evaluations below). Accordingly, the EPU has no impact on Inservice Testing Program / Technical Specification requirements for safety-related pumps and valves in the component cooling water system.

Spent Fuel Pool Cooling and Cleanup System

Spent fuel pool cooling and cleanup system components included in the Inservice Testing Program include the spent fuel pool recirculation pumps and the check valves in the spent fuel recirculation pumps discharge piping.

As addressed in LR section 2.5.4.1, Spent Fuel Cooling and Cleanup System, the current spent fuel pool cooling flow rate provides acceptable heat removal at EPU conditions. As addressed in the design analysis which determines check valve safeguards flow rates, the check valves have a function in the open direction to pass 610 gpm / 1200 gpm cooling flow for cooling of the spent fuel pool. Since there are no changes in spent fuel pool cooling system flowrates for EPU conditions, the EPU does not affect Inservice Testing Program requirements for the spent fuel recirculation pumps and discharge check valves.

Containment Isolation System

The EPU does not affect the Inservice Testing Program / Technical Specification requirements for containment isolation valve leakage rate testing. The tests which measure containment isolation valve leakage rates (Type C tests) are performed at the peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a. Per the Technical Specifications, the value of P_a is 60 psig. As addressed in LR section 2.3.1, Environmental Qualification of Electrical Equipment, the limiting calculated peak containment pressure resulting from a loss-of-coolant accident at EPU conditions is less than 60 psig. Accordingly, no change is required in the containment leakage test pressure at EPU conditions.

Motor-Operated Valve Program

Generic Letter 89-10

The impact of the EPU on motor-operated valve operating parameters of GL 89-10 motor-operated valves in the following balance-of-plant systems was evaluated:

- Main steam system
- Component cooling water system
- Service water system

- Preferred auxiliary feedwater system
- Standby auxiliary feedwater system

In order to show the evaluation process used, Table 2.2.4-1 shows the evaluations of several valves in balance-of-plant systems. The results of the evaluations show that the EPU does not affect the maximum differential pressures / line pressures determined in the design analysis of GL 89-10 motor-operated valves in these systems, and therefore does not affect the calculations which determine motor-operated valve thrust and torque values.

As discussed in section 2.2.4.2.1 above, although not used as inputs in the calculations for thrust and torque values, flow rates are documented in the motor-operated valve operating condition evaluations. Evaluation of the impact of the EPU on this parameter notes that the standby auxiliary feedwater system accident analysis flow rate increases from 200 gpm at current conditions to 235 gpm at EPU conditions.

As discussed in section 2.2.4.2.1 above, although the documented fluid temperatures are not used as direct inputs in the calculations which determine motor-operated valve thrust and torque values, in the SW system Operating Basis Evaluations, the documented fluid temperature is used as an input in the determination of motor-operated valve differential pressure. The EPU does not affect the SW temperature used in the determination of SW motor-operated valve differential pressure.

The maximum fluid temperature for the preferred auxiliary feedwater system and standby auxiliary feedwater system increases from 100^{0}F to 104^{0}F at EPU conditions (refer to Table 2.2.4-1). This fluid temperature is not used as an input in the determination of motor-operated valve differential pressures in these systems and, therefore, does not affect the results of these analyses.

The impact of the EPU on motor-operated valve operating parameters of GL 89-10 motor-operated valves in the following NSSS systems was evaluated:

- Chemical & volume control system
- Reactor coolant system
- Residual heat removal system
- Safety injection system
- Containment spray system

Table 2.2.4-2 shows the evaluations of several valves in NSSS systems. The results of the evaluations of these systems show that the EPU increases the maximum differential pressure by 13 psi on valves 704A/B and 850A/B. These MOVs have sufficient capability to absorb this small increase in differential pressure. EPU does not affect the other NSSS valves. The maximum differential pressures / line pressures determined in the design analysis of GL 89-10 motor-operated valves remain bounding for EPU conditions. Therefore, the calculations which determine motor-operated valve thrust and torque values are not affected by the EPU. Flow rates in these systems are not affected by the EPU. The maximum refueling water storage tank temperature increases from 80°F to 104°F at EPU conditions, and therefore the fluid

temperature for applicable valves has increased to 104°F. However, this fluid temperature is not used as an input in the determination of motor-operated valve differential pressures and, therefore, the temperature increase does not affect the results of the current operating basis evaluations.

The EPU does not impact valve factors used in the capability calculations, nor does it impact the results of any calculations using the EPRI performance-prediction methodology.

The EPU does not affect the GL 89-10 motor-operated valve AC and DC degraded voltage values as documented in the applicable design analyses. This is confirmed in LR section 2.3.3, AC Onsite Power System, which indicates no change to AC degraded voltage or loss of voltage relay setpoints, and in LR section 2.3.4, DC Onsite Power System, which indicates negligible load impact to the DC system.

The impact of the EPU on environmental service conditions in the plant areas containing safety-related equipment is addressed in LR section 2.3.1, Environmental Qualification of Electrical Equipment. Based on information in this section, the impact of the EPU on the maximum ambient temperatures for GL 89-10 motor-operated valves used in the motor-derate evaluations is as follows:

- Containment: the pre-uprate maximum containment LOCA temperature, 286°F, bounds the maximum LOCA temperature for EPU conditions.
- Intermediate building: since the pre-uprate maximum containment LOCA temperature bounds the maximum LOCA temperature for EPU conditions, the maximum temperature of 115°F in the intermediate building due to a LOCA inside containment remains bounding for EPU conditions.
- Auxiliary building: the maximum ambient temperature for normal operation, 104°F, is not affected by the EPU.
- Turbine building: the maximum ambient temperature for normal operation, 104°F, is not affected by the EPU.
- Screen house: the maximum ambient temperature for normal operation, 104°F, is not affected by the EPU.
- Standby auxiliary feedwater building: the maximum ambient temperature for normal operation, 120°F, is not affected by the EPU.

Accordingly, the ambient temperatures used in the current motor-operated valve motor de-rate evaluations are not affected or remain bounding for EPU conditions.

Generic Letter 96-05

No motor-operated valves are required to be added to the Motor-Operated Valve Program as a result of the EPU. As discussed above, the EPU does not affect the differential pressures / line

pressures determined in GL 89-10 motor-operated valve design analyses. The risk significance of GL 89-10 MOVs is not affected by the EPU. Therefore, the EPU does not affect the requirements of the program for periodic verification of safety-related motor-operated valve capabilities in accordance with GL 96-05.

Generic Letter 95-07

The impact of the EPU on the analyses performed to show that motor-operated valve actuators have sufficient thrust to overcome the pressure-locked bonnet condition was evaluated; evaluations for the affected valves are shown in Table 2.2.4-3. The results show that EPU does not affect the pressure locking evaluations, and that therefore the thrust required to open the applicable motor-operated valves remains less than the motor actuator capabilities for EPU conditions.

For valves susceptible to pressure locking / thermal binding and justified on the basis of valve design, or on operational configuration / conditions imposed, the EPU does not affect valve design, the function of the valves, or the operational considerations / conditions imposed. The EPU does not create any new conditions which would affect susceptibility of valves to pressure locking or thermal binding.

Lessons Learned

Regulatory requirements, other utility experience, EPRI guidelines, User Group recommendations, and vendor information are reviewed and factored into the preventive maintenance / test frequencies to the extent practical. In particular, preventive maintenance procedures are enhanced as appropriate to include User Group and EPPRI recommendations and updates / notices from vendors.

2.2.4.2.3 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Aging effects of safety-related valves and pumps are primarily managed by the Periodic Surveillance and Preventive Maintenance Program. Because no new materials are being added within existing evaluation boundaries and because component internal and external environments remain within parameters previously evaluated, implementation of the EPU does not diminish the ability of this program to provide reasonable assurance that the aging effects of safety-related valves and pumps will be effectively managed and that their functional performance will be maintained through the period of extended operation.

2.2.4.2.4 Conclusion

Ginna has reviewed the assessments related to the functional performance of safety-related valves and pumps and concludes that the effects of the proposed EPU on safety-related pumps and valves have been adequately addressed. Ginna further concludes that the effects of the proposed EPU on motor-operated valve programs related to GL 89-10, GL 96-05, and GL 95-07 have been adequately evaluated, and that the lessons learned from those programs to other safety-related power-operated valves has been addressed. Based on this, Ginna concludes that it has been demonstrated that safety-related valves and pumps will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46, GDC-54, and 10CFR50.55a(f) following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to safety-related valves and pumps.

TABLE 2.2.4-1 IMPACT OF EPU on EVALUATIONS of GL 89-10 MOVs in BOP SYSTEMS			
MOV(s)	**Valve Name / Function**	**MOV Operating Parameters for Current Plant Conditions**	**Impact of EPU on MOV Operating Parameters**
		Notes: (1) Valve opening direction is identified by "(O)," valve closing direction is identified by "(C)." (2) Line pressures are normally the same as the upstream / downstream pressures. They are not identified in the table, unless shown in the analysis as different from upstream / downstream pressures.	
0813	CCW Supply to Rx Support Cooler	Upstream pressure (C): maximum system operating pressure, 100 psig.	Not affected by the EPU.
0814	CCW Return from Rx Support Cooler	Downstream pressure (C): 0 psig.	Not affected by the EPU.
		Differential pressure (C): 100 psid.	Not affected by the EPU.
		Flow rate (C): < 15 fps.	Not affected by the EPU.
		Temperature (C): 100°F.	Not affected by the EPU.

TABLE 2.2.4-1

IMPACT OF EPU on EVALUATIONS of GL 89-10 MOVs in BOP SYSTEMS

MOV(s)	Valve Name / Function	MOV Operating Parameters for Current Plant Conditions	Impact of EPU on MOV Operating Parameters
3504A 3505A	TDAFW Pump Steam Admission from SG A / B	Upstream pressure (O) / (C): 1118 psig, based on pressure setpoint of first bank of MSSVs, 1085 psig, and 3% accumulation.	Pressure setpoint of MSSVs is not affected by the EPU.
		Downstream pressure (O): 0 psig.	Not affected by the EPU.
		Differential pressure (O): 1118 psid.	Not affected by the EPU.
		Differential pressure (C): 200 psid, based on backpressure from TDAFW pump and actual DP test data. Downstream pressure (C): 918 psig.	Upstream steam conditions, based on MSSV set pressure, are not affected by the EPU. Since TDAFW pump capacity is not affected by the EPU, backpressure from the TDAFW pump is not affected. Therefore, the differential pressure across MOVs is unaffected by the EPU.
		Flow rate (O) / (C): > 15 fps.	Not affected by the EPU.
		Temperature (O) / (C): saturation temp. at 1133 psia, 560°F.	Not affected by the EPU.

TABLE 2.2.4-1 IMPACT OF EPU on EVALUATIONS of GL 89-10 MOVs in BOP SYSTEMS			
MOV(s)	**Valve Name / Function**	**MOV Operating Parameters for Current Plant Conditions**	**Impact of EPU on MOV Operating Parameters**
9701A 9701B	Standby AFW (SBAFW) Pump 1C / 1D Discharge	Upstream pressure (O) / (C): 1461 psig, based on SBAFW pump shutoff head, with suction from the service water (SW) System.	SBAFW pump head performance not affected by the EPU. SW pump head performance not affected by the EPU.
		Downstream pressure (O) / (C): 0 psig.	Not affected by the EPU.
		Differential pressure (O) / (C): 1461 psid.	Not affected by the EPU.
		Flow rate (O) / (C): nominal SBAFW flow rate, 200 gpm.	Accident analysis flow rate is 235 gpm for EPU conditions.
		Temperature (O) / (C): 100°F, based on maximum CST temperature.	Maximum temperature for EPU conditions is 104°F.

TABLE 2.2.4-2 IMPACT OF EPU on EVALUATIONS of GL 89-10 MOVs in NSSS SYSTEMS			
MOV(s)	**Valve Name /Function**	**MOV Operating Parameters for Current Plant Conditions**	**Impact of EPU on MOV Operating Parameters**
		Notes: (1) Valve opening direction is identified by "(O)," valve closing direction is identified by "(C)." (2) Line pressures are normally the same as the upstream / downstream pressures. They are not identified in the table, unless shown in the analysis as different from upstream / downstream pressures.	
0313	RCP Seal Water Return Line Containment Isolation	Upstream pressure (C): 150 psig, pressure setpoint of Relief Valve 314 (upstream of MOV). Downstream pressure (C): 0 psig. Differential pressure (C): 150 psid. Flow rate (C): 125 gpm, RV 314 flow rate. Temperature (C): 300°F, based on a temperature of 100°F above the RCP seal leak-off temperature alarm setpoint.	Pressure setpoint of RV 314 not affected by the EPU. Not affected by the EPU. Not affected by the EPU. RV 314 flowrate not affected by the EPU. RCP seal leak-off temperature alarm setpoint not affected by the EPU.

TABLE 2.2.4-2 IMPACT OF EPU on EVALUATIONS of GL 89-10 MOVs in NSSS SYSTEMS			
0704A 0704B	RHR Pump Suction Cross-Connect Isolation	Upstream pressure (C): 46 psig. Calculated pressure is based on: • Maximum containment pressure (54 psia) • Maximum containment water level elevation at switchover (242.59 ft).	Values calculated for EPU conditions. Not affected by the EPU.
		Differential pressure (C): 46 psid.	Values calculated for EPU conditions.
		Downstream pressure (C): 0 psig.	Not affected by the EPU.
		Flow rate (C): 50 gpm.	Not affected by the EPU.
		Temperature (C): 212°F.	Not affected by the EPU.
0856	RHR Pump Suction from RWST	Upstream pressure (C): 33 psig (minimum value) - calculated pressure is 6 psig, based on: • Elevation of RWST at 28% level. • Density of RWST water at 60°F.	Minimum RWST temperature at EPU conditions is 50°F. However, this does not affect the upstream pressure, since a minimum value of 33 psig is used.
		Downstream pressure (C): 0 psig.	Not affected by the EPU.
		Differential pressure (C): 33 psid (minimum value).	Not affected by the EPU.
		Flow rate (C): < 15 fps.	Not affected by the EPU.
		Temperature (C): 80°F.	Maximum RWST temperature at EPU conditions is 104°F.

TABLE 2.2.4-3

IMPACT OF EPU on EVALUATIONS of GATE VALVE PRESSURE LOCKING

MOV(s)	Valve Name / Function	Evaluation of Gate Valve Pressure Locking for Current Plant Conditions	Impact of EPU on Current Evaluation
0313	RCP Seal Water Return Line Containment Isolation	The Pressure Locking / Thermal Binding (PL/TB) Technical Report specifies that the valve must close against a differential pressure of 150 psid (based on the GL 89-10 MOV operating basis evaluation, which identifies valve upstream pressure as 150 psig, valve downstream pressure as 0 psig). The pressure locking analysis in the PL/TB Technical Report states that 150 psig is used as the bonnet pressure, with the upstream and downstream pressures conservatively assumed to be 0 psig. The pressure locking calculation shows that the thrust required to open the valve under design basis pressure locking conditions is less than the motor capability at degraded voltage conditions.	As indicated in the GL 89-10 MOV operating basis evaluation for this MOV, the valve upstream pressure, 150 psig, is based on the pressure setpoint of RV 314. This setpoint is not affected by the EPU. Therefore, the bonnet pressure of 150 psig used in the pressure locking analysis is not affected by the EPU, and the conclusions of the pressure locking calculation remain valid for the EPU.

TABLE 2.2.4-3 IMPACT OF EPU on EVALUATIONS of GATE VALVE PRESSURE LOCKING			
MOV(s)	**Valve Name / Function**	**Evaluation of Gate Valve Pressure Locking for Current Plant Conditions**	**Impact of EPU on Current Evaluation**
0871A 0871B	SI Pump 1A / 1B to SI Pump 1C Crossover	As addressed in the PL/TB Technical Report, these valves are required to re-open after an inadvertent closure against the shutoff head of the SI pump. The pressure locking analysis in the PL/TB Technical Report shows that the bonnet pressure is 1533 psig, based on: • SI pump shutoff head • RWST head (based on RWST temp. of 60°F) The upstream and downstream pressure is determined to be 33 psig (based on RWST temp. of 60°F). The pressure locking calculation shows that the thrust required to open the valve under design basis pressure locking conditions is less than the actuator motor capability.	SI pump performance is not affected by the EPU. Although the minimum RWST temperature at EPU conditions is 50°F, this does not affect the analysis results in the PL/TB Technical Report. Therefore, the bonnet pressure of 1533 psig used in the pressure locking analysis is not affected by the EPU, and the conclusions of the pressure locking calculation remain valid for the EPU.

TABLE 2.2.4-3 IMPACT OF EPU on EVALUATIONS of GATE VALVE PRESSURE LOCKING			
MOV(s)	**Valve Name / Function**	**Evaluation of Gate Valve Pressure Locking for Current Plant Conditions**	**Impact of EPU on Current Evaluation**
0759A 0759B	CCW from RCP A / B	As addressed in the PL/TB Technical Report, reopening of these valves after inadvertent or intentional closure is important to plant reliability. For the pressure locking analysis, valve bonnet pressure is 140 psig, based on an analysis referenced in the GL 89-10 MOV operating basis evaluation, and upstream and downstream pressures are assumed to be 0 psig. The pressure locking calculation shows that the thrust required to open the valve under design basis pressure locking conditions is less than the actuator motor capability.	As indicated in the analysis referenced in the GL 89-10 operating basis evaluation, the valve differential pressure, 140 psid, is based on the setpoint and capacity of RV 758 A and 758B. This RV setpoint and capacity is not affected by the EPU. Therefore, the bonnet pressure of 140 psig used in the pressure locking analysis is not affected by the EPU, and the conclusions of the pressure locking calculation remain valid for the EPU.

TABLE 2.2.4-3 IMPACT OF EPU on EVALUATIONS of GATE VALVE PRESSURE LOCKING			
MOV(s)	**Valve Name / Function**	**Evaluation of Gate Valve Pressure Locking for Current Plant Conditions**	**Impact of EPU on Current Evaluation**
0813 0814	CCW Supply to Rx Support Cooler CCW Return from Rx Support Cooler	As addressed in the PL/TB Technical Report, reopening of these valves after inadvertent or intentional closure is important to plant reliability. For the pressure locking analysis, valve bonnet pressure is 100 psig, based on the analysis in the GL 89-10 MOV operating basis evaluation, and upstream and downstream pressures are assumed to be 0 psig. The thrust determined in the pressure locking calculation is less than the actuator motor capability for MOV 0813 and MOV 0814.	As indicated in the GL 89-10 MOV operating basis evaluation, the valve upstream pressure, 100 psig, is based on the maximum system operating pressure. This operating pressure is not affected by the EPU. Therefore, the bonnet pressure of 100 psig used in the pressure locking analysis is not affected by the EPU, and the thrust determined in the pressure locking calculation remains less than the actuator motor capability for MOV 0813 and MOV 0814 for EPU conditions.

TABLE 2.2.4-3 IMPACT OF EPU on EVALUATIONS of GATE VALVE PRESSURE LOCKING			
MOV(s)	**Valve Name / Function**	**Evaluation of Gate Valve Pressure Locking for Current Plant Conditions**	**Impact of EPU on Current Evaluation**
9629A 9629B	Standby AFW (SBAFW) Pump 1C / 1D Suction	As addressed in the PL/TB Technical Report, these valves would be required to be opened if the Standby AFW System is needed. For the pressure locking analysis, valve bonnet pressure is 95 psig, based on the GL 89-10 MOV operating basis evaluation, and upstream and downstream pressures are assumed to be 0 psig. The pressure locking calculation shows that the thrust required to open the valve under design basis pressure locking conditions is less than the actuator motor capability.	As indicated in the GL 89-10 MOV operating basis evaluation, the valve upstream pressure, 95 psig, is based on the service water pump head for a system demand flow of 4642 gpm. The service water pump head performance is not affected by the EPU. Therefore, the bonnet pressure of 95 psig used in the pressure locking analysis is not affected by the EPU, and the conclusions of the pressure locking calculation remain valid for the EPU.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

2.2.5.1 Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant release of radioactive materials to the environment are also covered by this section. The Ginna Nuclear Power Plant, LLC (Ginna) review focuses on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on:

- GDC-1, insofar as it requires that Structures, Systems, and Components (SSC's) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed;
- GDC-2, insofar as it requires that SSC's important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions;
- GDC-4, insofar as it requires that SSC's important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture;
- GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical;
- 10CFR100, Appendix A, which sets forth the principal seismic and geological considerations of the seismic and geologic characteristics of the plant site; and
- 10CFR50, Appendix B, which sets quality assurance requirements for equipment important to safety

Specific review criteria are contained in SRP Section 3.10.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s, the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station safety-related SSC's being adequately designed for seismic events relative to conformance to:

- GDC-1 is described in Ginna UFSAR section 3.1.2.1, General Design Criterion 1, Quality Standards and Records. Conformance to the requirements of GDC-1 is described in the following:
  - The classification of structures, systems and components is discussed in Ginna UFSAR section 3.2.
  - A set of all quality assurance data generated during fabrication and erection of the essential components of the plant, as defined by the Ginna Station construction quality assurance program is retained.
  - The quality control and quality assurance program for Ginna Station construction is described in Ginna UFSAR section 17.1.
  - The current quality assurance program for Ginna Station is referenced in Ginna UFSAR section 17.2.

- GDC-2 is described in Ginna UFSAR section 3.1.2.2, General Design Criterion 2, Design bases for protection against natural phenomena. Conformance to the requirements of GDC-2 is described in the following:
  - Tornado and flood protection measures are discussed in Ginna UFSAR sections 3.3 and 3.4, respectively. Procedures have been written that will be followed in the event of such natural phenomena. The occurrence of such phenomena is discussed in Ginna UFSAR section 2.
  - Generic Letter 87-02, Supplement 1, dated May 22, 1992, transmitted Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure, Revision 2, dated February 14, 1992 (GIP-2).
  - SSER No. 2 approved the methodology in the SQUG Generic Implementation Procedure for use in verification of equipment seismic adequacy; including equipment involved in future modifications and replacement equipment. In letters dated November 30, 1992, and June 8, 1992, the NRC accepted RG&E's response to Generic Letter 87-02, Supplement 1.

- GDC–4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4, Environmental and Missile Design Bases. As described in this Ginna UFSAR section, Ginna Station received post-construction review as part of the systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC-4 is described in the following:
    - Seismic and Environmental Design of Mechanical and Electrical Equipment descriptions (Ginna UFSAR sections 3.10 and 3.11 respectively).
    - Missile Protection (Ginna UFSAR section 3.5.1) Conformance to the requirements of GDC-4 ensuring that safety-related SSC's are adequately protected from internally generated missiles
    - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping. (Ginna UFSAR section 3.6) A review of postulated breaks inside and outside containment was conducted as part of the Systematic Evaluation Program (SEP).
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)

- GDC-14 is described in Ginna UFSAR Section 3.1.2.14, General Design Criterion 14, Reactor Coolant Pressure Boundary. Conformance to the requirements of GDC-14 is described in the following:
    - All piping components and supporting structures of the reactor coolant system were designed as Class I, and later reevaluated as Seismic Category I equipment as defined in Ginna UFSAR section 3.7.
    - All pressure containing components of the reactor coolant system were designed, fabricated, inspected, and tested in conformance with the code requirements listed in Ginna UFSAR Table 5.2-1. Therefore, the probability of abnormal leakage, of rapidly propagating failure and of gross rupture is very low.

- GDC-30 is described in Ginna UFSAR section 3.1.2.30, General Design Criterion 30, Quality of Reactor Coolant Pressure Boundary. Conformance to the requirements of GDC-1 is described in the following:
    - Quality standards of material selection, design, fabrication, and inspection for the Ginna Station reactor coolant system conformed to the applicable provisions of recognized codes and good nuclear practice of that period.
    - Details of the quality assurance programs, test procedures, and inspection acceptance levels are given in Ginna UFSAR section 17.1
    - Particular emphasis was placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the code used. Ginna UFSAR Table 3.2-1 gives the code requirements used for the reactor coolant system.

- 10CFR100, Appendix A, Seismic and Geologic Siting Criteria.
    - Site location and description are described in Ginna UFSAR section 2.1.1.

- Information on the geology, seismology and geotechnical engineering of the site are described in Ginna UFSAR section 2.5

- 10CFR50, Appendix B, Quality Assurance 18 Point Criteria
  - The quality control and quality assurance program for Ginna Station construction is described in Ginna UFSAR section 17.1.
  - The current quality assurance program for Ginna Station is referenced in Ginna UFSAR section 17.2.

As described in these Ginna UFSAR sections, Ginna Station received a post-construction review as part of the Systematic Evaluation Program. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's systems and components have been evaluated for plant license renewal. Plant Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power plant, (NUREG-1786), dated May, 2004.

2.2.5.2 Technical Evaluation

2.2.5.2.1 Introduction

Safety-related SSC's at Ginna Station are designed for both seismic and dynamic events as described in Ginna UFSAR sections 3.5, 3.7, 3.9, 3.10, and 3.11. Ginna UFSAR section 3.7, "Seismic Design," provides the general requirements for seismic design. Ginna UFSAR section 3.9.2 includes discussions regarding dynamic system analysis and testing. Ginna UFSAR section 3.9 Tables and Figures provide additional information on seismic qualification. Ginna UFSAR section 3.10 provides details regarding seismic qualification of safety-related instrumentation and electrical equipment. Ginna UFSAR section 3.11 provides details regarding environmental qualification of safety-related mechanical and electrical equipment.

Seismic design is not impacted by EPU since seismic requirements remain unchanged. There is no change to seismic inputs (amplified response spectra) or loads resulting from EPU. The existing seismic design basis for piping and supports remains valid and unaffected by EPU. Hence, piping and support seismic loadings will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-2.

Dynamic qualification can be impacted if equipment operating conditions such as pressure, temperature, and fluid flow change as a result of EPU. Additionally, ability of the equipment to withstand effects of pipe-whip, jet impingement, internal, and external missiles may also be affected as a result of EPU impact on systems in physical proximity

of essential safety-related equipment. LR section that address these issues are included in LR section 2.2.2.1, NSSS – Piping and Supports (Class 1) and section 2.2.5.2.2, BOP (All Non-Class 1), below.

2.2.5.2.2 Description of Analyses and Evaluations

Seismic input and qualification requirements for safety-related equipment are not affected by EPU. Quality Assurance requirements related to 10CFR50, Appendix B are not affected. Effects of changes in pressure, temperature and fluid flow on safety-related equipment have been found to be within the rated capacity of the equipment or have been evaluated by the original equipment supplier, and found to comply with the appropriate qualification requirements.

Dynamic effects of internally and externally generated missiles under EPU have been evaluated and are addressed in LR section 2.5.1.2, Missile Protection. Dynamic effects of pipe-whip and jet impingement under uprate conditions have been evaluated and are addressed in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and LR section 2.5.1.3, Pipe Failures. Based on these evaluations, EPU will have no adverse impact on essential equipment as a result of pipe whip, jet impingement, internal, and external missiles.

Evaluations related to dynamic and environmental effects of the EPU are addressed in the following LR section:

- NSSS piping and supports – LR section 2.2.2.1, NSSS – Piping and Supports (Class 1)
- Piping and supports – LR section 2.2.2.2, BOP (All Non-Class 1)

  This LR section addresses modifications to the Main Steam and Feedwater systems resulting from EPU.
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Environmental qualification of electrical equipment – LR section 2.3.1, Environmental Qualification
- Protection against turbine missiles and internal missiles is discussed in LR section 2.5.1.2, Missile Protection
- Piping failures – LR section 2.5.1.3, Pipe Failures
- Addition of new operators to existing feedwater isolation valves – LR section 2.5.5.4, Condensate and Feedwater

As a result of the EPU evaluations described in the various license report sections, the following conclusions can be made:

- Routing of electrical cables/raceways/conduits maintained acceptable.

- Seismic anchorage of equipment maintained acceptable.
- All pipe routing and valve placement determined acceptable so that no new or previously unevaluated missile, pipe whip or jet impingement threat to equipment essential to safe shutdown, containment isolation, reactor cooling or containment or reactor cooling exists.
- All equipment mass, height or center of gravity, and physical separation between components, or between components and structures, determined not to cause new or previously unevaluated interactions.
- All equipment relocations shown not to result in seismic II/I concerns.

If there are any modifications that require changes to the above, as a result of EPU, these items will be controlled by the Station change process

2.2.5.2.3 Results

The evaluation of changes in system design configurations that are required for the proposed EPU concluded that safety-related equipment will continue to be protected from seismic and dynamic events, and will continue to meet the Ginna Station current licensing basis.

Evaluation of Impact on Renewed Plant Operation Licensing Evaluations and License Renewal Programs

With respect to the licensing renewal described in NUREG-1786, EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the Plant at EPU conditions do not add any new or previously unevaluated materials to the plant systems.

The plant changes proposed for the EPU have been evaluated to ensure that there are no additions to the scope of nonsafety-related SSCs whose failure could prevent the satisfactory accomplishment of a function required by 10CFR54.4 (a)(1) and (a)(3). (Additionally, the plant change process requires further evaluation of the change as the physical modifications are engineered for installation. These further evaluations will ensure that any changes subsequent to the initial review are evaluated for the effects on license renewal.)

System and component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring Aging Management Review (AMR) are identified.

2.2.5.3 Conclusion

The Ginna review of the effects of the proposed EPU on the qualification of mechanical and electrical equipment concludes that the review has 1) adequately addressed the effects of the proposed EPU on equipment and 2) demonstrated that the equipment will

continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-1, 2, 4, 14, 30 and 10CFR100, Appendix A and 10CFR50, Appendix B.

2.2.6 NSSS Design Transients

2.2.6.1 Regulatory Evaluation

NSSS design transients are developed for use in the analyses of the cyclic behavior of the NSSS SSCs. To provide the necessary high degree of integrity for them, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the transients resulting from various plant operating conditions. Ginna Nuclear Power Plant, LLC's (Ginna) review focused primarily on the effects of the proposed EPU on NSSS design parameters that are used in transient analyses, and how those differences in design parameters required revising NSSS design transients. Ginna's acceptance criteria for this review are based on:

- GDC-1 insofar as it relates to safety-related components being designed, fabricated, erected, constructed, tested and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-function to be performed.

- GDC-2 insofar as it relates to safety-related mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety function.

- GDC-14 insofar as it relates to the reactor coolant pressure boundary being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC-15 insofar as it relates to the mechanical components of the reactor coolant system being designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Specific review criteria are contained in the SRP, Section 3.9.1, and other guidance provided in Matrix 2 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna UFSAR Section 3.1 the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR section 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR

describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design basis transient analysis regarding conformance to:

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1. General Design Criterion 1 - Quality Standards and Records, wherein it is noted that all SSCs of the facility were classified according to their importance. The classification of structures and equipment is discussed in Ginna UFSAR Section 3.2. SSCs were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. Discussions of applicable codes and standards, quality assurance programs, test provisions, etc., that were used is given in UFSAR section 3.2, and in the ensuing sections wherein the SSCs subject to the design transients are described. The quality control and quality assurance program for Ginna Station construction, and the current quality assurance program, under which this system was installed and maintained, are described in Ginna UFSAR Section 17.1, and 17.2, respectively.

- GDC-2 is described in Ginna UFSAR section 3.1.2.1.2, General Design Criterion 2 - Design Bases for Protection Against Natural Phenomena. As described therein, Ginna Station, as participant in the Seismic Qualification Utility Group (SQUG), received NRC acceptance of its response to Generic Letter 87-02, including approval of the methodology in the SQUG Generic Implementation Procedure for use in verification of equipment seismic adequacy including equipment involved in future modifications and replacement equipment.

- GDC-14 is described in Ginna UFSAR section 3.1.2.2.5, General Design Criterion 14 - Reactor Coolant Pressure Boundary. In this UFSAR section it is noted that all piping components and supporting structures of the reactor coolant system were designed as Class I and later reevaluated as Seismic Category I equipment as defined in UFSAR Section 3.7. All pressure containing components of the reactor coolant system were designed, fabricated, inspected, and tested in conformance with the code requirements listed in Table 5.2-1 of the UFSAR. Therefore, the probability of abnormal leakage, of rapidly propagating failure and of gross rupture is very low, and compliance with this criterion is assured.

- GDC-15 is described in GINNA UFSAR section 3.1.2.2.6, General Design Criterion 15 - Reactor Coolant System Design. As described in this UFSAR section, the reactor coolant system and associated auxiliary systems were designed with sufficient margins so that design conditions are not exceeded during MODES 1 and 2, including anticipated

operational occurrences. There exists a reasonable range for maneuvering during plant operation with allowance for pressure transients without actuation of the safety valves. It is furthermore stated that the analysis presented in Ginna UFSAR Chapter 15 demonstrates the ability of the plant to safely undergo all anticipated transients with pressure peaks below 2485 psig. Further discussion of the NSSS design transients and the capability of the various components in the reactor coolant and connected auxiliary systems to withstand the effects of cyclic loads is presented in UFSAR sections 3.9 and 5.1.

In addition to the evaluations described in the UFSAR, the Ginna Station's NSSS and associated auxiliary system components were evaluated for the continued acceptability and applicability of the design basis transients for the purpose of plant license renewal. The results of that review are found in NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R.E. Ginna Nuclear Power Plant, May 2004. System and system component materials of construction, operating history and programs used to manage aging effects are documented in the SER. The SER, in sections 2.3.1, 3.1.2.3.8 and 4.3.2, considers the frequency and severity of the operating transients assumed in the design of the SSCs of the Ginna NSSS and associated auxiliary systems over the extended term of the operating license.

2.2.6.2 Technical Evaluation

2.2.6.2.1 Introduction

As discussed in Chapter 5 of the Ginna UFSAR, the systems, structures and components (SSCs) important to safety in the reactor coolant system and its auxiliary systems are designed to withstand the effects of the cyclic loads from reactor coolant system (NSSS) temperature and pressure changes. Such cyclic loading is the result of normal unit load transients, i.e., design basis transients. This evaluation compares the Ginna Station design parameters developed for the proposed EPU to the design parameters used in the current design basis transients. Where revisions were necessary, comparative analyses were performed and the transients revised, as needed, to reflect the operating conditions for the proposed EPU.

In 1994 through 1995 a T_{avg} reduction program was developed for use in the Ginna Station. As part of the program for implementing this T_{avg} reduction, NSSS design transients (i.e., temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS components. To provide the necessary high degree of integrity for the NSSS components, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. The transients selected for use in component fatigue analyses are representative of operating conditions that would be considered to occur during plant operations of possible significance to component cyclic behavior due to their severity or frequency. The selected transients are representative of plant transients that, when used as a basis for component fatigue analysis, would provide confidence

that the component is appropriate for its application over the operating license period of the plant.

2.2.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

NSSS design transients were based primarily on the NSSS design parameters developed for the proposed EPU, presented in LR section 1.1 Nuclear Steam Supply System Parameters. The design parameters upon which the current applicable NSSS design transients are based were compared to the design parameters for the EPU, and shown to be different for the values of reactor coolant system (RCS) vessel T_{avg}, steam pressure, and feedwater temperature. The differences are primarily due to RCS vessel T_{avg} and feedwater temperature windows. These differences were sufficient to require a reassessment of the original NSSS design transients, and that revised NSSS design transients be specified for the EPU.

It should be noted that the partial loss-of-flow and loss-of-load design transients, while not listed in the Ginna UFSAR, were included in WCAP-14460,T_{avg} reduction program (Reference: WCAP-14460, *Reactor Pressure Vessel and Internals System Evaluations for the R. E. Ginna Nuclear Power Plant T_{avg} Reduction Program*, December 1995), and so were reviewed for the proposed EPU.

2.2.6.2.3 Description of Analyses and Evaluations

The Ginna Station design parameters for the proposed EPU were compared to the design parameters used in the current design transients. Where revisions were necessary due to sufficient differences between the two sets of operating conditions, evaluations and analyses of the existing applicable NSSS design transients for the Ginna Station were performed, and the transients were revised, as needed, to reflect the operating conditions for the EPU.

2.2.6.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

NSSS design transients are only impacted if there is some plant change that results in a change in plant operating conditions (i.e., design condition T-hot, T-cold, RCS/pressurizer pressure setpoint, steam generator steam pressure, or feedwater temperature). If the NSSS design transient review results in no changes to plant operating methods or procedures then the transient set reviewed for plant license renewal remains applicable. Specific details of the transient result comparisons are provided within the LR section that evaluates the components of concern.

2.2.6.2.5 NSSS Design Transients Results

The NSSS design transients were input to the NSSS primary and secondary side component structural and fatigue analyses and evaluations. The final acceptance was determined by the

component stress and fatigue analyses discussed in LR section 2.2 Mechanical and Civil Engineering, individually for each component.

A list of the NSSS design transients applicable to the Ginna EPU, with their associated design value frequencies of occurrence are shown in Table 2.2.6-1. The transients listed and their associated frequencies of occurrence are unchanged from those in the current design basis transient list. The design transients that require revision for the EPU are also noted in Table 2.2.6-1.

Consistent with the current NSSS design transients, the revised NSSS design transients determined for the proposed EPU are conservative representations of transients that, when used as a basis for component fatigue analyses, provide confidence that the component remains appropriate for its application over the operating license period of the Ginna Station.

Revised NSSS design transients have been determined for the proposed Ginna EPU. These revised transients were used in the NSSS component structural and fatigue evaluations at EPU conditions. The results of the component structural and fatigue evaluations are provided in the individual LR section for each NSSS component.

Table 2.2.6-1 List of Design Basis NSSS Design Transients		
Transient Description	**Number of Occurrences[a]**	**Transient Required Revision due to the Uprating**
Plant Heatup	200	No
Plant Cooldown	200	No
Plant Loading at 5% of Full Power per Minute	14,500	Yes
Plant Unloading at 5% of Full Power per Minute	14,500	Yes
Step Load Increase of 10% of Full Power	2,000	Yes
Step Load Decrease of 10% of Full Power	2,000	Yes
Step Load Rejection from 100% to 50%	200	Yes
Reactor Trip	400	Yes
Partial Loss of Flow [b]	80	Yes
Loss of Load [b]	80	Yes
Hydrostatic test – 3125 psia	5	No
Hydrostatic test – 2500 psia	40	No
Steady-State Fluctuations	Infinite	No

Notes:
a. These are unchanged from the existing design basis.
b. Not a licensing basis design transient. However, it was included in the EPU program to be consistent with the list of design transients included in WCAP-14460.

2.2.6.3 Conclusions

Ginna has reviewed the evaluation of the effects of the EPU on the NSSS design transients and concludes that the required design transient revisions have been adequately addressed. Ginna further concludes that the revised NSSS design transients have been incorporated into the transient analysis of the safety-related NSSS systems and components and that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-1, -2, -14 and -15 following implementation of the EPU. Therefore, Ginna finds the EPU acceptable with respect to the NSSS design transients.

2.2.7 Bottom-Mounted Instrumentation Guide Tubes and Flux Thimbles

2.2.7.1 Regulatory Evaluation

The Bottom Mounted Instrumentation (BMI), includes guide tubes, flux thimbles, and the seal table, along with the in-core detectors (inserted in the flux thimbles to monitor the flux in the core). This system is employed to evaluate the core power distributions throughout core lifetime to verify that the thermal design criteria are met. The system provides a means for acquiring data and performs no operational plant control.

The Ginna Nuclear Power Plant, LLC (Ginna) review of the BMI focused on the effects of the proposed EPU on the structural integrity of BMI components and their continued functionality, including the capability to maintain integrity of the reactor coolant pressure boundary (RCPB), and withstand any adverse dynamic loads under the maximum temperatures and pressures associated with the proposed EPU. Ginna's acceptance criteria for the BMI components are based on:

- GDC-1 and 10CFR50.55a, insofar as they require that safety-related structures, systems, and components (SSCs) be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety functions to be performed
- GDC-2, insofar as it requires that safety-related SSCs be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions
- GDC-4, insofar as it requires that safety-related SSCs be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles
- GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected and tested so as to have an extremely low probability of rapidly propagating fracture

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of the Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to,

conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, Ginna Station safety-related SSCs will continue to be protected against the failure of the BMI consistent with the following:

GDC-1 is described in Ginna UFSAR section 3.1.2.1.1, General Design Criteria 1 - "Quality Standards and Records." As described in this UFSAR section, safety-related systems and components were designed, fabricated, inspected, and erected to the applicable provisions of the then recognized codes, good nuclear practice, and the quality standards that reflected their importance. As part of the SEP, the original codes and standards used in the design of systems and components were compared with later licensing criteria based on Regulatory Guide (RG) 1.26 and 10CFR50.55a. Specifically, SEP Topic V-6 reviewed the safety aspects that affect reactor vessel and nozzle integrity for compliance with 10CFR50. As noted in that SEP assessment, the components of the BMI system are designed, fabricated and installed meeting criteria equivalent to GDC-1 and 10CFR50.55a.

GDC-2 is described in Ginna UFSAR section 3.1.2.1.2, "Design Basis for Protection Against Natural Phenomena." As described in this UFSAR section, all systems and components designated seismic category I are designed so there is no loss of function in the event of a safe shutdown earthquake. The evaluation of the BMI guide tubing and the flux thimbles confirms the operability and the structural integrity for all applicable loadings and load combinations.

GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, "Environmental and Missile Design Bases." As described in this UFSAR section, a review of postulated pipe breaks inside containment, including dynamic effects, and internally generated missiles was conducted as part of the SEP. BMI Guide tubing and flux thimbles are designed to accommodate the effects of, and are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

GDC-14 is described in Ginna UFSAR section 3.1.2.2.5, "Reactor Coolant Pressure Boundary." As described in this UFSAR section, all pressure retaining components of the reactor coolant system were designed, fabricated, inspected and tested in conformance with code requirements as discussed in UFSAR section 5.2. The BMI guide tubing and the flux thimbles are designed and fabricated as Seismic Category I components to maintain the RCPB. They have a very low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

The current licensing basis for the reactor vessel and reactor internals is addressed in Licensing Report (LR) sections 2.2.2.3, "Reactor Vessel and Supports," and 2.2.3, "Reactor Pressure Vessel Internals and Core Supports," respectively.

The BMI guide tubes and flux thimbles are described in UFSAR sections 3.9.5.1.3, "In-Core Instrumentation Support Structures," 7.7.4.2.1, "General (System Design of In-core

Instrumentation)," 7.7.4.2.3, "Movable Miniature Neutron Flux Detectors," and 7.7.4.2.4, "Control and Readout System."

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's systems and components have been evaluated for plant license renewal. Plant system and component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The BMI guide tubes and flux thimbles were previously evaluated for plant license renewal. The results of the NRC review for the BMI guide tubing are documented in section 2.3.1.2 (Table 2.3.1-2) and section 2.3.1.3 of NUREG-1786. The flux thimbles inspection program is evaluated in section 3.1.2.3.7 of NUREG-1786.

2.2.7.2 Technical Evaluation

2.2.7.2.1 Introduction

The analyses and evaluations of the Ginna Station BMI presented herein assess the impact of EPU on the structural integrity of the BMI. This assessment considered the thermal transients, maximum operating temperatures and pressures, and design basis accident displacements that would result from the proposed EPU operating conditions. The results of these analyses and evaluations show that the stresses in the BMI guide tubing for the EPU remain within the allowable limit.

The BMI consists of guide tubing, flux thimbles, retractable miniature flux detectors, and the seal table. The retractable miniature detectors are inserted into the flux thimbles, which enter the guide tubing at the seal table, pass through the tubing into the reactor vessel, through the lower internals instrument columns and then into the fuel assemblies. Each detector provides axial flux distribution data along the center portion of a fuel assembly. These data are then processed to obtain a core flux map.

The guide tubing and flux thimbles serve as pressure barriers between the RCS and the containment atmosphere. At the seal table, the pressure boundary for the guide tubing and flux thimbles is maintained by compression fittings where one end of each guide tube has a compressed fitting connection. The other end of each guide tube is welded to the bottom penetration nozzle of the reactor vessel bottom head. The leading ends of the flux thimbles are closed and bullet-nosed.

The guide tubing material is ASTM A276 TP 304 stainless steel, cold drawn, and annealed. The guide tubes are pressure boundary components designed to ANSI B31.1 specifications and

guide tubing is classified as Seismic Category I. The flux thimble material is ASTM A213 Type 316, cold-drawn and heat-treated. The flux thimble for the BMI system is considered an instrument tube. Leakage is readily detectable at the seal table by dedicated leak detection equipment, and each flux thimble can be individually isolated by closing a manual valve on the respective guide tube.

2.2.7.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Today, BMI guide tubing is typically designed to meet the ASME Boiler and Pressure Vessel Code, Section III, Class 1 criteria. However, in keeping with the design code of record for the Ginna Station, the guide tubing was analyzed to the requirements of the ANSI B31.1. In 1990, the Ginna Station performed an evaluation (referred to in the remainder of this LR section as the "initial evaluation") which confirmed the structural acceptability of the BMI tubing, nozzles, valves, compression fittings and supports. The load combinations and allowable stress limits used in the EPU analysis were the same as those used in the initial evaluation. A flux thimble is classified as an instrument tube, so it is outside the jurisdiction of the ASME Code per NA-1130(c). The flux thimbles were qualified as part of the BMI guide tubing, and as such, no separate qualification of the flux thimbles was needed. The weight of the flux thimbles was considered in the qualification of the BMI guide tubing.

The following sets of input parameters were considered in the evaluation for the EPU:

- Nuclear Steam Supply System (NSSS) Parameters for 1817 MWt NSSS Power in LR Table 1-1, section 1.0, Nuclear Steam Supply System Parameters
- NSSS design transients in LR section 2.2.6, NSSS Design Transients
- Displacements at the bottom of the reactor vessel head during a LOCA
- Initial evaluation
- The Ginna technical specification for the reactor BMI tubing supports

The EPU evaluation verified no changes have been made to the facility that would invalidate the support stiffness values determined in the previous study and then utilized the EPU operating parameters (Table 1-1 of LR section 1.0, Nuclear Steam Supply System Parameters) to confirm acceptability under EPU conditions.

2.2.7.2.3 Description of Analyses and Evaluations

Three aspects of BMI guide tubing qualification are potentially affected by EPU changes and were therefore evaluated. They are:

1. Pressure increase during transients,

2. Temperature increase during transients, and the increased normal operating core inlet temperature from LR Table 1-1, and
3. Reactor vessel bottom dome displacements during a LOCA.

The BMI guide tubing was originally qualified for 2580 psig and a reactor coolant temperature of 680°F. The temperature of the guide tube at the seal table is 60-120^{0}F during normal conditions and 286^{0}F during a LOCA. The temperature in the BMI guide tubing is attenuated from the reactor pressure vessel end to the seal table. Since the component weights and seismic loadings are unchanged for EPU, the stresses due to dead weight, operating basis earthquake (OBE) and safe shutdown earthquake (SSE) seismic loadings remain unchanged.

Equations 11 through 14 from ANSI B31.1 – 1973 were re-evaluated for the above three areas of interest. A new pressure stress value was calculated due to a revised maximum pressure during the transient (2638 psig) which is 2.2% higher than the previous qualified value of 2580 psig. Since the maximum temperature for EPU conditions did not exceed 680^{0}F, the stresses due to temperature were not changed and were bounded by the previous analysis. The new displacement values were bounded by the displacement values of the initial evaluation, thus the stresses due to the displacement were also bounded by the initial site evaluation values. The stresses due to seismic and LOCA were combined by the absolute method. The revised total stress values of Equations 11 through 14 were then compared with the respective allowable stress value for each condition.

2.2.7.2.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The BMI system components are included within the scope of license renewal as identified in the NRC License Renewal SER for Ginna Station, NUREG-1786, sections 2.3.1.2 and 2.3.1.3. As discussed in SER section 3.1.2.3.7, "Thimble Tube Inspection Program," BMI components are subject to aging management review programs which have been found acceptable by the NRC in NUREG-1786 for the extended period of operation of Ginna Station. The proposed EPU does not add new materials or components to the BMI system. Therefore, there are no unevaluated material changes to the BMI system with respect to license renewal. The potential effects of operational parameters on the BMI system are already subject to aging management review programs. Thus, no new aging effects requiring management are identified.

2.2.7.2.5 BMI Guide Tubing and Flux Thimbles Results

As shown in Table 2.2.7-1, the results of analyses and evaluations of the proposed EPU operating conditions indicate that the stresses in the BMI guide tubing remain within the allowable limit. The maximum stress ratio for the actual-to-allowable stress is 78%, with a minimum stress margin of 22%.

Table 2.2.7-1 Stress Summary of ANSI B31.1-1973 Equations 11 through 14			
Equation No.	Stress (psi)	Allowable Stress (psi)	Ratio (Actual/Allowable)
11 (Design)	[ ]a,c	15,900	[ ]a,c
12 (Upset)	[ ]a,c	19,080	[ ]a,c
12 (Emergency)	[ ]a,c	28,620	[ ]a,c
13 (Normal)	[ ]a,c	27,350	[ ]a,c
13 (Upset)	[ ]a,c	27,350	[ ]a,c
13 (Emergency)	[ ]a,c	27,350	[ ]a,c
13 (Faulted)	[ ]a,c	27,350	[ ]a,c
14 (Normal)	[ ]a,c	43,250	[ ]a,c
14 (Upset)	[ ]a,c	43,250	[ ]a,c
14 (Emergency)	[ ]a,c	43,250	[ ]a,c
14 (Faulted)	[ ]a,c	43,250	[ ]a,c

2.2.7.3 Conclusions

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the In-core Bottom Mounted Instrumentation and has determined that it has adequately accounted for the effects of changes in plant conditions associated with the proposed EPU on the design of the BMI. Ginna concludes that the BMI will maintain its structural integrity under the operating conditions of the proposed EPU. Ginna further concludes that the BMI will continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR50.55a and GDC-1, GDC-2, GDC-4 and GDC-14. Therefore, Ginna finds the EPU acceptable with respect to the BMI.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

2.3.1.1 Regulatory Evaluation

Environmental Qualification of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from Design Basis Accidents. The Ginna Nuclear Power Plant, LLC (Ginna) review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation including anticipated operational occurrences, and design bases accidents. The Ginna review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for environmental qualification of electrical equipment are based on 10CFR50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Ginna Current Licensing Basis

As described in Ginna UFSAR section 3.11.5, the environmental qualification program is embedded in procedures for design, installation, and maintenance of systems and components. The Nuclear Policy Manual defines the additional quality assurance program requirements for replacement and maintenance of environmentally qualified equipment to ensure compliance with the requirements of 10CFR50.49. The Nuclear Policy Manual is the controlling document for the environmental qualification program and assigns the Engineering Department the responsibility for establishing an evaluation process that documents the basis for any changes in the Environmental Qualification Master List.

As described in Ginna UFSAR section 3.11.1.2, the review of the environmental qualification of safety-related electrical equipment was initiated in 1977 under Topic III-12 of the Systematic Evaluation Program (SEP). In February 1980, the NRC redirected the review program for SEP plants and provided Division of Reactors (DOR) guidelines for evaluating environmental qualification and for identifying safety-related equipment for which environmental qualification was to be addressed. In response to NRC Generic Letter 84-24, RG&E certified program compliance with 10CFR50.49. The NRC concluded that the Environmental Qualification Program complied with 10CFR50.49 and the NRC issues were satisfactorily resolved.

As described in Ginna UFSAR section 3.11.3, Ginna Station is divided into various environmental zones. The limiting environmental conditions for normal operation and post accident are described for the building areas containing safety-related electrical equipment.

As discussed in Ginna UFSAR section 3.11, the post accident radiological environmental conditions utilized for equipment qualification are based on the Loss-of-Coolant Accident (LOCA), whereas the normal operation radiological environments are based on radiation surveys at Ginna Station.

As noted in Ginna UFSAR section 3.11 the integrated dose values provided in Ginna UFSAR Tables 3.11-2 and 3.11-3 represent an accident duration of 1 year and are based on the utilization of a power level and containment volume ratio on the gamma and beta dose estimates developed in Appendix D of Regulatory Guide 1.89, Rev 1. Justification of the above approach, which is based on a comparison of containment performance parameter values used in Regulatory Guide 1.89 versus Ginna Station, is provided in Ginna UFSAR Table 3.11-6.

Ginna UFSAR section 3.11.3.2.3 directs the user to Ginna UFSAR section 12.4.3.3 for the evaluations performed to develop the outside-containment post-accident environmental gamma doses for purposes of equipment qualification. As noted in Ginna UFSAR section 12.4.3.3, the outside containment post-LOCA radiation environments were developed as part of the vital area access / environmental qualification effort associated with demonstrating compliance with NUREG 0578, and are documented in "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces / Systems which may be used in Post Accident Operations Outside Containment at R. E. Ginna Nuclear Power Plant", dated December 1979. The post accident integrated gamma doses outside containment presented in Ginna UFSAR Table 3.11-1 were derived from Table 5-1 of the above report, and represent an accident duration of 6 months.

On February 25, 2005, via Amendment No. 87 to the Renewed Facility Operating License, NRC approved the implementation of Alternative Source Terms as outlined in 10CFR50.67, SRP 15.0.1 and Regulatory Guide 1.183 for post accident dose assessments associated with the site boundary, and on-site locations that require continuous occupancy such as the Control Room at Ginna Station. The source terms used to establish post-accident radiation and vital area access environments were not impacted by this Amendment.

Ginna UFSAR Figure 6.1-1 presents the accident temperature profile for equipment qualification inside the containment. Ginna UFSAR Figure 6.1-2 provides the associated pressure profile. The peak values are 286°F and 60 psig. Material compatibility with the post accident chemical environment is discussed in detail in Ginna UFSAR section 6.1.2.1.

As described in Ginna UFSAR section 3.11.3.1.2, the temperature associated with the main steam line break is higher than the large break LOCA but was determined by the NRC not to be limiting for qualification of equipment required following a main steam line break because:

- The high temperature transient is very brief and there is super-heated steam (with a lower heat transfer capability) as opposed to saturated steam.
- The equipment is protected from the direct effects of the steam line break by concrete floors and shields.
- The sensitive portions of the electrical equipment are not directly exposed to the environment but are protected by housing, cable jackets, and the like.

As described in Ginna UFSAR section 3.1.1.4.5, the components of the protection system are qualified such that the mechanical and thermal adverse environment resulting from emergency situations during which the components are required to function does not prevent them from accomplishing their safety function.

As described in Ginna UFSAR section 3.11.4, complete and auditable records which include supporting documentation for environmental qualification of safety related electrical equipment are maintained. The documentation includes test results, specifications, reports and other data that has been identified by references in the RG&E reports to the NRC on the environmental qualification program.

As described in these Ginna UFSAR sections, Ginna Station received a post-construction review as part of the Systematic Evaluation Program. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program (IPSAR), R.E. Ginna Nuclear Power Plant.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's systems and components have been evaluated for plant license renewal. Plant systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, the environmental qualification of electrical equipment is covered in the Time-Limited Aging Analysis Section 4.4. The License Renewal Safety Evaluation Report concurs with the Ginna approach to the time-limited aging analysis for environmental qualification of electrical equipment, including the acceptance of the use of the actual plant temperature data. The EPU time-limited aging analysis for equipment qualification is discussed in section 2.3.1.2.2.

The application for license renewal addressed the Environmental Qualification Program in Appendix B, Section B3.1. The Environmental Qualification Program is identified as a time limited aging analysis for the purpose of license renewal. The Environmental Qualification Program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10CFR50.49 (f). Preventive actions of the Environmental Qualification Program are (a) establishing service conditions and (b) requiring specific installation, inspection, monitoring and periodic maintenance actions to

maintain the aging effects within the bounds of the qualification basis. Environmental parameter monitoring is used to ensure the components are within the qualification basis.

Aging detection is not an Environmental Qualification Program function but monitoring of the environmental conditions such as temperature in the immediate vicinity of the component may be performed to maintain the qualification basis. Monitoring of component time of installation, operation mode and environmental conditions is performed to ensure the selected components are within their qualification basis. Corrective actions are made in accordance with the requirements of 10CFR50 Appendix B as described in the Ginna Quality Assurance Program for Station Operation.

Identification of unexpected adverse conditions during operation or maintenance that could affect the component would result in evaluation and appropriate corrective actions, such as changes to the qualification basis and conclusions.

Administrative controls are used to implement the Environmental Qualification Program through the station policy, directives, and procedures in accordance with the station's quality assurance program. Qualification files are maintained at the plant site in auditable form and controlled by the quality assurance program.

2.3.1.2 Technical Evaluation

2.3.1.2.1 Introduction

Safety-related structures, systems and components at Ginna Station are designed for environmental events as described in Ginna UFSAR sections 3.10, and 3.11. Ginna UFSAR section 3.10 provides details regarding seismic qualification of safety-related mechanical, structural, instrumentation and electrical equipment. Ginna UFSAR section 3.11 provides details regarding environmental qualification of safety-related electrical equipment.

As described in Ginna UFSAR section 3.11.2, the master list of equipment that is necessary to achieve or support (1) emergency reactor shutdown, (2) containment isolation, (3) reactor core cooling, (4) containment heat removal, (5) core residual heat removal, and (6) prevention of significant release of radioactive material to the environment is contained in a plant procedure. This master equipment list includes the equipment to which the EQ requirements of 10CFR50.49 are applied.

The pre-EPU environmental conditions are presented in Ginna UFSAR section 3.11.3 and Table 3.11-I.

The components in the Ginna Environmental Qualification Program have been evaluated for 60 year license operation as reflected in the License Renewal SER, NUREG-1786.

2.3.1.2.2 Description of Analyses and Evaluations

Description of Analyses

The environmental qualification of electrical equipment is performed for the components identified in the Environmental Qualification Master Equipment List. The equipment qualification parameters are compared to the EPU parameter values to demonstrate the continued qualification of the equipment under proposed EPU conditions.

The environmental parameters for both normal operation (including anticipated operational occurrences) and design basis accidents are temperature, pressure, radiation dose, and humidity, and in addition only for accidents are spray chemistry and submergence. The EPU values for these parameters are discussed below and in other sections of this report. The transient temperatures associated with anticipated operational occurrences, such as turbine trips or the loss of a ventilation system, could increase slightly as a result of EPU but will have no impact on the equipment qualification. The time weighted average operational temperature for the buildings, if it were to be made over 60 years, would not be changed by the short temperature spikes associated with these anticipated transients and the time weighted average temperature would be bounded with margin by the building normal design temperature used for the equipment qualification. The peak temperature values for the design basis accidents bound the temperature transients of the anticipated operational occurrences.

The time-limited aging analysis performed for the license renewal has been included in the EPU evaluation. Equipment that has a qualified life greater than 40 years is subject to this time-limited aging analysis. Two considerations for the time-limited aging analysis were taken. The analysis performed for license renewal was used to identify the normal operating temperature for which the equipment is qualified for 60 years (or qualified for sufficient years to reach the end of the license). Further analysis was performed using the Arrhenius equation to calculate a temperature resulting in a 60 year qualified life. Equipment with time-limited aging analysis temperatures that are close to or lower than the building design temperature were identified for area temperature monitoring during EPU operation to confirm the validity of the time-limited aging analysis.

Core power uprate will typically increase the radioactivity level in the core by the percentage of the uprate. The radiation source terms in equipment / structures containing radioactive fluids, and the corresponding radiation zone doses, will increase for the uprate. In addition, factors that impact the equilibrium core inventory and consequently, the estimated dose, are fuel enrichment and burnup. These additional changes could result in activity levels in the core that are higher than predicted by the core power ratio associated with the uprate.

The radiation dose rate currently utilized to determine the normal operation component of the total integrated dose used for radiological environmental qualification is provided in Ginna UFSAR Table 3.11-1. These values are based on radiological surveys

performed in early 1980, and correspond to a core power level of 1520 MWt and a historical one-year fuel cycle length.

The impact of EPU on the normal operation radiation environment used for equipment qualification is determined by using recent survey data which reflects Ginna operation with an 18 month fuel cycle, and by utilizing the assessment provided in LR section 2.10.1.2 regarding impact of the EPU on normal operation radiation levels. The above assessment is used to verify or update the dose rate values currently provided in Ginna UFSAR Table 3.11-1. These dose rates are used to develop the normal operation component of the total integrated dose supporting equipment qualification following EPU. The normal operation dose values assume operation at the EPU power level for the duration of 60 years that includes the license renewal.

The impact of EPU on the post-accident gamma and beta environmental dose estimates provided for the environmental zones listed in Ginna UFSAR section 3.11 is evaluated utilizing scaling factors that are based on a comparison of the accident source terms developed based on the original core inventory used to develop the post-LOCA environmental doses, to the corresponding accident source terms developed based on the EPU core inventory. Since the relative abundance and the average gamma energy of each isotope are the key parameters that affect direct exposure, using a scaling factor that addresses the change in these parameters is sufficient to assess the radiological impact of EPU and fuel cycle length.

The referenced EPU dose scaling factors are based on TID 14844 source terms and applied to current post-LOCA environmental dose estimates listed in Ginna UFSAR Tables 3.11-1 through 3.11-3, to establish the EPU environmental levels. It is noted that although Ginna Station has recently been approved for the implementation of alternative source terms for post accident dose assessments associated with the site boundary and control room, the EPU assessment supporting equipment qualification is based on TID 14844 source terms. This approach is acceptable based on Section 1.3.5 of Regulatory Guide 1.183 which indicates that though equipment qualification analyses impacted by plant modifications should be updated to address the impacts, no plant modification is required to address the impact of the difference in source term characteristics (i.e., AST vs. TID 14844) on environmental doses.

The estimated increase in radiation levels reflect, in addition to the EPU power level, a) the change in fuel cycle length, and b) the use of current computer codes, methodology and nuclear data in developing the uprated core inventory (vs. the methodology, computer tools and nuclear data used in the development of the original licensing basis core inventory) and c) a factor of 1.04 to address uncertainty for variations in fuel management schemes. As a result, the calculated EPU dose scaling factor values are higher than the core power ratio.

Inside Containment: The EPU gamma and beta dose scaling factors inside containment have three components, a) a component to address the increase in core power level from 1550 MWT (the existing values include a 2% margin for power uncertainty) to the

analyzed core power level of 1811 MWt, b) a component of 1.04 to address the NSSS uncertainty for variations in fuel management schemes, and c) a component to update the current values that are based on a 12 month fuel cycle to an 18 month fuel cycle. Ginna Station is currently operating with an 18 month fuel cycle.

To determine the gamma and beta dose scaling factors to address the change in fuel cycle length, a 12 month and an 18 month fuel cycle were generated using the ORIGEN-S computer code and parameters reasonably representative of Ginna. Core average enrichments were used. Based on Regulatory Guide 1.89, Rev 1, Appendix D (which was the basis of the pre-EPU doses inside containment), an enrichment of 3.0 w % was used for the 12-month cycle. The EPU fuel enrichment of 4.752 w% was used for the 18 month fuel cycle. The core activities were utilized to develop integrated doses vs. time from a finite cloud model. The fuel cycle length scaling factors were the ratio of the finite cloud gamma and beta dose based on the 18-month fuel cycle to the 12-month fuel cycle. No depletion other than decay was considered.

The noble gas and iodine doses and associated scaling factors were developed separately. This was done to account for the fact that sprays and plateout are credited in Regulatory Guide 1.89, and therefore inherently considered in the current Ginna inside containment dose estimates. The fuel cycle length gamma and beta dose scaling factors associated with the iodines and noble gases were therefore applied to the fraction of the total gamma and beta dose attributable to iodines and noble gases, respectively, as documented in Tables D1 and D2 of Regulatory Guide 1.89. The factor increase in the 1 year integrated dose inside containment due to the EPU, the change in fuel cycle length, the margin incorporated for variations in fuel management schemes, and the differences due to updated computer codes was estimated at 1.2 and 1.37, for gamma and beta, respectively.

Outside Containment: The pre-EPU core inventory documented in the "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces / Systems which may be used in Post Accident Operations Outside Containment at R. E. Ginna Nuclear Power Plant", dated December 1979, and the EPU core inventory were used to develop the post-LOCA gamma energy release rates (MeV/sec) per energy group vs. time, for containment atmosphere, sump water, and pressurized recirculating fluid.

For the "unshielded" case, the factor impact on post-accident integrated gamma doses was estimated by ratioing the gamma energy release weighted by the flux to dose rate conversion factor, as a function of time, for the EPU power level, to the corresponding weighted source terms based on the pre-EPU power level. To address the fact that outside containment the sources are contained, the "unshielded" values included the shielding effect of a pipe wall thickness associated with a 2-inch nominal diameter pipe. This insured that the results were not skewed by photons at energies less than 25 keV which will be substantially attenuated by any piping sources.

To evaluate the impact of EPU on post-LOCA gamma doses (vs. time) in areas that are "shielded," the pre-EPU as well as EPU source terms discussed above were weighted

by the concrete reduction factors for each energy group. The concrete reduction factors for 1 and 3 feet of concrete, (developed in the EPU analysis), were used to provide a basis for comparison of the post LOCA spectrum hardness of source terms, with respect to time, for both original design and EPU cases, for lightly shielded and heavily shielded cases.

Since the EPU gamma dose scaling factors varied with source, time, as well as shielding, to cover all types of analysis models/assessments, the maximum dose scaling factor of 1.32 developed from all of the above assessments, was used for all source/receptor combinations, with or without shields, and at all time periods after LOCA.

The qualified life of the equipment is defined as the longest time the equipment can remain in normal operation and still be able to perform its intended safety function during and following an accident. The qualified life of equipment is a function mainly of the temperature and in some cases of the radiation dose environment. The evaluation of the qualified life for the EPU compares the normal operation temperature basis pre-EPU with the temperature for EPU operation. If the temperature for EPU operation does not change from pre-EPU, the qualified life of the equipment is not impacted. For equipment whose qualified life is based on specific local temperature data, the data will be verified by temperature monitoring during EPU operation.

The evaluation compared the pre-EPU temperature basis to the EPU temperature. The qualified life on certain types of equipment such as cable is for the life of the plant. This equipment had been re-evaluated for the 60 year license extension. The area temperature that allows 60 years of operation (i.e., for the equipment to last to the end of the license) has been identified for comparison to EPU area temperatures.

Design basis accident conditions for equipment qualification inside the containment are the result of the loss of coolant accident. Refer to LR section 2.6.1, Primary Containment Functional Design. The loss-of-coolant temperature and pressure vs. time profiles for EPU conditions are compared on graphs to the temperature and pressure profiles that are the pre-EPU basis for equipment qualification. The results of overlaying the profiles show that the EPU temperature and pressure conditions are bounded.

The post accident operability time is also reviewed using the EPU temperature vs. time profile overlaid on the pre-EPU accident qualification profile. The results of overlaying the profiles show whether the EPU conditions are bounded or whether additional steps are necessary to assess the qualification to the EPU accident conditions. It was determined that the EPU accident temperature at 24 hours is marginally not bounded, but becomes bounded in about one day (see Figure 1). An analysis, using the Arrhenius equation, to convert the EPU temperature profile to an equivalent temperature for a spectrum of activation energies was compared to an analysis for the qualification accident temperature profile. This showed margin in the qualification profile over the EPU profile and therefore verified the post accident operability time of the equipment.

Radiation dose qualification is based on the sum of the normal operational dose plus the accident dose. As described in section 2.3.1.2.2, the EPU operation and other factors increased the total integrated dose to components inside the containment and to certain components outside the containment. Inside the containment, the dose from airborne activity includes beta radiation. The evaluation of the equipment qualification compared the EPU total integrated dose to the component qualification dose. The margin between the qualification and the EPU dose was determined. The total accident dose included both the gamma and beta contributions. If the radiation qualification dose did not bound the total center line gamma plus beta dose, the beta dose was reduced using:

- the shielding provided by the equipment casing, cable conduit or cable jacket utilizing the DOR guidelines that state "...the conservative beta surface dose of 1.40×10^8 RADS reported in Appendix D of NUREG-0588 would be reduced by approximately a factor of ten within 30 mils of the surface of electric cable insulation of unit density. An additional 40 mils of insulation (total of 70 mils) results in another factor of 10 reduction in dose", and/or

- a reduction factor, as applicable, for the finite airborne beta source in a junction box or an enclosed volume. The finite source volume beta dose reduction factor was calculated based on beta energy spectra of noble gases and iodines in the post-LOCA containment, and the Loevinger's empirical beta dose formula.

Environmental parameters of humidity, spray water chemistry and submergence for EPU operation were confirmed to be within the qualification values of the equipment.

Evaluations

Inside Containment

Temperature - normal operation: The normal operating temperature for equipment qualified life is the design temperature of 120°F. This does not change for EPU operation. Therefore, the qualified life of equipment that is based on this temperature is not changed by the EPU operation. Refer to LR section 2.7.7, Other Ventilation Systems (Containment).

However, for certain components, the EQ documentation for the 60 year qualified life for the license renewal indicates credit is taken for local area temperatures that are less than the design bulk air temperature. The Ginna local area temperature data has provided the justification for the lower (or higher) temperatures used in this equipment qualification documentation. Components that may have their documented qualified life shortened due to local area temperatures being impacted by the EPU operation are:

- TransAmerica DeLaval Gems Sump B level switch
- Conax Buffalo reactor vessel level RTD

- Brand Rex cable
- BIW coax and triax cable
- NAMCO limit switches on PORVs
- Penetration pigtail cables on Crouse Hinds penetrations

The Crouse-Hinds penetration extension pigtails, for example, have a documented qualified life of 52.5 years at a service temperature of 120°F. The qualified life can be extended to 60 years with a temperature of 119.5°F. Temperature monitoring at the penetrations is needed to determine the EPU local area temperature.

Additional monitoring of these local area temperatures during EPU operation will identify any impacts. These components have qualified lives of 60 years. Components with qualified lives less than 40 years (established before the license extension), based on the building design temperature, are already in the maintenance program. Their replacement schedules are not impacted since the design temperature is not changed by EPU.

Pressure - normal operation: The pressure in the containment of 0 psig is not changed by EPU operation.

Radiation dose - normal operation: See containment total integrated dose discussed below.

Humidity – normal operation: The humidity of 50% is not changed by EPU.

Temperature – accident: The accident temperature inside the containment, as shown in Figure 1, is bounded by the pre-EPU accident temperature profile used for the equipment qualification with the exception of the initial conditions (irrelevant due to short duration) and the 24 hour temperature for post accident operability time. Refer to LR section 2.6.1, Primary Containment Functional Design, for a discussion of the containment LOCA analysis.

The peak accident temperature for the EPU operation is bounded by the peak equipment qualification temperature. The equipment is therefore qualified for the EPU accident transient conditions.

The EPU temperature at 24 hours is greater than the post accident operability time temperature. The accident temperature drops below the qualification profile. An aging equivalency analysis was performed for a range of activation energies. There is 40% greater aging using the equipment qualification profile than the EPU accident profile. Therefore, the components inside the containment continue to be qualified for the post accident operability time.

Pressure - accident: As shown in Figure 2, the accident peak pressure is bounded by the pre-EPU qualification pressure.

Figure 1
Containment Accident Temperature Profile

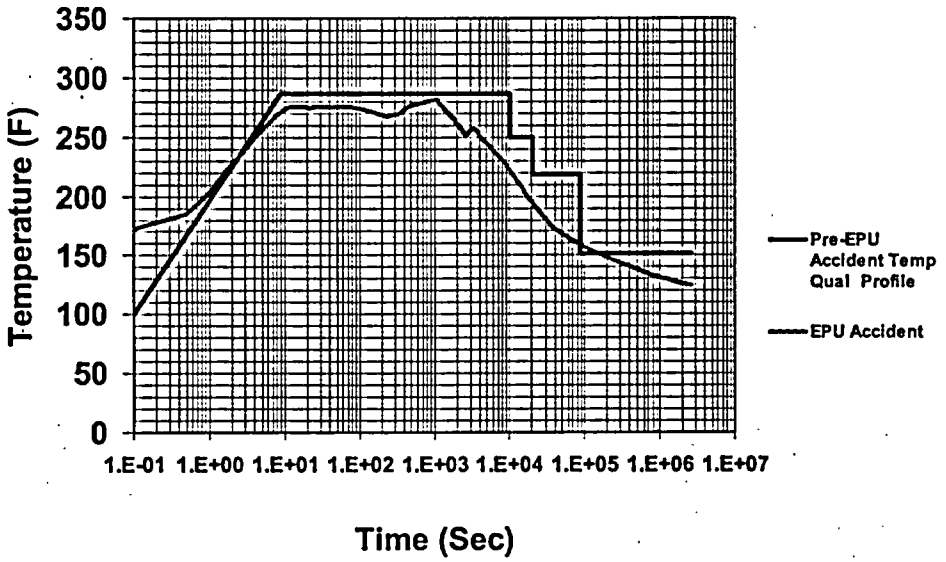
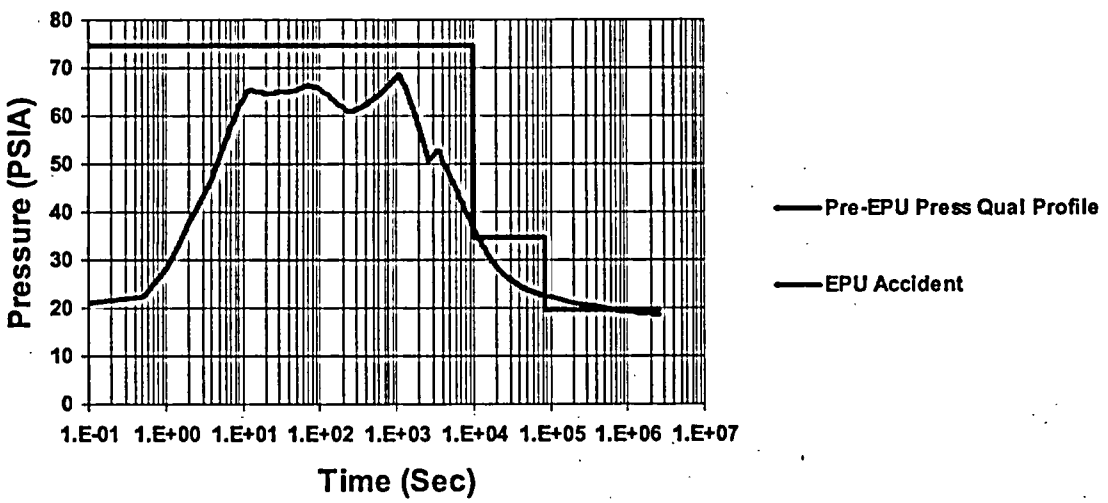


Figure 2
EPU Accident Pressure Profile



Note: Curves on Figures 1 and 2 are differentiated by color on the CD version of the Licensing Report; for differentiation in black & white, curves with right angles are the Pre-EPU profiles.

Radiation – total integrated dose: The radiation total integrated dose for the 60 years license renewal, 18 month fuel cycle and one year accident duration under EPU operation has increased to 5.3E5 Rads normal gamma, 1.77E7 Rads accident gamma

and 2.91E8 Rads accident beta in containment general areas. The beta dose has been evaluated with respect to the component's inherent shielding (equipment sealed case or cable jacket) and/or its configuration that limits beta exposure. All equipment has been determined to remain qualified, with the following exception.

- The localized dose from the HEPA filters adjacent to the containment recirculation fan cooler motors and the associated cables is currently being evaluated. The total integrated dose is greater than the qualification dose for the motor.

These qualification issues will be resolved prior to EPU implementation by performing additional evaluations, qualifications or modifications.

Humidity – accident: The accident humidity of 100% is not impacted by EPU operation.

Spray chemistry – accident: The accident spray chemistry maximum pH of 10.25 is not exceeded by EPU operation. The boron concentration increases in the refueling water storage tank and the accumulators result in a slightly lower minimum pH in the spray and sump following an accident, but above the 7.0 requirement.

Submergence – accident: The submergence depth of 7 feet above the basement floor is not impacted by the EPU operation. The only system volume change is the accumulator active volume which increases by 1 cubic foot. The total volume increase of 2 cubic feet from both accumulators does not change the depth measurably.

Intermediate Building

Temperature - normal operation: The normal operating temperature for equipment qualified life is a continuous design temperature of 104°F. The temperature increase for EPU operation is approximately 0.3 °F. This does not impact the qualified life of the equipment in the intermediate building, since operating temperatures are below 104°F most of the time at the equipment location.

Pressure - normal operation: The pressure in the intermediate building for equipment qualification of 0 psig is not changed by EPU operation.

Radiation dose - normal operation: See intermediate building total integrated dose below.

Humidity – normal operation: The humidity of 60% is not changed by EPU.

Temperature – accident: The accident temperature in the intermediate building of 212°F for the 6 inch main steam line break used for the equipment qualification is not changed by EPU operation. The only equipment that is required to function in the harsh environment created by this HELB is the reactor trip switchgear. This equipment

performs its function in the very early stages of the temperature transient. Long term superheated conditions are not applicable for its qualification.

Pressure – accident: The pressure in the intermediate building for equipment qualification of 0.25 psig is not changed by EPU operation.

Radiation – total integrated dose: The normal radiation dose in the intermediate building is negligible. The accident dose increases due to the increased dose in the containment following EPU. The area where the dose increase is greatest is near the containment penetrations. The equipment located in this area which is required for long term operation is qualified to radiation doses higher than its pre-EPU qualification requirements. Therefore, it is not impacted by the increased dose in the intermediate building due to EPU operation.

Humidity- accident: The accident humidity of 100% in the intermediate building is not changed by EPU operation.

Submergence – accident: The water level in the intermediate building is not changed by EPU operation.

Auxiliary Building

Temperature - normal operation: The normal operating temperature for equipment qualified life is the design temperature of 104°F. This does not change for EPU operation. Refer to LR section 2.7.5, Auxiliary and Radwaste Area and Turbine Building.

Pressure - normal operation: The pressure in the auxiliary building for equipment qualification of 0 psig is not changed by EPU operation.

Radiation dose - normal operation: See auxiliary building total integrated dose below.

Humidity – normal operation: The humidity of 60% is not changed by EPU.

Temperature – accident: The accident temperature in the auxiliary building of 150°F for the HELBs is not changed by EPU operation.

Pressure – accident: The pressure in the auxiliary building for equipment qualification of 0.1 psig is not changed by EPU operation.

Radiation – total integrated dose: The normal radiation dose in the auxiliary building general areas is 1.3E4 Rads. The bounding dose near RHR piping is 6.3E4 rads. The total post accident 6 month integrated dose in the auxiliary building increases because of the increase in the dose of containment sump water being recirculated by pumps in the auxiliary building. At elevation 271 feet the dose is 1.32E2 Rads, for elevation 253 feet the dose is 1.19E3 Rads, for elevation 236 feet general area the dose is <1E4 Rads, for elevation 236 feet in contact with pumps and piping recirculating LOCA sump water the

dose is 3.7E6 Rads and 10 feet from the pipe the dose is 6.6E4 Rads. The equipment impacted by these dose increases remains qualified.

Humidity- accident: The accident humidity of 100% in the auxiliary building is not changed by EPU operation.

Submergence – accident: The water level in the auxiliary building of 8.2 inches is not changed by EPU operation. Refer to LR section 2.5.1.3, Pipe Failures.

Turbine Building

Temperature - normal operation: The normal operating temperature for equipment qualified life is the design temperature of 104°F. Refer to LR section 2.7.5, Auxiliary and Radwaste Area and Turbine Building. The building actual temperature increases by approximately 0.1°F for EPU operation. This is an insignificant impact with respect to the building design temperature.

Pressure - normal operation: The pressure in the turbine building of 0 psig is not changed by EPU operation.

Radiation dose - normal operation: None.

Humidity – normal operation: The humidity of 60% is not changed by EPU.

Temperature – accident: The accident temperature in the turbine building of 220°F for 30 minutes is not changed by EPU operation. Refer to LR section 2.5.1.3, Pipe Failures.

Pressure – accident: The pressure in the turbine building of 1.14 psig is not changed by EPU operation.

Radiation – total integrated dose: Negligible since there are no radiation sources.

Humidity- accident: The accident humidity of 100% in the turbine building is not changed by EPU operation.

Submergence – accident: The water level of 18 inches in the turbine building is not changed by EPU operation.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

With respect to the licensing renewal described in NUREG-1786, EPU activities do not add any new components, any new or previously unevaluated materials, nor introduce any new functions for existing components that would change the license renewal

system evaluation boundaries. Thus, no new aging effects requiring Aging Management Review (AMR) are identified. The potential effect of slightly higher normal ambient temperatures will be addressed by the Ginna local temperature monitoring program as described earlier in this subsection.

2.3.1.2.3 Results

Radiation dose values increase for EPU operation. All EQ equipment has been evaluated and is qualified for the increased total integrated dose both inside and outside containment with the following exception. The localized dose from the HEPA filters adjacent to the recirculation fan cooler motors and the associated cable is being developed. This qualification issue will be resolved prior to EPU implementation by performing additional evaluation, qualification or modification.

The containment accident temperature profile is bounded by the existing equipment qualification profile with the exception of the long term post accident operability time period. However, no revision to the equipment qualification profile is needed;, because the pre-EPU profile is a more conservative integrated energy condition than the EPU accident analysis. All equipment inside containment is qualified for the accident temperatures resulting from operation under EPU conditions.

The HELBs, which are the bases for equipment qualification outside containment, do not change as a result of EPU operation. All equipment outside containment is qualified for the appropriate EPU accident temperatures.

The qualified life of the equipment is not changed by EPU operation when the building design temperature is used. For equipment that has the qualified life based on local area temperatures, there could be some temperature increases due to normal operation at EPU power levels. Local area temperature data during EPU operation is needed to confirm these 60 year qualified life evaluations for license renewal. The data taken during EPU operation should be compared with pre-EPU temperature data. Any increases must be factored into the qualified life evaluations.

The time-limited aging analysis of the environmental qualification of electrical equipment remains valid for EPU radiation and temperature conditions.

2.3.1.3 Conclusions

Ginna has performed an assessment of the effects of the proposed EPU on the environmental qualification of electrical equipment and concludes that it has adequately addressed the effects of the proposed EPU on the environmental conditions for, and the qualification of, electrical equipment. Ginna further concludes that the equipment will continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR50.49 following implementation of the proposed EPU. Ginna finds the proposed EPU acceptable with respect to the environmental qualification of the electrical equipment.

2.3.2 Offsite Power System

2.3.2.1 Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the descriptive information, analyses, and referenced documents for the offsite power system, as well as the stability studies for the electrical transmission grid. The Ginna review focused on whether the loss of the nuclear unit, the largest operating generating facility on the grid, or the most critical transmission line will result in the loss-of-offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC-17.Specific review criteria are contained in SRP sections 8.1 and 8.2, and Appendix A to SRP section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station Offsite Power System design relative to conformance to:

- GDC-17 is described in the Ginna UFSAR section 3.1.2.2.8, General Design Criterion 17 - Electrical Power Systems. SEP Topic VII-3, Systems Required for Safe Shutdown provided additional verification that the offsite power system conforms to the intent of GDC-17.Additional details that define the licensing basis are described in Ginna UFSAR section 8.2.

In addition to the evaluations described in the UFSAR, the offsite power system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, portions of the offsite power system were determined to be within the scope of the license renewal and components subject to age management review are evaluated on a plant wide basis as commodities, where the generic commodity groups are described in SER section 2.5 and age management is described in SER section 3.6.

2.3.2.2 Technical Evaluation

2.3.2.2.1 Introduction

The offsite power system and its components are discussed in the Ginna UFSAR section 8.2. The system consists of station 13A (115 kV switchyard, including transformer 6 and voltage regulator 767) that provides interconnection to the transmission grid, generator step up transformer, four 115 kV underground oil pipe cables, station auxiliary transformers 12A and 12B, and circuit 767 and 751 cables.

Grid stability studies were performed to evaluate the impact of Ginna station on the reliability of the local 115 kV and New York Independent System Operator (NYISO) bulk power systems.

The function of the 115 kV switchyard, which is arranged in a breaker-and-a-half configuration, is to interconnect the station output to the transmission grid, and provide one (circuit 767) of two independent offsite power sources to the station auxiliary transformer 12B. The second independent offsite power source (circuit 751) is provided from the 34.5 kV switchyard station 204 to station auxiliary transformer 12A.

The function of the three-phase, two winding generator step up transformer is to provide a means to transmit the generator output power to the switchyard by stepping up the generator voltage from 19 kV to the switchyard voltage of 115 kV. Also, a back up offsite power supply can be provided by back feeding from station 13A (115 kV switchyard) through the generator step up transformer with the isolated phase bus flexible links disconnected from the generator.

The function of the four oil pipe cables is to deliver the station output power from the generator step up transformer to station 13A (115 kV switchyard).

The function of the two separate three-phase, three winding station auxiliary transformers 12A and 12B is to step down the offsite voltage from 34.5 kV to 4.16 kV and power the engineered safeguards auxiliary loads during normal operation. In addition, during start up, shutdown and loss of station generating capacity, the function is to supply all auxiliary loads when station unit transformer 11 is not available.

2.3.2.2.2 Description of Analyses and Evaluations

The offsite power system and its components were evaluated to ensure they are capable of performing their intended function at EPU conditions.The evaluation is based on the system's required design functions and attributes, and upon a comparison between the existing equipment ratings and the anticipated operating requirements at EPU conditions. In addition, grid stability is evaluated and Ginna Station current licensing basis is assessed with respect to the requirements of GDC-17.

2.3.2.2.3 Results

2.3.2.2.3.1 Grid Stability

The transmission system is discussed in Ginna UFSAR section 8.2.2.1.A System Reliability Impact Study was performed to evaluate the impact of the Ginna Station EPU on the reliability of the local 115 kV and NYISO bulk power systems. The study was performed in accordance with the NYISO requirements, including the New York State Reliability Council's Reliability Rules for Planning and Operating the New York Bulk Power System, the Northeast Power Coordinating Council's Basic Criteria for Design and Operation of Interconnected Power Systems, and the North American Electric Reliability Council's Planning Standards.

Thermal, voltage, stability, and short-circuit results, with and without the Ginna Station at EPU, were compared to determine any detrimental impact of the proposed EPU. Thermal, voltage, and stability analyses were performed on the summer cases and thermal analysis was performed on the winter cases. Pre-contingency and contingencies were evaluated with load flow analysis for each seasonal condition. This analysis involved an extensive examination of contingencies of local and cross-state transmission facilities located around the Ginna plant area.

Several analyses were performed in the study, including thermal, voltage, stability and short circuit cases. Extreme contingencies (i.e. loss of entire generation units) were also evaluated and found to be acceptable. There is essentially no voltage variation between the base case and EPU case. Key results provided in the study are as follows:

- The thermal analysis concluded there would be no significant overloading of the transmission system. In most cases where an overload was projected, the EPU case resulted in lower levels than the existing base case. NYISO agrees that the identified overloads can be mitigated through various techniques.
- The voltage analysis also concluded that in most cases the voltages at EPU are improved over the base case. Violations of voltage levels have no adverse impact on the system's voltage. Degraded voltage and loss of voltage relay settings are addressed in LR section 2.3.3, AC Onsite Power System.
- The study showed that the system will remain stable during stuck breaker scenarios and tower outages.

- Short circuit current magnitudes and breaker clearing times were shown to be acceptable.
- In the case of extreme contingency analysis, the EPU project will have no adverse impact on the system.

The results of this study indicate that the thermal, voltage, and stability performance is not degraded by implementation of the EPU. In addition, the Ginna Station EPU does not compromise the interrupting capabilities of RG&E system equipment.

2.3.2.2.3.2 Offsite Power System Components

Station 13A (115 kV switchyard)

The 115 kV oil circuit breakers located in station 13A are discussed in Ginna UFSAR section 8.2.1.1.3.The equipment has been evaluated for EPU conditions. The 115 kV oil circuit breakers 1G13A72 and 9X13A72, their associated 115 kV disconnect switches and other circuit components (including current transformers and bushings which are integral to the circuit breakers) will be subjected to anticipated currents exceeding their continuous rating at full EPU power levels. This occurs only when one of these two output circuit breakers is out of service (open), resulting in the full EPU output being carried by the other circuit breaker. If these circuit breakers and associated components are not replaced to support maximum EPU, then operational restrictions will be put in place to limit loading on the 115 kV circuit breakers to within their 3000 amp rating. Voltage regulator 767, located in the 115 kV switchyard, ensures acceptable offsite voltage is provided to station auxiliary transformer 12B.Under worst case EPU loading conditions on voltage regulator 767, the total calculated load of 28.95 MVA is greater than its 26.67 MVA (Forced Air) maximum rating. The voltage regulator loading was evaluated and it was determined that actual loads are lower than calculated and do not exceed its maximum rating.

Transformer 6, also located in the 115 kV switchyard, steps down the 115 kV switchyard voltage to 34.5 kV and connects to voltage regulator 767.Under worst case EPU loading conditions on transformer 6, the total load of 29.04 MVA is enveloped by its 30 MVA (Oil & Air)/56 MVA (Forced Oil & Air) rating.

All other 115 kV equipment that was evaluated proved to be acceptable at EPU conditions.

Generator Step Up Transformer

The generator step up transformer is discussed in Ginna UFSAR sections 8.2.1.1.1 and 8.2.2.2.1.The equipment has been evaluated for EPU conditions. The evaluation confirms that the existing generator step up transformer design rating at 65° C is inadequate to support unit operation at EPU conditions. As a result a transformer study

was performed and it was determined modifications are required to upgrade the transformer rating from 616 MVA to 680 MVA at 65° C, which envelops the anticipated worst-case generator step up transformer loading at EPU conditions. Refer to Table 2.2.3-1 for worst-case loading and rating comparison. The required modifications include replacement of all 3 high voltage bushings and replacement of all 4 of the transformer coolers and cooler oil pumps, and the addition of a fifth cooler/pump unit for back up capability. The modifications will be implemented prior to EPU.

Operating at EPU does not affect the capability of back feeding from the 115 kV switchyard through the generator step up transformer and station unit transformer 11 after flexible links at the generator terminals are removed.

The existing 4000/5 amp rated, multi-ratio current transformers located on the generator step up transformer high voltage bushings require a tap connection change for operation at EPU. The current transformers are presently connected to the 3000/5 amp tap and provide differential relay protection for the oil pipe cables.

Oil Pipe Cables

The oil pipe cables are discussed in Ginna UFSAR sections 8.2.1.1.1 and 8.2.1.1.3.The equipment has been evaluated for EPU conditions. The evaluation indicates that the existing static oil pipe cables' design rating is exceeded by the cable loading required at EPU.A study to evaluate the general condition and capacity of the four 115 kV pipe cables with oil circulating was performed. The study determined that the continuous current rating of the oil pipe cables can be increased with oil circulating and will bound the maximum ampacity required at EPU conditions. This is true under normal operation with the station unit transformer supplying auxiliary loads, or with only the station auxiliary transformers supplying station loads.

The oilstatic cable is protected by a differential current relay scheme with inputs from current transformers (CTs) located at the generator step-up transformer and generator output breakers. The highest tap setting on the generator step-up CTs is 4000/5 and the highest tap setting on the output breaker CTs is 3000/5. The 4000/5 setting on the generator step-up encompasses the maximum plant output current at EPU. The 3000/5 CTs are below the maximum plant output current; however, output is normally split between the two output breakers. Ginna is evaluating options to modify the differential current relay protection for operation at EPU as well as operational actions to limit unit output to 3000 amps for specific scenarios as discussed in the 115 KV switchyard section.

Station Auxiliary Transformers

Station auxiliary transformers 12 A and 12B are discussed in Ginna UFSAR section 8.2.1.2.The evaluation confirms that the station auxiliary transformer 12 A and 12B design rating of 41.8 MVA at 65° C is adequate to support unit operation at EPU conditions. The calculated worst-case transformer loading is shown in LR Tables 2.3.2-2

and 2.3.2-3.In addition, circuits 751 and 767, which provide separate offsite power sources from stations 204 and 13A to station auxiliary transformers 12A and 12B, respectively, have sufficient capacity at EPU conditions.

EPU evaluations have determined that following the modifications identified above, the offsite power system will continue to have sufficient capacity and capability to supply power to all safety loads and other preferred operating equipment. Two separate and independent offsite power sources continue to be maintained in accordance with the Ginna Station current licensing basis with respect to the requirements of GDC-17. If the 3000 amp generator output circuit breakers (1G13A72 and 9X13A72) are not replaced, then to minimize the possibility of Ginna Station initiating a grid disturbance when only one output circuit breaker is aligned, compensatory action to limit generator load and MVARS to within the single breaker design rating will be imposed.

License Renewal

As discussed above, portions of the offsite power system are within the scope of license renewal. However, the changes associated with operating the offsite system at EPU conditions do not add any new or previously unevaluated materials to the system, nor require any changes to the Aging Management Program. No new aging effects requiring management are identified.

2.3.2.3 Conclusion

The evaluation concludes that the offsite power system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-17 following implementation of the proposed EPU. This conclusion considers the effect of modifications to the generator step up transformer cooling system, differential relaying current transformers and voltage regulator 767.There is adequate physical and electrical separation and the offsite power system has the capacity and capability to supply power to all safety loads as well as other preferred operating equipment. The impact of the proposed EPU does not degrade grid stability. Therefore, the proposed EPU is acceptable with respect to the offsite power system.

Table 2.3.2-1 GSU Transformer Output loading No Station Loads Supplied From SUT							
Generator Output			GSU Output Load			GSU Uprated Rating	Notes
MW	MVAR	MVA	MW	MVAR	MVA	MVA	
Unit Operating at Lagging Power Factor (Exporting VARs)							
613.5	261	666.7	612.3	207.4	646.5	607/680 @55°C/65°C (FOA)	1,2,3
Unit Operating at Leading Power Factor (Importing VARs)							
613.5	-140	629.3	612.3	189.8	641.0	607/680 @55°C/65°C (FOA)	1,2,3

Notes for Table 2.3.2-1:
1. The main generator output at EPU is provided from LR section 2.3.3
2. The GSU output loading is derived from load flow/voltage profile analysis.
3. The GSU uprated rating is after specified transformer modifications.

Table 2.3.2-2 SAT-12A Maximum Input Loading on the H Winding					
Primary Winding	H- Winding			Rating	Reference
	MW	MVAR	MVA	MVA	
Existing	24.46	10.20	26.50	37.3@55°C 41.8@65°C (FA)	Note 1
EPU+Existing	26.03	11.06	28.28		Note 2
Increment	1.57	0.86	1.78		Note 3
Total Input Load	**26.03**	**11.06**	**28.28**	**41.8@65°C**	

Notes for Table 2.3.2-2 :
1. Existing loading is derived from load flow/voltage profile analysis. These values represent the present calculated loading on the transformer prior to EPU.
2. EPU+Existing loading is derived from load flow/voltage profile analysis. These values represent the total calculated loading on the transformer after EPU.
3. Increment loading is the difference between the Existing loading (Note 1) and EPU + Existing loading (Note 2). These values represent the additional loading on the transformer as a result of EPU.

Table 2.3.2-3 SAT-12B Maximum Input Loading on the H Winding					
Primary Winding	**H- Winding**			**Rating**	**Reference**
	MW	**MVAR**	**MVA**	**MVA**	
Existing	24.44	10.29	26.52	37.3@55°C 41.8@65°C (FA)	Note 1
EPU+Existing	26.03	11.10	28.30		Note 2
Increment	1.59	0.81	1.78		Note 3
Total Input Load	**26.03**	**11.10**	**28.30**	**41.8@65°C**	

Notes for Table 2.3.2-3:

1. Existing loading is derived from load flow/voltage profile analysis. These values represent the present calculated loading on the transformer prior to EPU.
2. EPU+Existing loading is derived from load flow/voltage profile analysis. These values represent the total calculated loading on the transformer after EPU.
3. Increment loading is the difference between the Existing loading (Note 1) and EPU + Existing loading (Note 2). These values represent the additional loading on the transformer as a result of EPU.

2.3.3 AC Onsite Power System

2.3.3.1 Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems that supply power to safety-related equipment. The Ginna Nuclear Power Plant, LLC's (Ginna) review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station AC Onsite Power System design relative to conformance to:

- GDC-17 is described in the Ginna UFSAR section 3.1.2.2.8, General Design Criterion 17 - Electrical Power Systems. Additional details that define the licensing basis are described in Ginna UFSAR sections 8.1 and 8.3.

The following NRC Safety Evaluation Reports (SERs) and other documents provide additional details that define the licensing basis as described in the Ginna UFSAR with respect to the ac onsite power system and GDC-17:

- Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Station Blackout Safety Evaluation, dated January 30, 1992.
- Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Station Blackout Rule Supplemental Safety Evaluation, dated September 23, 1992.
- Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topics VI-7.F, VII-3, VII-6, and VIII-2, Safety Evaluations for Ginna, dated June 24, 1981.

- Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topic VIII-1.A, Potential Equipment Failures Associated with Degraded Grid Voltage, dated January 29, 1982.
- Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Amendment 38 to Provisional Operating License DPR-18, dated March 26, 1981.

In addition to the evaluations described in the UFSAR, the ac onsite power system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the ac onsite power system was determined to be within the scope of the license renewal and components subject to age management review are evaluated on a plant wide basis as commodities, where the generic commodity groups are described in SER Section 2.5 and age management is described in SER Section 3.6.

2.3.3.2 Technical Evaluation

2.3.3.2.1 Introduction

The ac onsite power system and its components are discussed in the Ginna UFSAR section 8.3. The ac onsite power system consists of station unit transformer 11, the 4160 V, 480 V, 120 V systems (including rectifier/inverters and constant voltage transformers), emergency diesel generators, associated buses, cables, electrical penetrations (where applicable), circuit breakers and protective relays. In addition, the main generator and isolated phase bus duct are included in the ac onsite power system evaluations.

The function of the three-phase, three-winding station unit transformer 11 is to provide power from the main generator output stepped down from 19 kV to 4.16 kV to supply the onsite ac electrical distribution system normal loads during normal power operation.

The function of the 4160 V system is to supply power directly to non-safety related 4.16 kV loads and through step-down transformers to safety and non-safety related loads.

The function of the 480 V system is to step down the 4.16 kV to 480 V to supply non-safety and safety related buses, and through rectifiers/inverters to supply 120 V ac instrumentation and direct current (dc) controls. The normal supply to non-safety related buses 13 and 15 is from buses 11A and 11B, which are powered from station unit transformer 11. Power for safety related buses 14, 16, 17 and 18 is from buses 12A and 12B, which are connected to the station auxiliary transformers 12A and 12B.

The function of the 120 V system is to provide power to vital controls and instrumentation loads.

The function of the two emergency diesel generators is to provide emergency power to the Class 1E 480 V safeguards buses 14, 16, 17 and 18 to safely shutdown the reactor and maintain it in a safe condition for any accident coincident with loss of offsite power.

The function of the main generator is to provide a means of converting the mechanical energy of the main turbine into a supply of regulated and usable electricity. The generator output is delivered at 19 kV to the generator step up transformer and station unit transformer 11 through isolated phase bus ducts.

The function of the isolated phase bus duct is to conduct electrical power from the main generator to the generator step up transformer and station unit transformer 11.

2.3.3.2.2 Description of Analyses and Evaluations

The ac onsite power system and its components were evaluated to ensure they are capable of performing their intended function at EPU conditions. The evaluation is based on the system's required design functions and attributes, and upon a comparison between the existing equipment ratings and the anticipated operating requirements at EPU conditions. The EPU required that equipment operate at service conditions different than currently evaluated. To determine the impact of EPU operation on the ac onsite power system, a baseline for bus loading was developed to represent the existing plant loading conditions. New load flow/voltage profile and short circuit current analyses that include load changes as a result of EPU conditions were performed. The results of these analyses are used to ensure that the systems/equipment are capable of performing their intended functions and form the bases for the ac onsite power system evaluations.

2.3.3.2.3 Results

Station Unit Transformer

The station unit transformer is discussed in Ginna UFSAR section 8.3.1.1.2. The calculated worst-case transformer loading is shown in Table 2.3.3-1. The evaluation confirms that the existing station unit transformer 11 design rating of 37.3 MVA at 55° C and 41.8 MVA at 65° C is adequate to support unit operation at EPU conditions.

4160 V system

The 4160 V system is discussed in Ginna UFSAR sections 8.3.1.1.3. Evaluation of the 4160 V system and affected motors, confirms the following:

Switchgear Buses and Circuit Breakers, and Nonsegregated Phase Bus Duct.

- The calculated worst case continuous current for each 4160 V switchgear bus, incoming circuit breaker and nonsegregated phase bus duct, during maximum full load at EPU conditions, is less than the equipment design ratings, as indicated in Table 2.3.3-2. Therefore, the EPU loading requirements of switchgear buses, incoming circuit breakers and nonsegregated phase bus ducts are bounded by equipment design ratings.
- The calculated full load current for each affected motor during maximum full load at EPU conditions is less than the feeder circuit breaker design rating. Therefore, the EPU loading requirements for motor feeder breakers are bounded by equipment design ratings.
- The calculated short circuit current (interrupting and momentary) for the 4160 V switchgear buses and circuit breakers during maximum full load at EPU conditions is less than the equipment short circuit ratings, as indicated in Table 2.3.3-3. Therefore, the EPU fault duty requirements of switchgear buses and breakers are bounded by equipment design ratings.

System Voltage Level

Motor terminal voltage for running 4 kV motors at EPU conditions during steady state maximum full load conditions is calculated to be above the minimum required voltage. In addition, the worst-case maximum terminal voltage on the 4 kV motors is below the maximum allowable. Table 2.3.3-4 shows the existing and EPU minimum and maximum calculated steady state voltages on the affected motors.

4 kV Motor Load Requirements

The condensate pump, condensate booster pump, main feedwater pump, heater drain pump, and reactor coolant pump motors are affected by station operation at EPU conditions. The evaluation of the reactor coolant pump motors for operation during hot-loop and cold-loop at EPU conditions is provided in LR section 2.2.2.6. The evaluation determined that the new motor load requirements are within the 6000 hp nameplate rating for hot-loop and within the motor service factor rating for cold loop operation. The increased load for the other affected 4 kV motors at EPU conditions are discussed in LR section 2.5.5.4, Condensate and Feedwater.

The calculated full-load current values for the affected motors at EPU conditions, and the derated ampacities for associated motor feeder cables and electrical penetrations (as applicable) are provided in Table 2.3.3-5. The table indicates that for each affected motor, the calculated motor current at EPU conditions does not exceed the derated ampacities of motor feeder cable, except for the main feedwater pumps motors. The main feedwater pump motors and feeder cables are being replaced as a result of the modification to the main feedwater pumps (LR section 2.5.5.4, Condensate and Feedwater). The reactor cooling pump motor feeder cables have been evaluated with acceptable results.

Existing protective relay settings of the affected motors have been reviewed at EPU operating conditions and determined to be unaffected except for the condensate booster pump and main feedwater pump motors. These pump motors are being replaced and revised settings will be implemented prior to EPU (LR section 2.5.5.4, Condensate and Feedwater).

480 V system

The 480 V system is discussed in Ginna UFSAR sections 8.3.1.1.4. Evaluation of the 480 V system confirms the following:

The 480 V station service transformers are not adversely affected by EPU conditions because the loading is bounded by the equipment design ratings. Transformers 14 and 16 have a minor excursion into an overload condition during specific design basis accident conditions. The transformer loading will decrease below nameplate rating within a short period of time resulting in no adverse effect to the transformers.

Switchgear buses, incoming circuit breakers, motor feeder circuit breakers, and fault duty on switchgear buses and breakers at EPU conditions are not adversely affected by EPU conditions because the loading is bounded by the equipment design ratings. A short circuit current case exists that occurs only during testing of the emergency diesel generators when connected in parallel with the switchgear buses. Special precautions are in place for this short-time testing condition.

Motor Control Centers

The motor control centers are not adversely affected because there are no load changes on motor control centers at EPU conditions. Therefore, motor control center continuous current, and short circuit current on buses and circuit breakers at EPU conditions are within the equipment design ratings, except for an existing short circuit current case that occurs only during testing of the emergency diesel generators when connected in parallel with the motor control center buses. Special precautions are in place for this short-time testing condition.

System Voltage Level

The 480 V voltage levels are affected at EPU conditions on switchgear buses 13, 14, 15, 16, 17 and 18, as indicated in Table 2.3.3-6. To ensure adequate voltages to support equipment operation and function at EPU conditions, station auxiliary transformer 12A requires a tap setting change to increase voltage to the four Class 1E 480 V buses 14, 16, 17 and 18. This change resolves a transient voltage condition, as indicated in Table 2.3.3-6A, and ensures there is no effect on the loss of voltage relay setpoints. The results of the evaluation demonstrates that there is no adverse impact on the 4160 V system voltages as a result of this change and there is no impact to the degraded voltage relay setpoints.

480 V Motor Load Requirements

The service water pumps, component cooling water pumps, and auxiliary feedwater pumps motors are not affected by station operation at EPU conditions as discussed in LR section 2.5.4.2, Station Service Water System, LR section 2.5.4.3, Component Cooling Water System, and LR section 2.5.4.5, Auxiliary Feedwater System. There is no increase in loading on these motors due to EPU conditions. The two standby auxiliary feedwater pump motors will operate slightly above their nameplate rating of 300 hp for a short period of time. The maximum operating point at EPU will be 320 hp (107% of nameplate), which is within the 1.15 service factor. This system is normally not operating (in standby mode). When required to operate, the motors will perform their required function at the anticipated load, and there should be no significant impact on the motor design life.

120 V system

The 120V system is discussed in Ginna UFSAR sections 8.3.1.1.5. The evaluation determined that no new components requiring power from the four 120 V ac instrument buses is required to support EPU. Consequently, operation at EPU conditions does not result in load changes or equipment changes to the 120 V ac instrumentation buses and distribution system.

Emergency Diesel Generators

The emergency diesel generators are discussed in Ginna UFSAR section 8.3.1.1.6. Review of the loads for operation at the EPU conditions indicates that there are no load additions or modifications required to the existing 1950 kW (continuous rating) emergency diesel generators. Therefore, there is no impact to the existing emergency diesel generator loading analysis and their acceptability for EPU operation. No emergency diesel generator modifications are required to support EPU operation. Refer to LR section 2.5.7.1 for the Emergency Diesel Engine Fuel Oil Storage and Transfer System.

Main Generator

The main generator is discussed in Ginna UFSAR sections 8.1.3 and 10.2.2. The main generator existing rating is 608.4 MVA @ 0.85 power factor. In order to support unit operation at EPU conditions, a generator uprate study was performed and as a result, the generator rating will be revised to 667 MVA @ 0.9 power factor. The increased generator rating is adequate to support unit operation at EPU, including machine leading and lagging reactive power requirements as indicated in Table 2.3.3-7. Improved generator cooling is required and will be accomplished with the hydrogen cooler system modifications. Refer to LR section 2.5.1.2.2, Turbine Generator, for discussion on the cooler modifications.

Isolated Phase Bus Duct

The isolated phase bus duct is discussed in Ginna UFSAR sections 8.1.1 and 8.1.3. The existing isolated phase bus duct main bus continuous current design rating is 20 kA forced cooled. As a result of the EPU evaluations, the isolated phase bus duct main bus will be upgraded to 21.35 kA, which bounds unit operation at worst-case EPU loading conditions, as indicated in Table 2.3.3-8. The only change needed to accomplish the upgrade to 21.35 kA is to increase the main bus forced cooling which will be accomplished before EPU. . In addition, the evaluation of the isolated phase bus tap bus confirms that its continuous current rating envelops the anticipated worst-case bus loading at EPU conditions. Also, the evaluation indicates that the isolated phase main and tap bus short circuit design ratings envelop the available fault current levels at EPU conditions.

GDC-17 Requirements

The load flow/voltage profile analysis and short circuit current analysis indicate that the ac onsite power system equipment voltages and fault duties are not adversely affected by EPU conditions, when powered from the offsite power sources (station unit and station auxiliary transformers). Also, the loading requirements of switchgear buses and circuit breakers, nonsegregated phase bus ducts, unit substation transformers, and motor control centers are bounded by equipment design ratings. Therefore, the ac onsite power system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-17, and perform their intended functions during all plant operating and accident conditions, following implementation of the proposed EPU.

2.3.3.3 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

As discussed above, the onsite power system is within the scope of license renewal. However, the changes associated with operating the onsite power system at EPU conditions do not add any new or previously unevaluated materials to the system. No new aging effects requiring management are identified.

2.3.3.4 Conclusion

The evaluation concludes that the ac onsite power system adequately accounts for the effects of the proposed EPU on the systems functional design. It is further concluded that the ac onsite power system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-17 following implementation of the proposed EPU, including the following modifications:

- Replace the main feedwater pump motors, feeder cables, and change protective relay settings

- Replace the condensate booster pump motors and change protective relay settings
- Complete generator nameplate uprate and isolated phase bus duct uprate
- Change tap setting on station service transformer 12A to increase station bus voltages
- Upgrade isolated phase bus duct cooling

Based on the evaluation at EPU conditions, there is adequate physical and electrical separation and the onsite power system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the proposed EPU is acceptable with respect to the onsite power system after completion of the above modifications.

Table 2.3.3-1 Station Unit Transformer 11 Maximum Input Loading on the H Winding					
Primary Winding	**H- Winding**			**Max Rating**	**Note**
	MW	**MVAR**	**MVA**	**MVA**	
Existing	25.44	10.63	27.57	37.3@55°C	Note 1
EPU+Existing	26.58	11.26	28.87	41.8@65°C	Note 2
Increment	1.14	0.63	1.30	(FA)	Note 3
Total Input Load	**26.58**	**11.26**	**28.87**	**41.8 @65°C**	

Notes for Table 2.3.3-1:

1. Existing loading is derived from load flow/voltage profile analysis. These values represent the present calculated loading on the transformer prior to EPU.
2. EPU+Existing loading is derived from load flow/voltage profile analysis. These values represent the total calculated loading on the transformer after EPU.
3. Increment loading is the difference between the Existing loading (Note 1) and EPU + Existing loading (Note 2). These values represent the additional loading on the transformer as a result of EPU.

Table 2.3.3-2 Switchgear Bus Continuous Current Loading			
Bus	**Existing Load Amps**	**EPU Load Amps**	**Rating Amps**
	(Note 1)	(Note 1)	(Note 2)
11A	2043	2118	3000
11B	1892	1972	3000
12A	1943	2084	3000
12B	1819	1960	3000

Notes for Table 2.3.3-2:

1. Worst-case Existing and EPU calculated loading is taken from load flow/voltage profile analysis computer runs.
2. Switchgear buses, incoming circuit breakers and non-segregated bus ratings are all 3000 amps.

Table 2.3.3-3 Short Circuit Current at the 4160V Switchgear Buses Short Circuit Current, kA						
	Interrupting, Sym (Note 2)			Momentary, Asym		
Bus	Existing Calculated Duty Note 1	EPU Calculated Duty Note 1	Circuit Breaker Rating	Existing Calculated Duty Note 1	EPU Calculated Duty Note 1	Circuit Breaker Rating
11A	43.2	43.7	45.7	71.2	72.5	80.0
11B	42.0	42.5	45.7	68.8	70.1	80.0
12A	37.7	38.3	45.7	65.5	67.1	80.0
12B	38.3	38.8	45.7	66.1	67.6	80.0
12BX	29.4	29.3	44.7	68.9	70.3	78.0
12BY	29.6	29.6	44.7	70.1	71.5	78.0
12AX	26.1	26.2	44.7	62.4	63.7	78.0
12AY	25.9	26.0	44.7	65.2	66.7	78.0

Notes for Table 2.3.3-3:

1. The calculated interrupting and momentary short circuit current values are taken from worst-case short circuit analysis computer runs.
2. The calculated interrupting duty and the ratings of the circuit breakers are adjusted for pre-fault voltage of 1.05 per unit.

Table 2.3.3-4
Worst Case Running Motor Voltages
Existing and EPU Conditions (Steady State and Maximum Motor Terminal Voltage)

Motors	Bus	Rated Voltage	Minimum Steady State Voltage			Maximum Steady State Voltage			
			Existing Calc.Mtr. Term.Volts (Note 1)	EPU Calc.Mtr. Term.Volts (Note 1)	Min.Mtr.Term Volts Required. (Note 2)	Existing Calc.Mtr. Term.Volts (Note 1)	EPU Calc.Mtr. Term.Volts (Note 1)	EPU Calc.Mtr. Term.Volts (Note 3)	Max. Allow. Voltage (Note 2)
Aux.Bldg Exh.Fan 1A	11A	4000	4018	4012	3600	4358	4347	4347	4400
Aux.Bldg Exh.Fan 1B	11B	4000	4026	4027	3600	4365	4354	4354	4400
Cond.Booster PP A	12A	4000	4118	4118	3600	4343	4334	4386	4400
Cond.Booster PP B	11B	4000	4028	4063	3600	4366	4356	4355	4400
Cond.Booster PP C	11A	4000	4020	4055	3600	4359	4348	4348	4400
Condensate PP A	11A	4000	4017	4011	3600	4357	4346	4346	4400
Condensate PP B	11B	4000	4025	4027	3600	4364	4353	4353	4400
Condensate PP C	11A	4000	4018	4012	3600	4357	4347	4347	4400
Circ.Water PP 1A	11A	4000	4014	4007	3600	4355	4343	4344	4400
Circ.Water PP 1B	11B	4000	4023	4023	3600	4362	4351	4351	4400
Heater Drain PP A	11A	4000	4019	4055	3600	4359	4348	4348	4400
Heater Drain PP B	11B	4000	4027	4063	3600	4366	4356	4355	4400
Main Fdwr PP 1A	11A	4000	4013	4008	3600	4353	4342	4342	4400
Main Fdwr PP 1B	11B	4000	4022	4024	3600	4361	4350	4350	4400
React.Cool. PP 1A	11A	4000	4013	4006	3600	4353	4342	4342	4400
React.Cool. PP 1B	11B	4000	4019	4020	3600	4358	4347	4347	4400

Notes for Table 2.3.3-4:
1. The calculated voltages for Existing and EPU voltage values are taken from worst case load flow/voltage profile analysis computer runs.
2. Motors are rated for continuous operation at +/- 10% of rated voltage.
3. Motor terminal voltage after changing the high side tap position on SAT 12A (Refer to 480 V system voltage evaluation).

Table 2.3.3-5 4V Motor Loads and Feeder Cable Ratings Existing and EPU							
Affected Motors	**Rated Voltage**	**Existing BHP**	**(EPU) BHP**	**Rated HP**	**Existing Load (Amps)** (Note 2)	**EPU Load (Amps)** (Note 2)	**Cable Rating Amps**
Condensate PP CPA, B, C	4000	1175	1164	1250	148	145	210
Cond. Booster PP CBPA, B, C	4000	308	456	350 (Ex) 500 (EPU)	38	 58	158
Main Feedwater PP MFWP1A, B	4000	4770	5066	4500 (Ex) 5500 (EPU)	588	 612	566
Heater Drain PP HDTPA HDTPB	4000	 408 400	 371 371	400	 51 53	 46 49	 158 158
Reactor Coolant PP RCP1A, B	4000	5400 (Hot)	5620	6000	682	706	775/753 (Note 1)

Notes for Table 2.3.3-5:

1. The outside containment feeder cable is 2-3C 500MCM rated for 775 amps and the inside containment feeder cable is 2-750MCM rated for 753 amps.
2. The calculated load current for Existing and EPU conditions are taken from worst-case load flow/voltage profile analysis computer runs

Table 2.3.3-6 Worst-Case 480 Bus Voltages Existing and EPU Conditions Prior to Station Auxiliary Transformer 12A Tap Setting Change									
Bus	Min.Continuous Volt.			Min.Transient Volt.			Maximum Volt.		
	Existing (Note 1)	EPU (Note 1)	Accept. (Note 2)	Existing (Note 1)	EPU (Note 1)	Accept. (Note 4)	Existing (Note 1)	EPU (Note 1)	Accept. (Note 3)
Bus 13	442.0	441.3	430.5	371.7	366.0	377.2	483.1	481.9	506
Vital Bus 14	444.8	444.2	430.5	379.6	374.9	377.2	497.6	496.5	506
Bus15	443.1	443.3	430.5	370.7	364.8	377.2	484.1	482.8	506
Vital Bus 16	447.1	446.4	430.5	378.8	374.0	377.2	498.3	497.6	506
Vital Bus 17	455.3	451.3	430.5	380.8	374.8	377.2	496.6	495.7	506
Vital Bus 18	453.5	449.5	430.5	382.0	376.2	377.2	496.2	495.0	506

Notes for Table 2.3.3-6:

1. The calculated voltage for Existing and EPU conditions are taken from worst-case load flow/voltage profile analysis computer runs.
2. The acceptable minimum continuous voltage is based on the degraded voltage relay's reset voltage.
3. The acceptable maximum voltage is based on motors continuous operation at +10% of rated voltage (i.e., 110% of 460 V).
4. The acceptable transient voltage is based on the loss of voltage relay maximum dropout voltage. These values are prior to tap change on SAT 12A as stated in LR section 2.3.3.2.3 of this report.

Table 2.3.3-6A Worst-Case 480 V Bus Voltages Existing and EPU Conditions After Station Auxiliary Transformer 12A Tap Setting Change						
	Min.Transient Volt.			Maximum Volt.		
Bus	Existing	EPU (Note 1)	Accept. (Note 2)	Existing	EPU (Note 1)	Accept. (Note 3)
Bus 13	371.7	378.4	377.2	483.1	482.2	506
Vital Bus 14	379.6	386.8	377.2	497.6	502.5	506
Bus 15	370.7	377.2	377.2	484.1	483.2	506
Vital Bus 16	378.8	385.9	377.2	498.3	503.7	506
Vital Bus 17	380.8	386.9	377.2	496.6	501.9	506
Vital Bus 18	382.0	388.2	377.2	496.2	501.0	506

Notes for Table 2.3.3-6A:

1. The calculated voltage for EPU conditions are taken from worst-case load flow/voltage profile analysis computer runs after station auxiliary transformer 12A tap setting change from 0 to -2.5%.
2. The acceptable transient voltage is based on the loss of voltage relay maximum dropout voltage.
3. The acceptable maximum voltage is based on motors continuous operation at +10% of rated voltage (i.e., 110% of 460 V).

Table 2.3.3-7 Generator Output				
Generator Operation at Lagging Power Factor				
MW (Note 3)	**MVAR (Note 4)**	**MVA (Note 1)**	**Volts, kV**	**PF(%) (Note 2)**
613.5	261	666.7	19	92.0
Generator Operation at Leading Power Factor				
MW	**MVAR**	**MVA**	**Volts, kV**	**PF(%)**
613.5	-140	629.3	19	97.5

Notes for Table 2.3.3- 7:

1. MVA = $(MW^2 + MVAR^2)^{1/2}$
2. Power Factor (PF), % = $\frac{MW}{MVA} x 100$
3. Maximum required generator MW output at EPU
4. NYISO reactive power requirements

Table 2.3.3-8 IPB Main Bus Loading				
Generator Output (EPU)				**IPB Main Bus Load (kA) Note 1**
MVA	**MW**	**MVAR**	**Voltage (p.u.)**	
Unit Operating at Lagging Power Factor (Exporting VARs)				
666.7	613.5	261	0.95	21.33
Unit Operating at Leading Power Factor (Importing VARs)				
629.3	613.5	-140	0.95	20.13

Notes for Table 2.3.3-8:

1. Loading at EPU conditions is derived from load flow/voltage profile analysis. The existing design rating of the isolated phase bus duct, main bus is 20 kA (forced cooled). Revised rating will be 21.35 kA forced cooled after modifications.

2.3.4 DC Onsite Power System

2.3.4.1 Regulatory Evaluation

The direct current (dc) onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that supply motive or control power to safety-related equipment. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP, Sections 8.1 and 8.3.2.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station DC Onsite Power Systems design relative to conformance to:

- GDC-17 is described in the Ginna UFSAR section 3.1.2.2.8, General Design Criterion 17 - Electrical Power Systems. Additional details that define the licensing basis are described in Ginna UFSAR section 8.3.2. Additionally, in Generic Letter 91-06, dated April 29, 1991, the NRC staff identified actions to be taken by the licensees related to Generic Issue A-30, Adequacy of Safety Related DC Power Supplies. Rochester Gas and Electric (RG&E) Corporation responded with detail information in its letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: Resolution of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, dated October 28, 1991. The NRC action was completed upon its transmittal letter to RG&E from A.R. Johnson, NRC, to R.C. Mecredy, RG&E: Subject: Closure of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, dated June 21, 1993, indicating that RG&E's responses satisfied the reporting requirements of the generic letter.

In addition to the evaluations described in the Ginna UFSAR, the dc onsite power system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The above SER, determined the dc onsite power system to be within the scope of the license renewal. Components subject to age management review are evaluated on a plant wide basis as commodities, with the generic commodity groups described in SER Section 2.5.

2.3.4.2 Technical Evaluation

2.3.4.2.1 Introduction

The 125 V dc system is discussed in the Ginna UFSAR section 8.3.2. The 125 V dc system consists of two batteries with two chargers serving each battery and dc buses. The train A engineered safety features equipment is supplied from the 'A' battery while the train B engineered safety features equipment is supplied from the 'B' battery. The function of the batteries is to provide the sources of power for control, emergency lighting, and the inverters for critical 60 cycle instrument power. In addition, there is a technical support center battery that provides power to the plant process computer as well as the turbine emergency bearing oil pump, airside seal oil backup pump, circulating water discharge valves, 4kV breaker test cabinet, and anticipated-transient-without-scram mitigation system actuation circuitry inverter. The technical support center battery is capable of supplying both safeguards dc trains in the event of an emergency. The 125 V dc system provides the battery capacity to cope with Station Blackout and 10CFR50 Appendix R conditions.

2.3.4.2.2 Description of Analyses and Evaluations

The 125 V dc system and its components were evaluated to ensure they are capable of performing their intended function at EPU conditions. The evaluation is based on the system's required design functions and attributes, and upon a comparison between the existing dc equipment ratings and the anticipated operating requirements at EPU conditions. Station Blackout and 10CFR50 Appendix R program evaluations are included in this evaluation.

2.3.4.2.3 Results

The safety-related and non safety-related portions of the 125 V dc system were evaluated to determine potential impacts due to EPU.

Plant modifications associated with the safety-related feedwater system isolation valves operators and non safety-related condensate booster pump required for EPU will result in a slight increase in dc system loading. With the addition of two solenoids for the feedwater system and three digital ammeters associated with the condensate booster pumps as a result of EPU, loads increased on Train A by 0.376 amps and 0.329 amps on Train B. As a percentage of dc loads, these increases are insignificant, especially when it is considered that the calculated load amps used in the evaluation conservatively exceed the measured amps by approximately 23% for Battery A and approximately 30% for Battery B. The existing 125 V dc system power and

control configurations are essentially unaffected by EPU. The new loads are supplied from fused circuits, therefore, they will not introduce any new failure modes or effects into the dc system. Therefore, the battery duty cycle, voltages at equipment, and available fault currents are unaffected and remain within the existing design bases, as documented in the existing calculations. No other changes will affect the dc system. Condensate and feedwater system modifications are discussed in LR section 2.5.5.4.

The 125 V dc system continues to have the capability and capacity because the additional loads are insignificant. Separate and independent station battery systems are maintained to supply power to all safety loads in accordance with the Ginna Stations current licensing basis with respect to the requirements of GDC 17.

In addition, Station Blackout and 10CFR50 Appendix R program evaluations did not result in any 125 V dc load changes as discussed in LR section 2.3.5 and 2.5.1.4.

2.3.4.3 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

As discussed above, the dc onsite power system is within the scope of license renewal. However, the changes associated with operating the dc system at EPU conditions do not add any new or previously unevaluated materials to the system or exceed the operating or environmental parameters previously evaluated for equipment included within the scope of the rule. Thus, no new aging effects requiring management are identified.

2.3.4.4 Conclusions

The evaluation concluded that the dc system will continue to function as designed and continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC 17 following implementation of the proposed EPU. Adequate physical and electrical separation exist, and the dc system has the capacity and capability to supply power to all safety loads and other required equipment at EPU conditions.

2.3.5 Station Blackout

2.3.5.1 Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency ac power system. Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources." The Ginna Nuclear Power Plant, LLC's (Ginna) review focused on the impact of the proposed EPU on the plant's ability to cope with and recover from a station blackout event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for station blackout are based on 10CFR50.63. Specific review criteria are contained in Standard Review Plan, Section 8.1, and Appendix B to Standard Review Plan, Section 8.2. Other guidance is provided in Matrix 3 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to 10CFR50.63, "Loss of All Alternating Current Power," is addressed in Ginna UFSAR sections 8.1.4.4, "Potential Risk of Station Blackout," and 8.1.4.5, "Station Blackout Program."

As addressed in Ginna UFSAR section 8.1.4.4, the Ginna Station was evaluated against the requirements of the Station Blackout Rule, 10CFR50.63 using guidance from NUMARC 87-00, and Regulatory Guide 1.155, except for the analyses that determined the effects of loss of ventilation, where plant-specific analyses were used. Using the guidance of NUMARC 87-00, the Ginna station blackout coping duration was determined to be 4 hours.

As addressed in the Ginna UFSAR section 8.1.4.5, a station blackout program has been developed. The Ginna Station blackout program manual is a comprehensive document

that presents the history, regulatory commitments, documentation and calculation references, bases, procedure changes, and modifications that were implemented to reduce the risk of consequences during a station blackout. Contained in the SBO Program manual is the documentation required to substantiate the Station's submittals to the NRC pursuant to 10CFR50.63. Regulatory commitments for station blackout are also listed in the SBO Program manual with an implementation summary.

As addressed in Ginna UFSAR section 8.1.4.4, the station blackout rule requires that the following issues be addressed: station blackout duration, condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment isolation, reactor coolant inventory, procedures and training, quality assurance and Technical Specifications, and the emergency diesel generator reliability program. The NRC Safety Evaluation (Reference: Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, "R. E. Ginna Nuclear Power Plant Station Blackout Analysis (TAC M68548)," January 30, 1992) and supplemental Safety Evaluation (Reference: Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, "R. E. Ginna Nuclear Power Plant, Station Blackout Rule (10CFR50.63), Supplemental Safety Evaluation (TAC M68548), September 23, 1992) for SBO concluded that the following SBO issues were acceptably resolved: SBO coping duration, condensate inventory for decay heat removal, Class 1E battery capacity, containment isolation, and reactor coolant inventory.

As addressed in Ginna UFSAR section 8.1.4.5.1, station blackout assumptions as presented in NUMARC 87-00 stipulate that following the loss of all offsite power the reactor is assumed to automatically trip with sufficient shutdown margin to maintain subcriticality at Mode 3 (Hot Shutdown) or Mode 4 (Hot Standby). Based on the Ginna configuration, an automatic reactor trip will not necessarily result from the loss of offsite power. However, Ginna Station emergency procedures specify operator action to manually trip the reactor in the event of loss of all AC power.

In addition to the evaluations described in the UFSAR, station blackout coping equipment was addressed as part of License Renewal as documented in:

- License Renewal Safety Evaluation Report (SER) for the R. E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, station blackout is discussed in sections 1.4 (Interim Staff Guidance), 2.5.1.5.2 (Switchyard Bus), 3.4.2.4.1 (Main and Auxiliary Steam), 3.4.2.4.3 (Auxiliary Feedwater), and 3.5.2.4.2 (Other Structures).

2.3.5.2 Technical Evaluation

2.3.5.2.1 Introduction

As addressed in the Ginna UFSAR section 8.1.4.4, the station blackout rule requires the following issues be addressed: station blackout duration, condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment isolation, reactor coolant inventory, procedures and training, quality assurance, Technical Specifications, and the emergency diesel generator reliability program. A summary of the status of the following SBO issues for current plant conditions is provided, since these are potentially affected by the EPU: condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment isolation, reactor coolant inventory, plant procedures and training, and auxiliary feedwater system flow requirements for SBO.

Condensate Inventory for Decay Heat Removal

The required condensate inventory, at the current licensed power level of 1520 MWt, for decay heat removal and plant cooldown was determined to be 48,239 gallons. Since this required inventory exceeds the minimum usable volume of water in the condensate storage tanks as specified in the plant's Technical Specifications, a backup source of condensate is required. A design analysis showed that the fire water system is able to supply the condensate storage tanks at a flow rate which will satisfy the condensate storage volume requirement for a 4 hour station blackout coping period, given an initial volume in the condensate storage tanks at the Technical Specification minimum.

Class 1E Battery Capacity

As stated in Ginna UFSAR section 8.3.2.2, each of the two station batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 108.6 V. A design analysis demonstrates that the two 60 cell, lead-acid, 1495 amp-hour vital batteries A and B have sufficient capacity for the 4 hour station blackout coping duration.

Compressed Air

During a station blackout, station air can be supplied by a portable diesel-driven air compressor. However, in the event of loss of all plant air, the following air-operated valves required for decay heat removal during a station blackout have sufficient backup supplies or can be operated manually:

- Turbine-driven auxiliary feedwater pump flow control valves: These valves are normally open and fail as is upon a loss of compressed air. In the event that all plant air is lost, they can be locally operated via a handwheel. The "Loss of All AC Power Procedure" includes the following actions for control of auxiliary feedwater flow if the turbine-driven auxiliary feedwater pump flow control valves

cannot be operated remotely: (1) control flow by throttling the auxiliary feedwater turbine-driven DC-powered pump discharge valve, (2) locally throttle the flow control valves and (3) control flow by starting and stopping the turbine-driven auxiliary feedwater pump.

- Main steam atmospheric relief valves: These valves are normally supplied by instrument air. On loss of instrument air, a backup supply consisting of two nitrogen systems contained in six bottles each, automatically supplies the motive force for operation of these valves from the control room. The nitrogen supply has been sized to provide 8 hours of atmospheric relief valve operation in the event of loss of instrument air. The "Loss of All AC Power Procedure" requires that atmospheric relief valve nitrogen pressure be monitored during a station blackout event. The atmospheric relief valves can also be locally operated via a hand wheel.

The remaining air-operated valves important to safety fail in the safe position.

Effects of Loss of Ventilation

As addressed in Ginna UFSAR section 8.1.4.5.2, the dominant areas of concern for loss of ventilation were determined to be the areas near the atmospheric relief valves and the turbine driven auxiliary feedwater pump in the intermediate building. The control room, battery rooms, and the relay room were also identified as containing station blackout coping equipment. The following conservative maximum temperatures for a 4 hour coping period were determined:

- Atmospheric relief valve area, provided the doors from the area to the turbine building are opened: 179°F
- Turbine-driven auxiliary feedwater pump area, provided the double doors from the area to the turbine building are opened within 30 minutes of event initiation: 145°F
- Control room, provided door from the room to the turbine building and control equipment cabinet doors are opened within 30 minutes: 116°F
- Battery rooms: 108.2°F
- Relay Room: 103°F

Due to operator safety concerns regarding habitability in the atmospheric relief valve area of the intermediate building, the "Loss of All AC Power Procedure" contains the following caution: "Due to potentially extreme environmental conditions, caution should be used when entering the intermediate building for local actions." Personal protective equipment (ice vests) is available for use by operators in performing local valve operations in the intermediate building and is regularly inspected.

Containment Isolation

An evaluation was performed confirming that appropriate containment integrity can be provided during an SBO event, where "appropriate containment integrity" is defined as providing the capability for valve position indication and closure of certain containment isolation valves independent of the preferred or Class 1E power supplies. The Ginna Station Blackout Program Plan identifies the containment isolation valves reviewed and documents justification for their exclusion from consideration based on NUMARC 87-00 criteria. The following valves, which are motor-operated and normally closed, do not meet any of the above-listed criteria and therefore have been added to the step in the "Loss of All AC Power Procedure" which verifies containment isolation:

- 850A - RHR pump 1A suction from Sump B
- 850B - RHR pump 1B suction from Sump B

Reactor Coolant Inventory

As addressed in Ginna UFSAR section 8.1.4.5.1, sources of reactor coolant system leakage during a station blackout at Ginna are presumed to include normal system leakage (11 gpm) and reactor coolant pump seal leakage (25 gpm / pump), for a total leakage of 61 gpm. Under these conditions, the reactor core will remain covered with the reactor coolant inventory for the 4 hour duration of a station blackout.

Plant Procedures and Training

An emergency operating procedure, "Loss of All AC Power," provides the sequence and instructions for opening doors to areas containing SBO coping equipment, aligning backup cooling water to the turbine driven auxiliary feedwater pump, and initiating makeup water to the condensate storage tanks. A separate procedure addresses detailed operator actions for alignment of the fire water system to fill the condensate storage tanks.

Auxiliary Feedwater Flow

An analysis performed to determine the auxiliary feedwater flow rate for current plant conditions required to remove decay heat and cooldown following a station blackout event showed that a flow rate of 140 gpm to each steam generator from the turbine-driven auxiliary feedwater pump was sufficient.

2.3.5.2.2 Acceptance Criteria

The Station Blackout Rule, 10CFR50.63, identifies the factors that must be considered in specifying the station blackout duration. It requires that, for the station blackout duration, the plant be capable of maintaining core cooling and appropriate containment integrity. It also addresses requirements for station batteries, and requirements for alternate AC power sources.

2.3.5.2.3 Description of Analyses and Evaluations

The following issues are evaluated for impact of the EPU: condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment ventilation, reactor coolant inventory, plant procedures and training, and AFW System flow requirements for SBO.

Condensate Inventory for Decay Heat Removal

The total condensate inventory required for 4 hours of decay heat removal and reactor coolant system cooldown at the EPU core power level of 1775 MWt has been determined to be 54,300 gallons. The minimum usable volume of water in the condensate storage tanks for EPU conditions has increased from 22,500 gallons to 24,350 gallons (LR section 2.5.4.5, Auxiliary Feedwater Systems). Since the minimum usable volume in the condensate storage tanks remains less than total condensate inventory required, a backup source of condensate is required. Analysis shows that the fire water system will be able to supply the CSTs at a flow rate which will satisfy the condensate storage volume requirement for a 4 hour SBO coping period, given an initial volume in the CSTs of 24,350 gallons at EPU conditions.

Class 1E Battery Capacity

Evaluation of plant fluid systems affected by operation for EPU conditions shows that there are no new station blackout loads that require 125VDC control or motive power, and that there is no need to modify existing SBO loads that require 125VDC control or motive power. Accordingly, the station vital batteries are not affected by the EPU and continue to have sufficient capacity to meet station blackout loads for the 4 hour coping duration at EPU conditions.

Compressed Air

The backup nitrogen supply for the main steam atmospheric relief valves is sized to provide 8 hours of atmospheric relief valve operation in the event of loss of instrument air and is therefore adequate for atmospheric relief valve operation for the 4 hour SBO coping duration at EPU conditions.

The alternative methods for controlling auxiliary feedwater flow, in the event TDAFW pump flow control valves cannot be operated remotely due to loss of compressed air, are not affected by the EPU.

Effects of Loss of Ventilation

As discussed above, dominant areas of concern regarding loss of ventilation are the turbine-driven auxiliary feedwater pump area and the atmospheric relief valve area. The analyses which determined the ambient temperature rise during a 4 hour station blackout in these areas for current plant conditions established heat generation rates in these areas based on a main steam temperature of 550°F. The zero load main steam temperature at EPU conditions is 547°F. Therefore, since the main steam temperature used in the analyses for current plant conditions envelopes the main steam temperature at EPU conditions, the EPU does not affect the results of the current plant analyses for maximum temperatures in the turbine-driven auxiliary feedwater pump area and the atmospheric relief valve area.

The EPU does not affect the maximum temperatures determined in the analyses under current plant conditions for the Control Room, Battery Rooms, and Relay Room.

Based on the above evaluations, the operator actions specified in the "Loss of All AC Power" procedure for opening of the doors identified in LR section 2.3.5.2.1 above are not affected by the EPU.

Containment Isolation

The EPU does not add or remove any containment isolation valves. The ability to close required valves and the required valve position indication capability is not related to power level or other EPU-related changes. Accordingly, the evaluation of this issue for current plant conditions remains applicable for EPU conditions.

Reactor Coolant Inventory

The limits on reactor coolant system leakage are not changing for the EPU. The identified reactor coolant pump seal leakage rate of 25 gpm / pump is not affected by the EPU. Therefore, the reactor core will remain covered for the 4 hour duration of an SBO at EPU conditions.

Plant Procedures and Training

The station blackout event emergency operating procedure, "Loss of All AC Power," will be reviewed / revised as required to ensure that the required volume of makeup water from the fire water system is added to the CST within the 4 hour coping duration under EPU conditions. The procedure addressing alternate water supply to the auxiliary feedwater pumps is not affected by the EPU.

The "Loss of All AC Power" procedure specifies that the operator manually trip the reactor if not tripped automatically and also to stop both reactor coolant pumps. For the scenario where non-vital power is available, stopping the reactor coolant pumps ensures that the reactor coolant pumps will not continue to run without seal cooling. The EPU does not affect the requirement to perform these actions.

Auxiliary Feedwater Flow

As addressed in section 2.3.5.2.1 above, an AFW flow rate of 140 gpm per steam generator, or a total of 280 gpm, is sufficient for decay heat removal and cooldown for an SBO event at current plant conditions. Analysis shows that 280 gpm remains sufficient for decay heat removal and cooldown at EPU conditions.

License Renewal Evaluation

The impact of the uprate on individual systems, structures, and components credited for coping with a station blackout event is addressed in the respective system evaluation sections of the License Report.

2.3.5.3 Conclusion

The effects of the proposed EPU on the plant's ability to cope with and recover from a station blackout event for the period of time established in the plant's licensing basis have been reviewed. It is concluded that the effects of the proposed EPU on station blackout have been adequately evaluated and it has been demonstrated that the plant will continue to meet the requirements of 10CFR50.63 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to station blackout.

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

2.4.1.1 Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The Ginna Nuclear Power Plant, LLC (Ginna) staff conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The Ginna staff's review was also conducted to ensure that failures of the systems do not affect safety functions.

The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10CFR50.55a (a)(1), 10CFR50.55a(h), and:

- GDC-1, insofar as it requires that structures, systems, and components (SCCs) important-to-safety are designed, fabricated, erected, and tested to quality standards commensurate with their importance to functions to be performed.

- GDC-4, insofar as it requires that SSCs be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-13, insofar as it requires that instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and for accident conditions as appropriate to ensure safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

- GDC-19, insofar as it requires that a control room is provided from which actions can be taken to operate the nuclear unit safely under normal conditions, and maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs).

- GDC-20, insofar as it requires protection systems be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control

systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of important-to-safety systems and components.

- GDC-21 insofar as it requires protection systems be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

- GDC-22 insofar as it requires protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis.

- GDC-23 insofar as it requires protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- GDC-24, insofar as it requires that the protection system is separated from the control systems to the extent that a system satisfying all reliability, redundancy, and independence requirements of the protection systems is left intact in the event of a failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems. Interconnection of the protection and control systems will be limited so as to ensure that safety is not significantly impaired.

Specific review criteria are contained in SRP sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Ginna Current Licensing Basis

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), section 3.1, the general design criteria used during the licensing of the Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR, sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the

SEP review into the Current Licensing Basis. Specifically, as discussed in section 7.1.2 of the Ginna UFSAR, "Identification of Safety Criteria," the adequacy of Ginna Station instrumentation and control systems' design was reviewed in 1972 on the bases of the General Design Criteria contained in Appendix A to 10CFR50, and the criteria included in IEEE 279-1971, both of which were promulgated after the licensing of the Ginna Station.

Compliance of the design with 1972 General Design Criteria of Appendix A to 10CFR50 is discussed in section 3.1.2 of the Ginna UFSAR. Evaluation of the design with respect to guidance provided in Safety and Regulatory Guides effective in 1972 is discussed in section 1.8 of the UFSAR. The General Design Criteria discussed in section 3.1.2 as they apply to the Reactor Protection, Safety Features Actuation, and NSSS control systems include the following:

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1, General Design Criteria 1 – Quality Standards and Records. GDC-1 requires that safety-related SCCs are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

  All systems and components of the facility were classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity were designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity were designated Class II. Those items not related to reactor operation or safety were designated Class III. Note that RG&E no longer uses this classification scheme. The classification of structures and equipment is discussed in Ginna UFSAR section 3.2.

  Safety-related SSCs are essential to the protection of the health and safety of the public. Consequently, they were designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. Discussions of applicable codes and standards, quality assurance programs, test provisions, etc., that were used are given in the section describing each system.

  A complete set of as-built facility plant and system diagrams are maintained throughout the life of the plant. Records of modifications to the general arrangement and structural plans are also maintained throughout the life of the plant.

- GDC-2 is described in Ginna UFSAR section 3.1.2.1.2, General Design Criteria 2 – Design "Bases for Protection against Natural Phenomena." GDC-2 requires safety-related SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

  All systems and components designated Seismic Category I are designed so that there is no loss of function in the event of the safe shutdown earthquake. Measures

were also taken in the plant design to protect against high winds, sudden barometric pressure changes, seiches, and other natural phenomena.

On May 22, 1992, Generic Letter (GL) 87-02, Supplement 1, transmitted Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure, Revision 2, dated February 14, 1992 (GIP-2). Supplemental Safety Evaluation Report No. 2 approved the methodology in the Generic Implementation Procedure for use in verification of equipment seismic adequacy including equipment involved in future modifications and replacement equipment. In letters dated November 30, 1992, and June 8, 1993, the NRC accepted RG&E's response to Generic Letter 87-02, Supplement 1.

- GDC-4 is described in the Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 – "Environmental and Missile Design Bases." As described in this UFSAR section, Ginna Station received post-construction review as part of the SEP. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant.

  Environmental Design Of Mechanical And Electrical Equipment (UFSAR section 3.11)

  Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (UFSAR section 3.6)

  - Pipe Breaks Inside Containment (SEP Topic III-5.A)
  - Pipe Breaks Outside Containment (SEP Topic III-5.B)

- GDC-13 is described in Ginna UFSAR section 3.1.2.2.4, General Design Criteria 13 – "Instrumentation and Control." GDC-13 requires that instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

  Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain containment pressure, neutron flux, primary coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

  The fission process is monitored and controlled for all conditions from the source range through the power range. The neutron monitoring system detects core conditions that could potentially threaten the overall integrity of the fuel barrier due to excess power generation and provides a corresponding signal to the Reactor Trip System (RTS). In addition to the ex-core neutron monitoring system, movable in-core instrumentation provides the capability of mapping the core.

The non-nuclear regulating, process, and containment instrumentation measures temperatures, pressure, flow, and levels in the reactor coolant system, steam systems, containment and other auxiliary systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the plant are indicated, recorded, and controlled from the control room. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

- GDC-19 is described in Ginna UFSAR section 3.1.2.2.10, General Design Criteria 19 – "Control Room." GDC-19 requires that a control room is provided from which actions can be taken to operate the nuclear unit safely under normal conditions, and maintain it in a safe condition under accident conditions, including LOCAs.

  The station is equipped with a control room which contains controls and instrumentation as necessary for operation of the reactor and turbine generator under normal and accident conditions. The control room is capable of continuous occupancy by the operating personnel under all operating and accident conditions, within specified dose limits.

  Although the likelihood of conditions which could render the main control room inaccessible even for a short time is extremely small, provisions have been made so that plant operators can shut down and maintain the plant in a safe condition by means of controls located outside the control room. During such a period of control room inaccessibility, the reactor will be tripped and the plant maintained in a safe shutdown condition.

- GDC-20 is described in Ginna UFSAR section 3.1.2.3.1, General Design Criteria 20 – "Protection Systems Functions." GDC-20 requires protection systems be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety

  A plant protection system, as described in UFSAR section 7.2, is provided to automatically initiate appropriate action whenever specific plant conditions reach pre-established limits. These limits ensure that specified fuel design limits are not exceeded when anticipated operational occurrences happen. In addition, other protective instrumentation is provided to initiate actions which mitigate the consequences of an accident.

- GDC-21 is described in UFSAR section 3.1.2.3.2, General Design Criteria 21 – "Protection System Reliability and Testability." GDC-21 requires protection systems be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

  Sufficient redundancy and independence are designed into the RTS to ensure that no single failure results in loss of protection function. The system is designed such that it will accommodate any single component failure and still perform its protective function.

  Reliability and independence is obtained by redundancy within each tripping function. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Each channel is continuously fed from its own independent electrical sources. Failure to deenergize a channel when required would be a mode of malfunction that would affect only that channel. The trip signal furnished by the two remaining channels would be unimpaired in this event.

  All reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Bypass removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; (i.e., a two-out-of-three circuit becomes a one-out-of-two circuit). Testing does not trip the system unless a trip condition exists in a concurrent channel.

- GDC-22 is described in Ginna UFSAR section 3.1.2.3.3, General Design Criteria 22 – "Protection System Independence." GDC-22 requires protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

  The Ginna Station protection system was designed so that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions do not result in the loss of the protective function. The design includes the techniques of functional diversity or diversity in components design and principles of operation to the extent practical in preventing the loss of the protection functions (e.g., use of turbine-driven and motor-driven auxiliary feedwater pumps).

- GDC-23 is described in Ginna UFSAR section 3.1.2.3.4, General Design Criteria 23 – "Protection System Failure Modes." GDC-23 requires protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

  The RTS is designed to fail-safe upon loss of power. Each reactor trip circuit is designed so that trip occurs when the circuit is deenergized; an open circuit or loss of channel power, therefore, causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses. Failure to deenergize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

  Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on each drive, allowing the rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power. Automatic starting of either emergency diesel generator is initiated by redundant undervoltage relays on the 480-V safeguards bus with which the diesel generator is associated, or by the safety injection signal. Engine cranking is accomplished by a stored energy system supplied solely for the associated diesel generator. The undervoltage relay scheme is designed so that loss of 480-V power does not prevent the relay scheme from functioning properly.

- GDC-24 is described in Ginna UFSAR section 3.1.2.3.5, General Design Criteria 24 – "Separation of Protection and Control Systems." GDC-24 requires protection systems be separated from the control systems to the extent that a system satisfying all reliability, redundancy, and independence requirements of the protection systems is left intact in the event of a failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems. Interconnection of the protection and control systems will be limited so as to ensure that safety is not significantly impaired.

  Evaluation of the Ginna Station RTS isolation was performed as part of the SEP, Topic VII-1.A. The safety evaluation concluded that the RTS is adequately isolated from non safety systems and satisfies the criteria set forth in 10CFR50, Appendix A (GDC 24), and IEEE-279 (1971), section 4.7.2.

- GDC-25 is described in the Ginna UFSAR section 3.1.2.3.6, General Design Criterion 25 – "Protection System Requirements for Reactivity Control Malfunctions." GDC-25 requires protection systems be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

The RTS is designed to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems. Reactor shutdown with rods is completely independent of the normal control functions. The trip breakers interrupt the power to the rod mechanisms to trip the reactor regardless of existing control signals.

- GDC-29 is described in the Ginna UFSAR section 3.1.2.3.10, General Design Criterion 29 – "Protection Against Anticipated Operational Occurrences." GDC-29 requires protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

  The Ginna protection and reactivity control systems are designed to ensure extremely high reliability in regard to their required safety functions in any anticipated operational occurrences. Anticipated failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. Loss of power to the protection system will result in a reactor trip.

Other Ginna UFSAR sections that address the design features and functions of the reactor protection and reactor control systems and instrumentation include:

- Ginna UFSAR section 7.1.2, "Identification of Safety Criteria," which describes the reactor protection and reactor control Instrumentation design basis and the requirements for operability and testability.

- Ginna UFSAR section 7.2, "Reactor Trip System (RTS)," describes the design criteria for the reactor protection system and provides a description of reactor protection system operation, reactor trips, permissives, and the interaction of the control and protection systems.

- Ginna UFSAR section 7.3, "Engineered Safety Features Systems (ESFAS)," which describes the design criteria for the ESFAS system and provides a description of the operation, actuation signals, testability, redundancy and independence, and key instrumentation.

- Ginna UFSAR section 7.4, "Systems Required For Safe Shutdown," which identifies the minimum systems required to take the plant from operating conditions to MODE 5.

- Ginna UFSAR section 7.5, "Safety Related Display Instrumentation," identifies the Ginna NSSS and BOP instrumentation subject to the requirements of Regulatory Guide 1.97, Post Accident Monitoring Instrumentation and documents the NRC evaluation and approval of the Rochester Gas and Electric's position relative to the guidance provided in Regulatory Guide 1.97, Revision 3 (reference 1).

- Ginna UFSAR section 7.6, "Other Instrumentation Systems Required For Safety," which describes the instrumentation required for overpressure protection during low power operation, auxiliary feedwater system automatic initiation and flow indication, subcooling meter, DC power system voltage indication and annunciation, and reactor vessel level indication system.

- Ginna UFSAR section 7.7, "Control Systems Not Required For Safety," which provides a description of the reactor control system (rod control, steam dump, pressurizer pressure and level, Steam Generator level control and overfill protection, and turbine bypass), plant response to design loading and unloading, and incore instrumentation. Also included in this section is a description of the nuclear instrumentation system from source range to 120% power, reactor coolant temperature indication, the process computer, and the safety parameter display assessment system (SPDS).

In addition to the evaluations described in the UFSAR, the Ginna Station's electrical and instrumentation and control (I&C) systems were evaluated for plant license renewal. The evaluation of the electrical and I&C components, and the subsequent review and conclusions are discussed in section 2.5 of NUREG-1786, License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant dated May 2004. BOP system instrument and control systems are not specifically addressed in the SER however some BOP instrumentation, specifically turbine first stage pressure, is described in section 2.3.4.4, "Turbine Generator and Supporting Systems." The programs used to manage the aging effects associated with instrumentation is addressed in the SER, however, transmitters are identified as active and excluded from Aging Management Review.

2.4.1.2 Technical Evaluation

2.4.1.2.1 Introduction

With respect to EPU, the reactor protection system, engineered safety features actuation system (ESFAS), and the reactor control systems, are impacted by the increase in reactor thermal power from 1520 MWt to 1775 MWt and the transition from Westinghouse 14x14 OFA fuel to Westinghouse 14x14 422V+ fuel.

2.4.1.2.2 Input Parameters and Assumptions

The design parameters associated with the uprate and fuel transition are identified in LR section 1.1, "Nuclear Steam Supply System Parameters," Table 1-1. The initial best estimate nominal 1775 MWt full power operating parameters are identified in Table 2.4.1-1 below. The values listed in Table 2.4.1-1 are current best estimates and some values may change as turbine and core design are refined.

Table 2.4.1-1

Parameter	Value
Rated Reactor Core Power MWt	1775
NSSS Power (core Power + RCP Heat) MWt	1781
Main Steam Flow (total flow) lbm/hr	7.7×10^6
Main Steam Flow (per SG) lbm/hr	3.85×10^6
Main Feedwater Flow (plus blowdown total) lbm/hr	7.78×10^6
Main Feedwater Flow (per SG plus blowdown) lbm/hr	3.89×10^6
Main Steam Pressure psig	785
Rated Full Power ΔT °F	67°F
Rated Full Power Average T_{avg} °F	572°F - 574°F
No Load Average T_{avg} °F	547
Pressurizer Level program 0% - 100%	20% - 57%
Full Load Turbine First Stage Pressure psig (subject to final HP turbine design)	645
Feedwater Temperature °F	432

The impact of the physical differences between the Westinghouse 14x14 OFA fuel and the Westinghouse 422V+ fuel has been evaluated in LR section 2.8.1, "Fuel System Design," LR section 2.8.2, "Nuclear Design," and LR section 2.8.3, "Thermal and Hydraulic Design," and LR section 2.8.4.1, "Functional Design of the Control Rod System." As described in LR section 2.8.4.1, the difference in the 422V+ fuel top nozzle design will have an impact on the microprocessor rod position indication (MRPI) system. This change is also described below in LR section 2.4.1.2.3.3, "Control Systems."

2.4.1.2.3 Description of Analyses and Evaluations

The combined effects of the fuel transition and the increase in reactor thermal power have been evaluated for normal operation, operational transients, and accident conditions described in UFSAR sections 6.0, "Engineered Safety Features," 7.7.1.1.4," Reactor Control System Operation," and section 15, "Accidents." These analyses used the most conservative combination of Nuclear Steam Supply System (NSSS) design values from LR section 1.1, "Nuclear Steam Supply System Parameters," Table 1-1. In addition, these analyses included changes to specific emergency safety features actuation system (ESFAS) analytical limits described in LR section 2.4.1.2.3.2 below to provide additional instrumentation calibration margin. The results of the transient and accident analyses are described in the following LR section:

- LR section 2.4.2, "Plant Operability."
- LR section 2.6, "Containment Review Considerations"
- LR section 2.8.5, "Accident and Transient Analyses"

In addition to the ESFAS analytical limit changes requested by Ginna, the analyses identified additional instrumentation and trip setpoint changes that are required to ensure DNB, RCS pressure, and secondary system pressure remain within the allowable design margins and the response to the design basis operational transients remain acceptable. These changes are described in LR section 2.4.1.2.3.1, "Reactor Protection Systems," LR section 2.4.1.2.3.2, "Safety Features Actuation," and LR section 2.4.1.2.3.3, "Control Systems" below.

The above analyses determined that with the exception of the following instruments, the NSSS instrumentation ranges, scalings, and setpoints used in the reactor protection, engineered safety features actuation system (ESFAS), and reactor control instrumentation remained adequate for EPU. The specific changes to these instruments are described in LR section 2.4.1.2.3.1, "Reactor Protection Systems," LR section 2.4.1.2.3.2, "Safety Features Actuation," and LR section 2.4.1.2.3.3, "Control Systems below:

- Power Range and Intermediate Range nuclear instruments
- RCS Temperature instrumentation
- Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)
- Main Steam Flow instrumentation

Using best estimate data obtained from EPU heat balances (see Table 2.4.1-1, above), balance of plant (BOP) instrumentation was evaluated to determine required changes using the following methodology:

- System analysis were performed to determine how the EPU process conditions changed compared to the current system operating conditions for the BOP systems.
- For those systems (sub-systems) process conditions changed for EPU, the system instrumentation was evaluated to determine if the instrumentation ranges, scalings, and setpoints remained adequate for EPU conditions.
- For those instruments where the current instrument ranges, scalings, or setpoints are not adequate to support EPU conditions, recommend new ranges, scalings, setpoints, or instrument replacement as required.

Systems covered by this evaluation include the following fluid systems:

- Main Steam
- Extraction Steam
- Condensate and Feedwater
- Station Service Cooling Water
- Component Cooling Water
- Auxiliary Feedwater
- Steam Generator Blowdown
- Feedwater Heater and Moisture Separator Reheater Drains
- Spent Fuel Pool Cooling

- Circulating Water
- Main Turbine Control

With the exception of the following instrumentation, the BOP instrumentation ranges and setpoints were determined to be adequate for EPU. Changes to the following instrumentation is described in LR section 2.4.1.2.3.2, "Safety Features Actuation," and LR section 2.4.1.2.3.3, "Control Systems," below:

- Turbine First Stage Pressure instrumentation
- Main Steam Flow instrumentation
- Main feedwater flow instrumentation
- Main feedwater pump low suction pressure instrumentation
- Setpoint to LP feedwater heater bypass valve
- Heater drain pump flow instrumentation
- Heater drain tank inlet drain temperature instrumentation
- Standby Auxiliary Feedwater flow instrumentation
- Condensate storage tank level instrumentation
- Condensate booster pump discharge pressure instrumentation

UFSAR Table 7.5-1, "Comparison of Ginna Station Post Accident Instrumentation To Regulatory Guide 1.97, Revision 3, Criteria," identifies the Ginna NSSS and BOP instrumentation subject to the requirements of Regulatory Guide 1.97, "Post Accident Monitoring Instrumentation." Table 7.5-1 was reviewed for the impact of the identified changes to the NSSS and BOP instrumentation resulting from EPU. Although the setpoints of some of the instruments will be changing, the current calibration range of the instruments remain adequate for EPU. The evaluation determined that the only instruments listed in Table 7.5-1 which require changes resulting from EPU are:

- Main Feedwater flow instrumentation
- Main Steam flow instruments
- Standby Auxiliary Feedwater flow instrumentation

Following the implementation of the changes to these instruments described in LR section 2.4.1.2.3.2, "Safety Features Actuation," and LR section 2.4.1.2.3.3, "Control Systems," these instrumentation will continue to satisfy their Regulatory Guide 1.97 requirements.

Technical Specification Limiting Safety System Setting (LSSS) values and trip setpoint values are derived from analytical values used in the above described analyses corrected to account for the specific instrument or control system uncertainty. Ginna calculates instrument uncertainty and setpoints using the methodology in ISA-67-04 as described in Technical Specification Amendment 85 of Improved Technical Specifications and approved by the NRC in the SER dated September 22, 2004 (reference 2).

2.4.1.2.3.1 Reactor Protection

The design bases and description of the Ginna RPS is described in UFSAR section 7.2.1, "Reactor Trip System (RTS)," and includes a listing of the reactor trips, purpose of each trip, and any associated protection and control permissives. The RPS automatically trips the reactor to protect against reactor coolant system damage caused by high system pressure and to protect the reactor core against fuel rod cladding damage caused by a departure from nucleate boiling. The basic reactor tripping philosophy is to define a region of power and coolant temperature and pressure conditions allowed by the primary trip functions (overpower ΔT trip, overtemperature ΔT trip, and nuclear overpower trip). The allowable operating region within these trip settings is provided to prevent any combination of power, temperature, and pressure that would result in a departure from nucleate boiling with all reactor coolant pumps in operation.

Additional trip functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip, loss-of-flow trip, steam-generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary trip functions for specific accident conditions and mechanical failures.

The following is a list of the RPS instrumentation and setpoint changes necessary to ensure the RPS will continue to satisfy it's design functions at EPU conditions.

Nuclear Instrumentation

EPU redefines the 100% power neutron flux levels and will impact the flux level to percent power relationship for the Intermediate Range and Power Range nuclear instruments. Since the source range nuclear instrumentation is deenergized well below the power range, during reactor startup, there are no changes required to the Source Range instrumentation settings. The EPU accident and transient analyses determined that for some accidents the analytical limit for the Power Range high power trip would need to be reduced from the current 118% to 115% which will reduce the Technical Specification LSSS accordingly (112.27% to 109.27%). Although the Power Range high power trip LSSS is decreasing to 109.27%, the current field trip setpoint of 108% has adequate margin to accommodate the new LSSS limit and will not change. The change in the Power Range high power trip LSSS must be approved as part of the Technical Specification change being submitted in the EPU license amendment request.

The accident and transient analyses also determined the analytical limit for the Power Range low power reactor trip at ≤35% of rated thermal power remained adequate for EPU, therefore, the current Power Range low power reactor trip setpoint (24%) remains adequate for EPU. In addition, the Intermediate Range rod stop and reactor trip setpoints (20% and 25% respectively) will remain adequate for EPU.

The Power Range and Intermediate Range instruments are typically recalibrated as a part of the normal core reload process to account for the changes in core design. For EPU, this calibration must also account for the change in percent power level and the 100% power flux level. Once this initial calibration is complete, the Intermediate Range rod stop and trip as well as the Power Range low power reactor trip will function as required. Frequent secondary calorimetrics are used to calibrate the Power Range instruments to calorimetric power during power ascension which maintains the appropriate Power Range flux to percent power relationship. Once calibrated as described above, the power range reactor trips, rod stops and inputs to permissives P-1, P-7, P-8, P-9, and P-10 will function at the appropriate relative power setpoint.

RCS Temperature Instrumentation

LR section 2.8.5 made recommendations for the T_h, T_c, T_{avg} and ΔT instrument ranges and setpoints to ensure the instrumentation would provide the required indication, core DNB protection, and plant response during accidents and transients over the entire range of operation at EPU conditions. The current range of the T_c instruments (510°F– 620°F) satisfies the recommended range of 510°F – 590°F. The T_h, T_{avg}, and ΔT instruments including indications will be recalibrated for a range as follows:

- T_h – 540°F - 650°F
- T_{avg} - 540°F – 620°F
- ΔT – 0°F - ≥80°F (Ginna plans to initially scale the instruments 0°F - 85°F)

In addition, the transient analyses recommended a 3.5 second filter be added in the T_h input to the T_{avg} and ΔT protection channels upstream of the modules which calculate T_{avg} and ΔT. The filters are required to improve the margin to trip for the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) trips and also add stability to the rod control system. T_{avg} and ΔT associated alarm setpoints will be recalibrated as necessary to essentially maintain the same margin to alarm at the EPU conditions as existed prior to EPU.

Overtemperature ΔT (OTΔT) Trip

Typically the values for the OTΔT trip setpoints constants are listed in the cycle specific Core Operating Limits Report (COLR) for each fuel cycle. For the initial EPU startup. the OTΔT trip setpoint will be recalibrated with OTΔT constants changed as follows:

Parameter	Current	EPU
Analytical Limit	1.32073	1.30
Constant K1	1.20	1.19
Constant K2	0.0009/psi	.00093/psi
Constant K3	0.0209/°F	0.0185/°F

- Outside EPU Ginna has submitted a request to change from Constant Axial Offset Control (CAOC) to implement Relaxed Axial Offset Control (RAOC) to be implemented during EPU startup. This change was requested by Ginna in reference 3. The current f(ΔI) control function of the OTΔT trip setpoint only responds to a

positive axial offset, therefore, an additional module will be added to the system to account for a negative axial offset. The new module will be similar in design to modules originally provided with these circuits. The f(ΔI) function will be calibrated for EPU in accordance with the values listed in the cycle specific COLR.

Overpower ΔT (OPΔT) Trip

As with OTΔT trip setpoint, the values for the OPΔT trip setpoints constants are typically listed in the cycle specific COLR for each fuel cycle. The accident and transient analyses determined the rate sensitive temperature portion of the setpoint and the f(ΔI) function are not necessary for the OPΔT trip circuit to provide the required protection for maintaining the fuel design limits. For the initial EPU startup, the f(ΔI) function is being disabled and the OPΔT trip setpoint constants changed as follows:

Parameter	Current	EPU
Analytical Limit	1.14877	1.15
Constant K4	1.077	1.077
Constant K5	0.0011/°F	0.0014/°F
Constant K6	0.0262	0
T_3 time constant	10 seconds	0 seconds

Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Rod Stops

The setpoint for the P-1 Permissive from two-out-of-four high overtemperature ΔT or overpower ΔT at 1.71°F below trip setpoints is being redefined from a specific temperature value to a value 3% below the full power ΔT. Although stated as an absolute value, the current 1.71°F corresponds to a value 3% below the pre uprate full power ΔT, therefore there is no actual technical change but clarifies the basis for establishing the actual runback setpoint value. At EPU, the 3% below full power ΔT setpoint will correspond to a rod stop and turbine runback occurring at 64.9°F (2.01°F below the trip setpoint) assuming a nominal full power ΔT of 67°F with RCS temperature and pressure at nominal values.

Anticipated-Transient-Without-Scram Mitigation System Actuation Circuitry (AMSAC)

The Ginna Anticipated-Transient-Without-Scram Mitigation System Actuation Circuitry (AMSAC) as required by 10CFR50.62 is described in UFSAR section 7.2.6, "Anticipated-Transient-Without-Scram Mitigation System Actuation Circuitry." The changes to this circuitry are associated with the arming permissive C-20 which arms and disarms the circuit at a turbine first stage pressure equivalent to 40% nuclear power, and recalibrating the turbine first stage pressure, steam flow, and feedwater flow inputs for the EPU full load values. The C-20 permissive will be recalibrated to arm/disarm at the appropriate turbine first stage pressure consistent with the new 0% - 100% power nominal turbine first stage pressure range of 0 – 645 psig.

P-7 Permissive Changes

The P-7 permissive is used to bypass the low pressurizer pressure reactor trips during low power or startup operation. It is also used to bypass reactor coolant low flow, undervoltage, and under frequency trips. It is derived from a bistable circuit indicating less than 8.5%power as measured by both first stage turbine pressure (two-out-of-two) and power range (two-out-of-four) less than approximately 8.0%. The power range input is supplied by the P-10 permissive. Calibration of the Power Range input is discussed above in Nuclear Instrumentation. The input from turbine first stage pressure input will be recalibrated to actuate at the value consistent with the new 0% - 100% power nominal turbine first stage pressure range of 0 – 645 psig.

P-8 Permissive Change

The P-8 permissive is used to block a single loop loss of coolant flow reactor trip when 3/4 power range nuclear instruments are less than the permissive setpoint, currently 49% power. The single loop loss of coolant flow trip is unblocked when 2/4 power range nuclear instruments indicate greater than the P-8 setpoint. The analyses performed for EPU determined that an analytical limit of ≤35% power is required to ensure all accidents and transients impacted by RCS flow maintain DNB within acceptable limits. Therefore, the P-8 Technical Specification setpoint limit will be reduced from the current ≤49% power to <29% (analytical limit – instrument uncertainty). This change must be approved as part of the Technical Specification change being submitted in the EPU license amendment request. The field setpoint for P-8 will be changed from the current 49% nuclear power to 25% nuclear power.

2.4.1.2.3.2 Safety Feature Actuation System

The engineered safety features actuation systems (ESFAS) are used to provide protection against the release of radioactive materials in the event of a loss-of-coolant accident or a secondary line break accident. The engineered safety features systems function to maintain the reactor in a shutdown condition. They also provide sufficient core cooling to limit the extent of fuel and fuel cladding damage and to ensure the integrity of the containment structure. These functions rely on the ESFAS and associated instrumentation and controls. The following identifies the changes to the ESFAS instrumentation, analytical limits, and settings being implemented as part of EPU.

Main Steam Flow Instrumentation

As identified in LR section 2.4.1.2.3 above, the current main steam line flow transmitters require changes to support EPU. The transmitters are currently calibrated with a range of 0 – 3.8×10^6 which is less than the predicted EPU nominal steam flow of 3.85×10^6 lbm/hr. The main steam flow transmitters will be recalibrated for a range of 0 - 4.6×10^6 lbm/hr. This range ensures that the steam flow indication will continue to meet the required Regulatory Guide 1.97 range of 110% of design flow stated in UFSAR Table 7.5-1 plus provide additional scaling to ensure the high-high steam flow signal will occur within indicator range.

Changes to ESFAS Analytical Limits

As indicated previously, in order to increase the calibration margin on ESFAS parameter related setpoints, Ginna requested specific changes to the ESFAS analytical values used in the accident and transient analyses. Since acceptable results were achieved using these values, these values will become the basis for establishing the Technical Specification LSSS values (analytical limit – instrument uncertainty) and field setpoints. In addition, the accident analyses determined that the analytical limit for the high high steam line flow input to the steam line isolation be ≤155% of the nominal EPU full power steam flow (≤5.96E6 lbm/hr). The changes to the ESFAS analytical limits and the effect on the LSSS and field setpoints are shown in the following table. These changes must be approved as part of the Technical Specification change being submitted in the EPU license amendment request.

Parameter	Analytical Limit		Technical Specification LSSS		Field Setpoint	
	Current	EPU	Current	EPU	Current	EPU
Steam Line Isolation High High Steam Flow lbm/hr	≤3.7E6 @755 psig	≤5.96E6 @755 psig	≤3.6E6 @755 psig	≤4.53E6 @785 psig	3.6E6 @755 psig	4.44E6 @785 psig
Steam Line Isolation High Steam Flow lbm/hr @1005 psig	≤0.66E6	≤1.5E6	≤0.42E6	≤1.3E6	0.4E6	0.48E6
Steam Line Isolation Low Tavg °F	≥543	≥540	≥544.98	≥544	545	545
Containment Spray Containment Pressure High High Narrow Range - psig	≤32.5	≤33.5	≤31.11	≤32.11	28	28
Containment Spray Containment Pressure High High Wide Range - psig	≤32.5	≤33.5	≤28.6	≤29.6	28	28
Safety Injection Pressurizer Pressure Low - psig	≥1715	≥1700	≥1744.8	≥1729.8	≥1750	≥1750

Feedwater Line Isolation

A new feedwater line isolation valve is being installed in each main feedwater line to minimize the impact to containment integrity during a steamline break inside containment. These new valves will reduce the volume of water potentially available to reach the faulted steam generator for a steamline break in containment. The new valves will replace crediting the closure the main feedwater pump discharge valve in the accident analyses. Approval of this change is independent of the license amendment required for EPU. License approval for crediting the new isolation valves was requested on April 29, 2005 (reference 4).

Reactor Vessel Level Indication

The reactor vessel level indication is described in UFSAR section 7.3.2.3.1, Reactor Vessel Level Indication System. As described in LR section 2.8.1, "Fuel System Design," the differences in core differential pressure during the transition to the 422V+ fuel is expected to be very small with the RCPs running and should fall within the uncertainty of the reactor vessel

level instrumentation and therefore, there is no impact expected to the reactor vessel level indication.

2.4.1.2.3.3 Control Systems

The various reactor control systems are described in UFSAR section 7.7.1, "Control Systems Not Required For Safety." The reactor control systems are designed to limit nuclear plant transients for prescribed design load perturbations, under automatic control, within prescribed limits to preclude the possibility of a reactor trip in the course of these transients. During steady-state operation, the primary function of the reactor control is to maintain a programmed average reactor coolant temperature that rises in proportion to load. The control systems also limit nuclear plant system transients to prescribed limits about this programmed temperature for specified load perturbations. Complete supervision of both the nuclear and turbine generator plants is accomplished from the central control room. This supervision includes the capability to test periodically the operability of the RPS.

The current design basis operational transients described in UFSAR, section 7.7.1 are:

- Step-load change of ±10% or ramp load change of 5% per minute within the load range of 12.8% to 100% of rated power
- Step load decrease of 245 MWe with steam dump
- Turbine trip below 245 MWe with steam dump

Since 245 MWe will no longer represent 50% load at uprate condition, as part of EPU analyses, the reference to a specific MWe is being omitted from the definition of the design basis step-load decrease and the definition revised as a rapid ramp load decrease equivalent to 50% of the EPU rated thermal power (RTP) at a maximum turbine unloading rate of 200% per minute. For the turbine trip load reject, the reactor is assumed to be below the P-9 permissive which defeats the reactor trip due to turbine trip when indicated nuclear power is less than 50%. This change in the design basis load rejection from a step change to a rapid ramp load change at a maximum rate of 200% per minute redefines the load rejection in a more realistic manner and is consistent with uprating projects previously performed on other Westinghouse plants. Following implementation of EPU, the design basis operational transients will be defined as:

- Step-load change of ±10% or ramp load change of 5% per minute within the load range of 12.8% to 100%
- A rapid ramp load decrease equivalent to 50% rated thermal power at a maximum turbine unloading rate of 200% per minute with steam dump
- Turbine trip below 50% reactor power (P-9) with steam dump

The analyses evaluating the response to design basis operational transients at EPU conditions are described in LR section 2.4.2, "Plant Operability." The acceptable response to the design basis operation transients and accidents and transients associated control system failures are based on the changes described for the rod control system and steam dump system being implemented.

Turbine First Stage Pressure Instrumentation

When the turbine generator is on line, turbine first stage pressure increases essentially linear from 0% - 100% turbine load and provides a close correlation of secondary power to reactor power. This allows turbine first stage pressure to be used as a reliable input demand signal or permissive to the various reactor control systems between 0% and 100% reactor power. The pre-EPU 0% - 100% turbine load turbine first stage correlates to 0 – 495 psig. For EPU, a new HP turbine rotor is being installed which currently is expected to generate a 0% - 100% power nominal first stage turbine pressure of 0 – 645 psig. Actual full power turbine first stage pressure may change slightly as the HP turbine design is refined and instrument calibrations will be revised accordingly.

The existing turbine first stage pressure transmitters and associated indications will be recalibrated and scaled to a range of 0 – 700 psig. The inputs to each of the following systems will be recalibrated to respond at the appropriate value for the new 0 – 100% power nominal turbine first stage pressure of 0 - 645 psig.

- AMSAC – arm/disarm circuit permissive C-20 at first stage pressure equivalent to 40% reactor power
- P-2 Permissive – blocks Automatic Rod Withdrawal block at less than 12.8% turbine load
- P-4 Permissive - arms the steam dump system on a sudden drop in turbine load
- P-7 Permissive- in conjunction with P-10, bypasses low pressurizer pressure and low RCS flow, undervoltage, and under frequency trips
- Rod Control power mismatch and non linear gain controls
- Advanced Digital Feedwater Control System (ADFCS)
- T_{ref} input to the Reactor Coolant T_{avg} Control program
- EHC Turbine Control

Rod Control System Changes

The rod control system responds to changes in RCS temperature and secondary load as sensed by the RCS measured T_{avg} instrumentation and turbine first stage pressure instrumentation. The rod control system is designed to maintain average RCS temperature within ±1.5°F of the 0% - 100% T_{avg} program reference value (T_{ref}) derived from 0-100% power turbine first stage pressure (0 – 645 psig). In addition, the rod control system responds to deviations between the reactor power and turbine load as sensed by the mismatch between power range instruments and turbine first stage pressure instrumentation. Both the T_{avg} program and the power mismatch program controls rod speed and direction during normal and transient operation.

The EPU 0 – 100% power T_{avg} temperature program (T_{ref}) is changing from the current 547°F to 561°F to 547°F to approximately 572°F - 574°F based on a 0 – 645 psig turbine first stage pressure. Once the T_{ref} program is calibrated with the turbine first stage pressure range and temperature control band, the rods are expected to respond as designed to average T_{avg} temperature deviations from T_{ref}.

The power mismatch circuits will be calibrated with the new 0 – 100% turbine first stage pressure values which will ensure the power mismatch circuits will continue to provide maximum rod speed with a deviation between nuclear power and turbine power of 10%.

In addition the accident and transient analyses identified changes required to the non linear gain portion of the rod speed control circuits to reduce the speed of the rods to ensure the fuel design limits are not exceeded during response to a single rod drop or rod withdrawal event in addition to providing the stability during the design load change operational transients. The non linear gain inputs are being changed as follows. The range in which only the Low Gain is active is being changed from ±2% to ±1%:

Turbine Load	Low Gain	High Gain
70% -100%	from 1.5°F/% to 0.30°F/%	from 5°F/% to 1.5°F/%
20% - 70%	from 2.25°F/% to 0.45°F/%	from 7.5°F/% to 2.25°F/%
0% - 20%	from 3.0°F/% to 0.6°F/%	from 10°F/% to 3°F/%

Control Rod Position Indication

The Ginna control rod position indication systems are described in UFSAR section 7.7.1.2.6, "Rod Position Indication System." LR section 2.8.4.1, "Functional Design of the Control Rod Drive System," identified that the difference in the top nozzle length of the Westinghouse 14x14 422V+ fuel will affect the microprocessor rod position indication (MPRI) system. Operation of the MRPI system is described in the Ginna UFSAR section 7.7.1.2.6 and Technical Specification Bases 3.1. The transition point at which the MRPI system indication changes from 0 steps to 12 steps withdrawn occurs when the RCCAs in the bank have been withdrawn 6 steps. The 3 inch height increase in the rod bottom position corresponds to approximately 5 steps, resulting in the transition point occurring at approximately 1 step withdrawn. This could potentially result in the rods not providing a rod bottom indication when inserted. In addition, RCCAs will reach the fully-withdrawn position in 422V+ fuel at 225 steps instead of the current 230 steps. Also, the potential would exist to receive unnecessary rod deviation alarms.

Changes to the rod position indication systems, including possible modifications to the MRPI and/or plant process computer software, or the MRPI hardware itself are currently being assessed to ensure that correct rod position indications are available to the operator.

Pressurizer Level Program

The pressurizer level control system maintains the pressurizer level within a programmed band consistent with measured average T_{avg}. The programmed level is designed to maintain a sufficient margin above the low level alarm where the heaters turn off and Letdown isolation occurs while maintaining the level low enough that a sufficient steam volume is maintained to ensure the pressurizer does not go solid during accidents and transient conditions.

Analyses described in LR section 2.4.3, "Pressurizer Component Sizing," and LR section 2.8.5, "Accident and Transient Analyses," determined the nominal pressurizer level program for EPU must be changed from the current 35% - 50% program to a new nominal program of 20% at no load conditions to 54.5% - 57% for a full power average T_{avg} of 572°F to 574°F.

Steam Dump Control and Turbine Bypass Systems

The steam dump control and turbine bypass system is comprised of the main steam atmospheric relief valves (ARVs) and the condenser steam dumps. The ARVs can be used to remove sensible heat stored in the RCS at shutdown and cooldown when the condenser steam dumps are not available. The condenser steam dump system removes sensible heat stored in the RCS for a large rapid load decrease or a reactor trip. With condenser steam dump not available, a large rapid turbine load reduction would result in a large steam pressure increase and could potentially challenge the Main Steam Safety Valves (MSSVs). Steam is dumped in order to remove the stored heat in the primary system at a rate fast enough to prevent lifting of the MSSV for a large rapid load decrease, or a reactor trip. The evaluation of the steam bypass system is described in LR section 2.5.5.3, "Turbine Bypass", and LR section 2.4.2, "Plant Operability".

If the condenser is available, the condenser steam dumps (groups A – D) are armed based on a rapid decrease in turbine first stage pressure (equivalent to >10% load decrease) and the dump valves either modulate open or are tripped open based on the magnitude of error (ΔT) between the measured average T_{avg} and the reference temperature(T_{ref}) programmed off turbine first stage pressure.

As described in LR section 2.4.2, "Plant Operability," the current steam dump valve capacity is sufficient to accommodate a rapid load decrease equivalent to 50% reactor thermal power (RTP) at a rate of 200% per minute or a turbine trip at less than 50% reactor thermal power (RTP) at EPU conditions provided the following changes are implemented in the steam dump control system:

Parameter	Current		EPU	
Turbine Operating Dead band	5°F		2°F	
Turbine Operating Tavg Lead Time Constant	20 Seconds		16 Seconds	
Proportional Gain in Percent Valve Lift per °F	6.7%/°F		6.7%/°F (Turbine Tripped) 11.76%/°F (Turbine Operating)	
Turbine Operating - □T (°F) Required to Modulate Valves Open	Group A Group B Group C Group D	5 – 8.75 8.75 – 12.5 12.5 – 16.2 16.2 – 19.9	Group A Group B Group C Group D	2 – 4.125 4.125 – 6.25 6.25 – 8.375 8.375 – 10.5
Turbine Operating - ΔT(°F) Required to Snap Open Valves	Group A and B Group C and D	12°F 20°F	Group A and B Group C and D	6.25 10.5

Analyses to optimize these settings are still in progress and the final settings may change slightly from those presented.

In addition to the setting changes for the "turbine operating" mode described above, a modification to the steam dump circuit will be implemented to apply the "turbine trip" mode on a reactor trip instead of a turbine trip. This modification will prevent the application of the more aggressive "turbine operating" steam dump settings shown above in scenarios when the reactor trips and less aggressive steam dump is desired.

Condensate and Feedwater System Instrumentation

The changes in the condensate and feedwater system for EPU are driven by the increased flow and associated pressure drops through the system at uprate conditions. As identified above and in LR section 2.5.5.4, "Condensate and Feedwater," the following changes are necessary to condensate and feedwater system instrumentation and setpoints.

- The main feedwater flow transmitters will be replaced and the loop will be re-calibrated from the current 0 – 3.8x10^6 lbm/hr to 0 – 4.6x10^6 lbm/hr. The new instrument range will continue to satisfy the Regulatory Guide 1.97 monitored variable of 110% of design flow stated for the main feedwater flow in UFSAR Table 7.5-1.
- Heater drain pump flow measurement loop will be recalibrated and rescaled from the current 0 - 2.684x10^6 lbm/hr to 0 – 3.0x10^6 lbm/hr.
- Heater drain tank inlet drain temperature measurement loop will be recalibrated from the current 175°F - 375°F to 175°F - 400°F.
- Main feedwater pump suction flow transmitters and control room indicators will be re-calibrated and rescaled from the current 0 – 3.5x10^6 lbm/hr to 0 – 4.6x10^6 lbm/hr.
- The condensate pump discharge pressure alarm and standby pump auto start setpoint are being changed to provide sufficient operating margin.

- The condensate booster pump standby pump auto start setpoint is being increased to ensure adequate discharge pressure margin is maintained at EPU.

- The main feedwater pump suction pressure setpoint that provides the pump start permissive and auto open signal to the LP heater bypass valve is being changed to provide the required margin for feedwater pump net positive suction pressure (NPSH) at uprate feedwater flows consistent with the design of the replacement main feedwater pump impellers. In addition, a delay is being added to the LP heater bypass valve open circuit to minimize the potential for spurious actuation and resultant condensate and feedwater system instability associated with events such as a loss of a condensate pump, condensate booster pump or heater drain pump.

- The main feedwater pump NPSH calculator setpoint which provides an alarm and also opens the LP heater bypass valve on low NPSH is being reset to provide the required margin for feedwater pump NPSH at uprate feedwater flows consistent with the design of the replacement main feedwater pump impellers. As with the main feedwater pump low suction pressure signal, the signal to the LP heater bypass valve will be delayed to minimize the potential for spurious actuation and resultant condensate and feedwater system instability associated with events such as a loss of a condensate pump, condensate booster pump or heater drain pump.

Auxiliary Feedwater System Instrumentation

- The standby auxiliary feedwater pump flow transmitters will be replaced and the flow loop recalibrated for a full scale measurement range of 0 - 300 gpm to accommodate the increased flow required at EPU as described in LR section 2.5.4.5, Auxiliary Feedwater System. The new instrument range for will continue to satisfy the Regulatory Guide 1.97 monitored variable of 110% of design flow stated for the standby auxiliary feedwater flow in UFSAR Table 7.5-1.

Steam Generator Level Control

The steam generator level control system is described in UFSAR section 7.7.1.5, "Steam Generator Level Control." The steam generator water level is controlled by a digital microprocessor controlled steam generator feedwater control system termed the advanced digital feedwater control system (ADFCS). The ADFCS provides automatic control of the programmed level in the steam generators without the need for operator intervention over the range of power operation. This range of operation extends from the point at which the transition is made from feeding via the preferred auxiliary feedwater system to feeding via the main feedwater system on the main feedwater bypass valve (approximately 2-3% power) up to full power. One control system operates on both the Main Feedwater Regulating Valve (MFRV) and main feedwater bypass valves without the need for manual action to switch operating modes or switch between valves. The following is a list of the signals input to the ADFCS. With respect to EPU, of the following inputs to the system, the steam generator levels and valve positions are not impacted, however, for the remaining inputs, the ADFCS program software will need to be updated as necessary with the expected EPU full power values.

- Narrow-range steam generator water level
- Wide-range steam generator water level
- Steam flow
- Feedwater flow
- Feedwater temperature
- Steam generator pressure
- Turbine first stage pressure
- Feedwater header pressure
- Main Feedwater Regulating Valve position

Turbine Generator Control

As part of EPU, a new HP turbine rotor is being installed. As indicated previously, this will result in a new predicted 0 – 100% turbine first stage pressure range of 0 – 645 psig. In addition, with the new turbine, the control valve program will be changed from partial arc emission control (load change controlled by sequential valve opening) to full arc emission control (load change controlled by all valves moving together). The turbine controls will require calibration with the new turbine first stage pressure range to provide the appropriate valve position feedback and appropriate valve demand and position indication. New control valve curves will be required for the change to full arc emission control.

The overspeed protection system for the main turbine includes a mechanical overspeed trip mechanism. This device is an eccentric weight mounted on the turbine shaft rotor extension shaft. It is designed to trip the main turbine unit to ensure the turbine speed remains less than the 120% of design speed (2160 rpm). There is also an Overspeed Protection Controller incorporated into the Electro Hydraulic Control (EHC) system. This includes a load drop anticipator and an auxiliary governor function. The load drop anticipator logic will rapidly close all control and intercept valves on a complete loss of load, and rapidly close the intercept valves on a partial loss of load. If the auxiliary governor senses an overspeed condition at 103%, the system will close the reheat intercept valves and modulate close the control valves until the overspeed condition clears.

Presently the turbine mechanical overspeed trip allowable setpoint is less than 110% of rated speed (1980 rpm). An evaluation of the increased mass flow and other EPU hydraulic conditions indicate the allowable overspeed setpoint needs is to be reduced to less than 109.3% rated speed (1969 rpm). Results from overspeed tests performed between 1997 through 2005 indicate the current average overspeed setting to be 108.81% ±0.2%. Since the current setting is less than the new allowable setpoint, the current mechanical overspeed setting is acceptable for EPU.

The load drop anticipator circuit will need to be recalibrated with the EPU 0% - 100% full load megawatts and the reheat crossover pressure to the LP turbines. The 0% - 100% reheat pressure will be recalibrated from the current 0 – 125 psig to 0 – 148 psig.

Plant Computer

The plant process computer system (PPCS) is described in UFSAR section 7.7.6, "Plant Process Computer System and Safety Parameter Display Assessment System." Although EPU will impact the range of many process parameters monitored by the PPCS, the functions performed by the plant computer will not change as a result of EPU. The PPCS inputs associated with the instrumentation changes mentioned above will be rescaled consistent with the range of the PPCS input changes using the station plant change process.

Computer changes associated with the core reload for EPU will be performed in accordance with the cycle specific core reload process.

In-core Instrumentation

The in-core thermal thermocouples (T/Cs) and in-core movable detectors are described in UFSAR section 7.7.4, "In-Core Instrumentation." With respect to EPU, the in-core T/Cs will be exposed to higher core exit temperatures, however, these temperatures are well within the design values for these instruments and will not impact the ability of the in-core thermocouples to perform their design function. With respect to the in-core movable detectors the full power EPU flux levels will be higher, however, it is still within the design capability of the detectors. Therefore, the in-core thermocouples and movable detectors will continue to provide indication as designed.

2.4.1.2.3.4 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Safety related instrumentation or instrumentation that performs a function necessary to accomplish one of the five regulated events are scoped within license renewal, however instruments typically are scoped as active components and are excluded from aging management review. Cables, connectors, pipes and tubes that service the in-scope instruments are passive and require aging management review. The changes to instrumentation for power uprate are predominately rescaling and recalibration of existing instrumentation and introduce no new components or configuration of the instruments. The rescaling and recalibration of these instruments do not impact the design function of the instruments and do not effect the conclusions stated in the licensing renewal evaluations.

For the limited number of cases discussed above, instruments or active instrument components must be changed to ensure the operability of the instruments for EPU conditions. These instrument changes are being performed in accordance with the plant modification process which evaluates the impact of the change with regard to license renewal and aging management.

2.4.1.3 Results

The changes to the instrumentation and controls for EPU are the result of accident and transient analyses and system evaluations to verify the systems and controls will continue to provide the required indication, protection actions, and plant response as originally designed. The changes ensure the DNB values remain within acceptable limits and the RCS pressure boundary and the main steam pressure boundary are maintained within the design values. There are no new protection or control systems required to support EPU. The identified instrumentation recalibration and instrument rescaling will ensure the instrumentation continues to allow monitoring plant process parameters during normal, transient and accident conditions and provide protective functions as required.

2.4.1.4 References

1. Letter from A. R. Johnson (NRC) to R. C. Mecredy (RG&E), Subject: Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3, dated February 24, 1993.

2. Letter from Robert L. Clark (NRC) to Mary G. Korsnick (Ginna), Subject: R. E. Ginna Nuclear Power Plant - Amendment Re: Revision to Core Safety Limits and Safety System Instrumentation Setpoints (TAC No. MB4789), dated September 22, 2004.

3. Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), Subject: License Amendment Request Regarding Adoption of Relaxed Axial Offset Control (RAOC), dated April 29, 2005.

4. Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), Subject: License Amendment Request Regarding Main Feedwater Isolation Valves, dated April 29, 2005.

2.4.1.5 Conclusions

The Ginna staff has reviewed the instrumentation and control systems relevant to the effects of the proposed EPU on the functional design of the reactor protection, safety features actuation, and control systems. The Ginna staff concludes that the evaluation has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis, including the revised load rejection design basis to a rapid ramp load reduction equivalent to 50% rated thermal power at a maximum unloading rate of 200% per minute. The Ginna staff further concludes that the systems will continue to meet the Ginna current licensing basis with respect to the requirements of 10CFR50.55a(a)(1) and 10CFR50.55(a)(h) and GDC-1, GDC-2, GDC-4, GDC-13, GDC-19, GDC-20, GDC-21, GDC-22, GDC-23, GDC-24, GDC-25, and GDC-29. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to instrumentation and controls.

2.4.2 Plant Operability

2.4.2.1 Regulatory Evaluation

The nuclear steam supply system (NSSS) instrumentation and control systems are required to respond to the initiation of design basis plant operational transients without initiating a reactor trip or engineered safety features signal. Ginna Nuclear Power Plant, LLC (Ginna) conducted an evaluation of the NSSS instrumentation and control systems response to design basis operational transients at EPU conditions to ensure the responses remain acceptable.

The acceptance criteria for the NSSS control systems are based on GDC-13, insofar as it requires that instrumentation and control systems be provided to monitor variables and systems over their anticipated ranges during normal operation and anticipated operational occurrences , and maintain these variables and systems within prescribed operating ranges.

Ginna Current Licensing Basis

As noted in the *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the general design criteria (GDC) used during the licensing of the Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna instrumentation and control system design relative to the GDC discussed in UFSAR Sections 3.1.1 and 3.1.2 is discussed in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

The Ginna UFSAR section 7.7.1.1.4, defines the current design basis operational transients that the Ginna Station must be able to sustain without initiating a reactor trip or an engineered safety feature (ESF) actuation signal as:

- Step-load increase of 10%, or ramp-load increase of 5% per minute within the load range of 12.8% to 100% of rated power
- Step-load decrease of 10% or ramp-load decrease of 5% per minute within the load range of 100% to 12.8% of rated power
- Step-load decrease of 245 MWe with steam dump
- Turbine trip below 245 MWe with steam dump

Ginna UFSAR sections that address the design features and functions of the instrumentation and controls of the reactor protection systems include:

- Ginna UFSAR section 7.1.2, Identification of Safety Criteria, which describes the reactor protection and reactor control Instrumentation design basis and the requirements for operability and testability.
- Ginna UFSAR section 7.2, Reactor Trip System (RTS), describes the design criteria for the reactor protection system and provides a description of reactor protection system operation, reactor trips, permissives, and the interaction of the control and protection systems.

- Ginna UFSAR section 7.3, Engineered Safety Features Systems (ESFAS), which describes the design criteria for the ESFAS system and provides a description of the operation, actuation signals, testability, redundancy and independence, and key instrumentation.

- Ginna UFSAR section 7.5, Safety Related Display Instrumentation, identifies the Ginna NSSS and BOP instrumentation subject to the requirements of Regulatory Guide 1.97, Post Accident Monitoring Intrumentation, and documents the NRC evaluation and approval of the Rochester Gas and Electric's position relative to the guidance provided in Regulatory Guide 1.97, Revision 3.

- Ginna UFSAR section 7.7 Control Systems Not Required For Safety, which provides a description of the reactor control system (rod control, steam dump, pressurizer pressure and level, Steam Generator level control and overfill protection, and turbine bypass), plant response to design loading and unloading, and incore instrumentation. Also included in this section is a description of the nuclear instrumentation system from source range to 120% power, reactor coolant temperature indication, the process computer, and the safety parameter display assessment system (SPDS).

In addition to the evaluations described in the UFSAR, the Ginna Station's electrical and I&C systems were evaluated for plant license renewal. The evaluation of the passive portions of electrical and I&C systems, and the subsequent review and conclusions of the NRC, are discussed in section 2.5 of the License Renewal SER, NUREG-1786, dated May 2004.

2.4.2.2 Technical Evaluation

2.4.2.2.1 Introduction

As described in the Ginna UFSAR, Section 7.7.1.1.4, the current design basis operational transients that the Ginna Station must be able to sustain without initiating a reactor trip or an engineered safety feature (ESF) actuation are:

- Step-load increase of 10%, or ramp-load increase of 5% per minute within the load range of 12.8% to 100% of rated power
- Step-load decrease of 10% or ramp-load decrease of 5% per minute within the load range of 100% to 12.8% of rated power
- Step-load decrease of 245 MWe with steam dump
- Turbine trip below 245 MWe with steam dump

Analyses of the design basis transients were performed using the proposed EPU NSSS control system settings and setpoints to demonstrate adequate margin exists to relevant reactor trip and ESF actuation setpoints over the entire range of EPU operating conditions. The Ginna Station EPU operating conditions are shown in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1.

Since 245 MWe will no longer represent 50% load at uprated condition, as part of EPU analyses, the reference to a specific MWe is being omitted from the definition of the design basis step-load decrease and the definition revised as a rapid load decrease equivalent to 50%

of the EPU rated thermal power (RTP) at a maximum turbine unloading rate of 200%/minute. This change in the licensing basis load rejection from a step change to a ramp load change at a maximum rate of 200% per minute redefines the load rejection in a more realistic manner and is consistent with uprating projects previously performed on other Westinghouse plants. For the turbine trip load reject, the reactor is assumed to be below the P-9 permissive which defeats the reactor trip due to turbine trip when nuclear power is less than 50%. Following implementation of EPU, the design basis operational transients will be defined as:

- Step-load change of ±10% or ramp load change of 5% per minute within the load range of 12.8% to 100%
- A rapid load decrease equivalent to 50% rated thermal power (RTP – 1817 MWt) at a maximum turbine unloading rate of 200% per minute with steam dump
- Turbine trip below 50% reactor power (P-9) with steam dump

The 5% per minute loading and unloading transients are not limiting transients and are enveloped by analysis of the other design transients, therefore no specific analysis was performed for the 5% per minute loading and unloading transient.

The analyses were performed using the Westinghouse LOFTRAN computer code. This computer code is a system-level program code and models the overall NSSS, including the detailed modeling of the control and protection systems. A LOFTRAN computer model was developed of the Ginna Station at the EPU conditions. The degree of limitation of the above transients was analyzed to show that it resulted in acceptable plant response. Details of the analyses are described below.

2.4.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The following assumptions were made for all normal transients analyzed:

- All applicable NSSS control systems were assumed to be operational and in the automatic mode of control (that is, rod control, steam dump control, pressurizer level, steam generator level control, and pressurizer pressure control).
- A 2% initial power level uncertainty is already included in the LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1, full-power operating conditions, so no additional power level uncertainty was assumed. The remainder of the plant parameters (that is, reactor coolant system (RCS) T_{avg}, pressurizer pressure, pressurizer level, steam generator level) were assumed to be at their nominal post EPU control system setpoints.
- Best-estimate reactor kinetics parameters were modeled (that is, rod worth, moderator temperature coefficient (MTC), Doppler power defect, etc.). Since beginning-of-life (BOL) core physics parameters have lower differential rod worth and a less negative MTC, modeling BOL core characteristics yield more conservative results that bound the full cycle of operation.
- A 10% steam generator tube plugging (SGTP) was assumed for all transients except the 10% load decrease which used 0% SGTP. In general, analysis of 10% SGTP conditions

bounds the 0% tube plugging conditions. Higher SGTP is somewhat more conservative for short-term heatup transients due to a slower rate of heat transfer from the primary to secondary side of the plant. Furthermore, lower nominal steam temperatures and pressures reduced the steam dump capacity during heatup transients, and reduced the margin to safety injection (SI) actuation on low steam pressure during cooldown transients.

- The transient duration was a 500-second interval (about 8 minutes). Most challenges to the reactor trip and ESF actuation setpoints occurred within the first minute of the initiation of the design basis normal condition transients, therefore this simulation time frame was considered more than adequate for assessing control system response and stability considerations.

- The analysis used current (as-installed) steam dump valve capacities.

- The analyses assumed reactor protection and control system settings initially derived from the EPU accident and transient analyses discussed in LR section 2.8.5, Accident and Transient Analyses. These changes are listed below and specifically described in detail in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.
  - Proposed Technical Specification overtemperature and overpower ΔT constant values
  - 564.6°F low T_{avg} pressurizer level program for all analyses except the 10% load step decrease which used the 576°F high T_{avg} pressurizer level program
  - Changes in the rod control non-linear gains in the power mismatch circuits
  - Installation of a 3.5 second filter on the T_{hot} input to the Tavg and ΔT calculation
  - Changes to the steam dump control system temperature dead band, ΔT snap open values, and the ΔT proportional gains.
  - Change to the steam dump interlock mode of control from turbine trip to reactor trip signal.

- The reactor protection setpoints used in this analyses are:

High-pressurizer pressure reactor trip:	2377 psig	
Low-pressurizer pressure reactor trip:	1873 psig	
Low-pressurizer pressure SI:	1750 psig	
Low-steamline pressure:	514 psig	Lead time constant = 12 seconds
		Lag time constant = 2 seconds
Low T_{avg}:	545°F	

2.4.2.2.3 Description of Analyses and Evaluations

Design operational transients were performed using the current capacity of the condenser steam dumps and the revised rod control and steam dump settings. With the revised settings, these analyses achieved acceptable results. The required changes to rod control and steam dump systems are described in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

The 50% load rejection transient was analyzed as a rapid ramp load decrease of 200% per minute rather than a step change. This is more realistic and representative of an actual load rejection transient in the plant and is consistent with uprating projects previously performed on other Westinghouse plants.

10% Step-Load Decrease

The 10% step-load decrease transient is intended to avoid the plant from reaching the pressurizer power-operated relief valve (PORV) setpoint. The 10% load decrease transient was analyzed as part of the pressurizer pressure control component sizing analysis described in LR section 2.4.3, Pressurizer Component Sizing. The analyses performed for the spray capacity included additional conservatisms not normally used in the plant operability analyses (that is, T_{avg} uncertainty of 4°F) and, therefore, enveloped the best-estimate analyses normally used in the plant operability analyses. The results indicated that no reactor trip setpoints were challenged and the control system response was stable and not oscillatory. Pressurizer pressure reached a maximum of 2317 psig for the high T_{avg} case which is less than the 2335 psig PORV setpoint, therefore the PORVs were not challenged. Therefore, the plant response for the 10% step-load decrease transient is acceptable for the EPU.

10% Step-Load Increase

This transient was analyzed to verify that there is adequate margin to the low pressurizer pressure reactor trip setpoint and the engineered safety features actuation function on low-steamline pressure. This transient was analyzed as a step-load increase from 90% to 100% power with all NSSS control systems active, except the steam dump control system (steam dump is not activated for 10% step-load changes). The transient was analyzed for the limiting case, low T_{avg} (564.6°F), 10% SGTP condition as it has the minimal margin to the low-steamline pressure setpoint and demonstrates that ESF actuation will not occur on low steam pressure.

The control system response was smooth during the transient with no oscillatory response noted. No reactor trip or ESF actuation setpoints were challenged. The steam pressure reached a minimum of approximately 613 psig (lead/lag compensated), which is greater than the low steamline flow SI actuation setpoint of 514 psig. The minimum pressurizer pressure reached was approximately 2203 psig, which is greater than the low-pressure reactor trip setpoint of 1873 psig. Pressurizer level dropped to approximately 36% of span due to the cooldown, which is well above the low-level heater cutoff setpoint of 13% of span. The low T_{avg} value reached was approximately 560°F and is above the low T_{avg} setpoint of 545°F. Therefore, the plant response for the 10% step-load decrease transient is acceptable for the EPU.

50% Load Rejection

A 50% load rejection transient is the most severe operational transient that the plant would normally undergo without potentially actuating a reactor trip or engineered safety features function. This transient was analyzed as a rapid change in turbine load from 100% to 50% of the nominal power level at a maximum 200% per minute turbine unloading rate. The 200% per minute transient is the fastest unloading rate that the turbine can normally perform, so this was used in the analysis. The low 564.6°F T_{avg}, 10% SGTP was used in the analysis since it provided the lowest steamline pressure operating condition and the lower steam dump capacity, resulting in the largest initial NSSS heatup and lowest margin to reaching a reactor trip setpoint.

Based on the analyses results (reference Figures 2.4.2-1 through 2.4.2.-5) using the revised rod control and steam dump setpoints, a 50% rapid load reduction at a turbine runback of 200% per minute can be sustained for full-power T_{avg} values of 564.6°F and above. The PORVs will open for all cases analyzed and limit the pressurizer pressure which is acceptable. The peak pressurizer pressure was controlled by the pressurizer PORV actuation at the PORV actuation setpoint value of 2335 psig, thereby preventing the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint of 2377 psig and showing acceptable capacity for the pressurizer PORVs. The minimum predicted pressure of all the cases analyzed is approximately 2002 psig, therefore the low-pressurizer pressure reactor trip setpoint of 1873 psig is not challenged. During the transient, pressurizer level remained less than the pressurizer high level trip setpoint of 87% for both the low T_{avg} (66.7%) and the High Tavg (83%) cases.

The margin to the limiting reactor trip setpoints (OTΔT and OPΔT) was a minimum of 2.0%. The peak steam pressure is approximately 890 psig, which is less than the no-load steam pressure of 1005 psig. Therefore, with the revised rod control and steam dump system settings the response to a 50% rapid load reduction transient will be acceptable for the EPU.

Turbine Trip without Reactor Trip

This transient was analyzed to verify that the pressurizer PORVs are not challenged on a turbine trip below the P-9 setpoint of 50% of RTP. This transient was modeled as a step-load decrease in turbine load from 50% of the nominal power level to 0% power with all NSSS control systems active. The transient was analyzed for the low T_{avg}, 10% SGTP condition as this condition has the minimum steam pressure, therefore, the lowest steam dump capacity that results in the highest pressurizer insurge flow and highest pressurizer pressure.

The peak-pressurizer pressure reached for a full-power T_{avg} value of 564.6°F was approximately 2331 psig, in comparison with the pressurizer PORV setpoint of 2335 psig. The maximum Pressurizer level during the transient was less than 46% compared to the pressurizer level trip setpoint of 87%. The peak secondary steam pressure remained less than the no-load steam pressure of 1005 psig. Therefore, the plant response for turbine trip without reactor trip from the P-9 setpoint or below is acceptable for the EPU.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The components of the NSSS instrumentation and control system are treated for license renewal purposes as a commodity group discussed in the License Renewal SER, NUREG-1786, section 2.5, "Electrical and Instrumentation and Controls Systems." The aging management programs applicable to this commodity group is discussed in SER section 3.6, "Electrical and Instrumentation and Controls." EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operation of components of the NSSS instrumentation and control systems at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus, no new aging effects requiring management are identified.

2.4.2.3 Results

These analyses concluded that the changes to the reactor protection and reactor controls identified in Section 2.4.2.2.2, Input Parameters, Assumptions, and Acceptance Criteria, above, will enable the plant to continue to satisfy the requirements of the design operational transients listed below.

5%/Minute Loading and Unloading

Acceptable results were obtained for the 10% step-load increase and decrease, 50% load rejection, and turbine trip without reactor trip transients which envelop the 5% per minute loading and unloading transients. Therefore, the response to a 5% per minute unit loading and unloading transients at EPU are acceptable.

10% Step-Load Decrease

The design basis for this transient is that a 10% step-load decrease transient can be accommodated without challenging the pressurizer PORVs or resulting in a reactor trip or ESF actuation. The analyses show the pressurizer pressure remains below the PORV setpoint and the control system response was smooth during the transient with no oscillatory response noted. Therefore, the response to a 10% step-load decrease transient at EPU is acceptable.

10% Step Load Increase

The analyses show the primary pressure remained well above the low pressurizer pressure trip and the main steam pressure remains above the low steam line pressure ESFAS actuation. The analyses indicate the control system response was smooth during the transient with no

oscillatory response noted. Therefore, the response to 10% step-load increase transient at EPU is acceptable.

50% Load Rejection

The design basis for this transient is that a 50% rapid ramp load reduction transient can be accommodated without challenging any reactor trip or ESF. The analyses for the most limiting case shows the pressurizer PORVs open and limit the pressurizer pressure to less than the high pressure trip and the minimum predicted pressurizer pressure remains well above the low pressure trip. Pressurizer level remains less than the high level trip setpoint. In addition, the secondary steam pressure remains less than the no load steam pressure during the transient. The analyses indicate the control response is smooth and stable. Therefore the response to a 50% rapid ramp load reduction at EPU is acceptable.

Turbine Trip without Reactor Trip

The design basis for this transient is that a turbine trip without reactor trip transient actuated from below the P-9 setpoint (50% reactor power) can be accommodated without challenging the pressurizer PORVs or resulting in a reactor trip or ESF actuation. While not a design requirement, it is desirable to avoid actuating the steam generator power operated atmospheric relief valves (ARVs) and Main Steam Safety Valves (MSSVs) as well. The analyses show these requirements are met. Pressurizer pressure remains less than the pressurizer PORV setpoint therefore, the pressurizer PORVs are not challenged and the pressurizer level remains less than the high pressurizer level trip setpoint. The main steam pressure remains less than the no load steam pressure therefore, the steam generator ARVs and MSSVs setpoints are not challenged. The analyses show the control system responses are smooth and stable. Therefore, the response to a turbine trip at less than 50% load at EPU is acceptable.

Redefining the Step Load Decrease

Removing the MWe specific reference in the design basis load step decrease and turbine trip definition and defining the design basis 50% load step decrease as a rapid ramp load decrease equivalent to 50% of the EPU rated thermal power (RTP) at a maximum turbine unloading rate of 200% per minute is consistent with the assumptions and methodologies used in the supporting analyses above. This change in definition is consistent with uprating projects previously performed on other Westinghouse plants.

The analyses show that a rapid ramp load decrease equivalent to 50% of rated thermal power and a turbine trip from less than 50% of rated thermal power can be supported over the entire EPU T_{avg} range identified in Table 1-1 of LR section 1.1, Nuclear Steam Supply System Parameters, once the changes to the instrumentation and controls identified above and described in LR section 2.4.1., Reactor Protection, Safety Features Actuation, and Control Systems are implemented.

2.4.2.4 Conclusion

The Ginna staff has reviewed the effects of the proposed EPU on the plant capability of meeting its response to design basis operational transients. The Ginna staff concludes that it has

adequately addressed the effects of the proposed EPU on the plant operational capability and that the changes that are necessary to achieve satisfactory results at EPU are consistent with the plant's design basis including the revised 50% load rejection definition. to a rapid load ramp decrease equivalent to 50% of rated thermal power at a maximum unloading rate of 200% per minute. Therefore, Ginna finds that, with appropriately revised control settings, the responses of the plant to operational transients at the proposed EPU are acceptable with respect to the plant capability of meeting its design basis operational transients and continuing to meet the current licensing basis with respect to the requirements as specified in GDC-13.

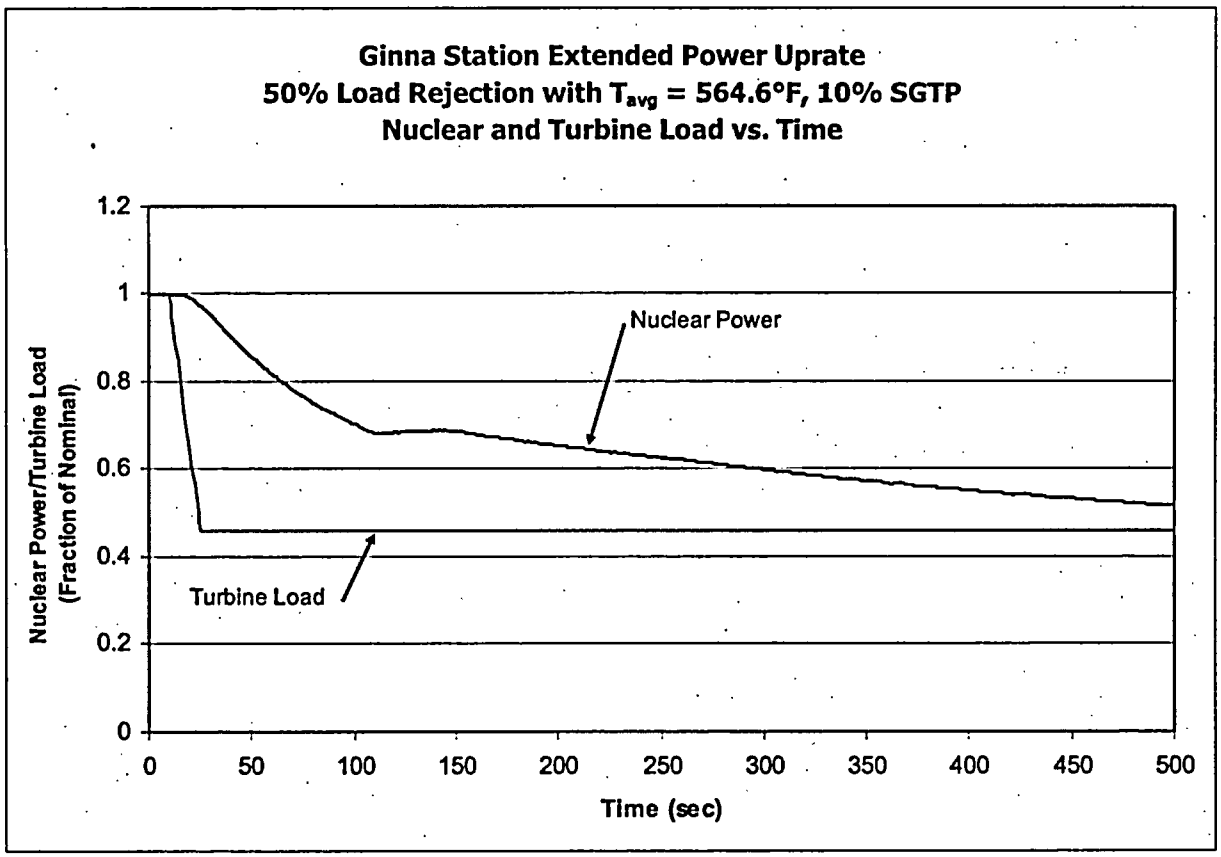


Figure 2.4.2-1
50% Load Rejection, Nuclear and Turbine Load vs. Time

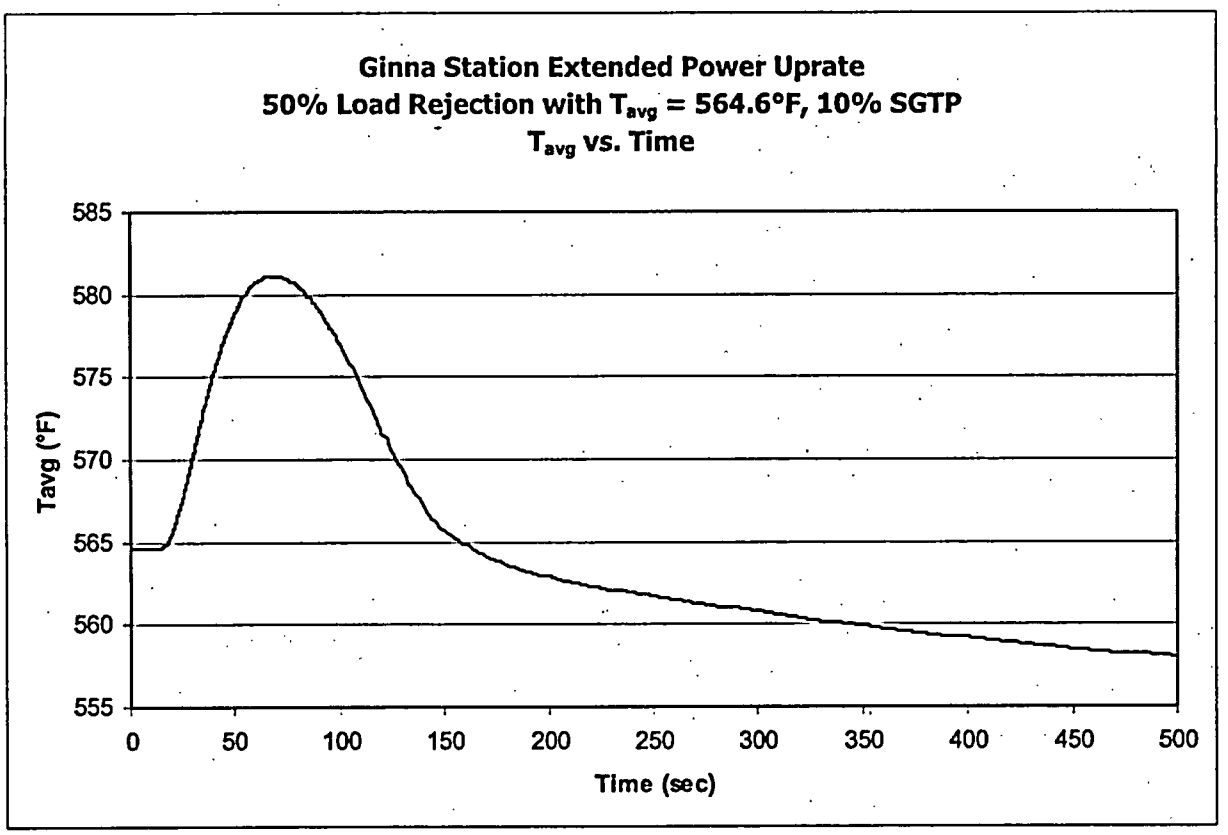


Figure 2.4.2-2

50% Load Rejection, T_{avg} vs. Time

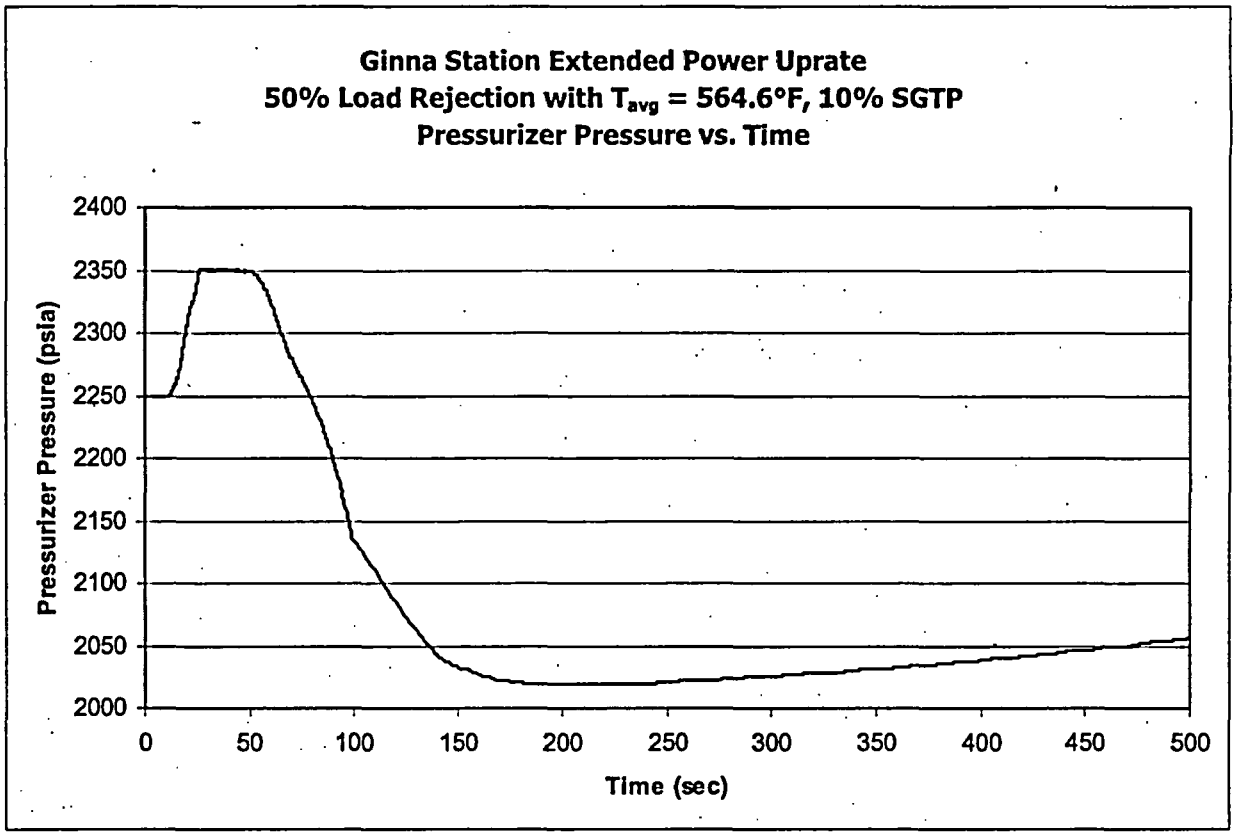


Figure 2.4.2-3

50% Load Rejection, Pressurizer Pressure vs. Time

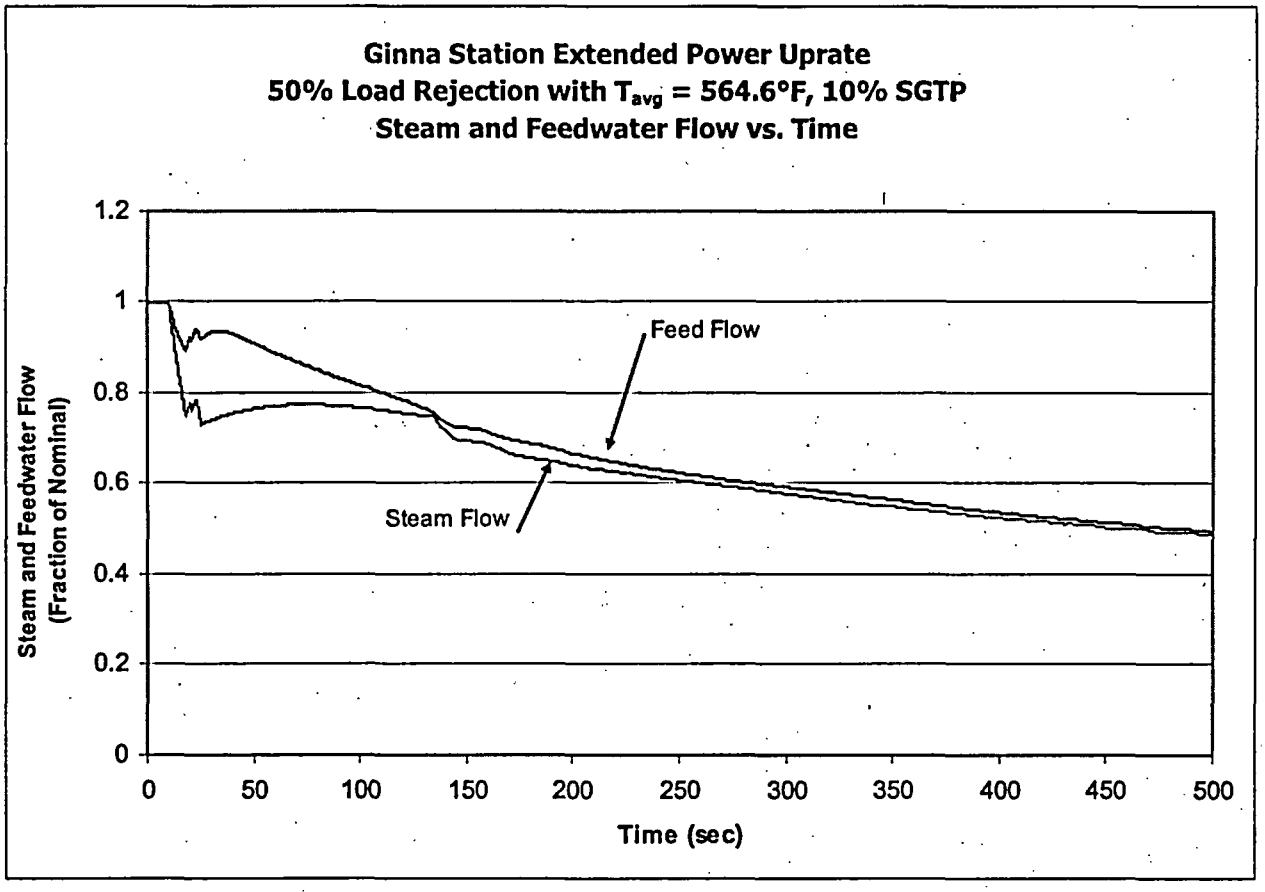


Figure 2.4.2-4

50% Load Rejection, Steam and Feedwater Flow vs. Time

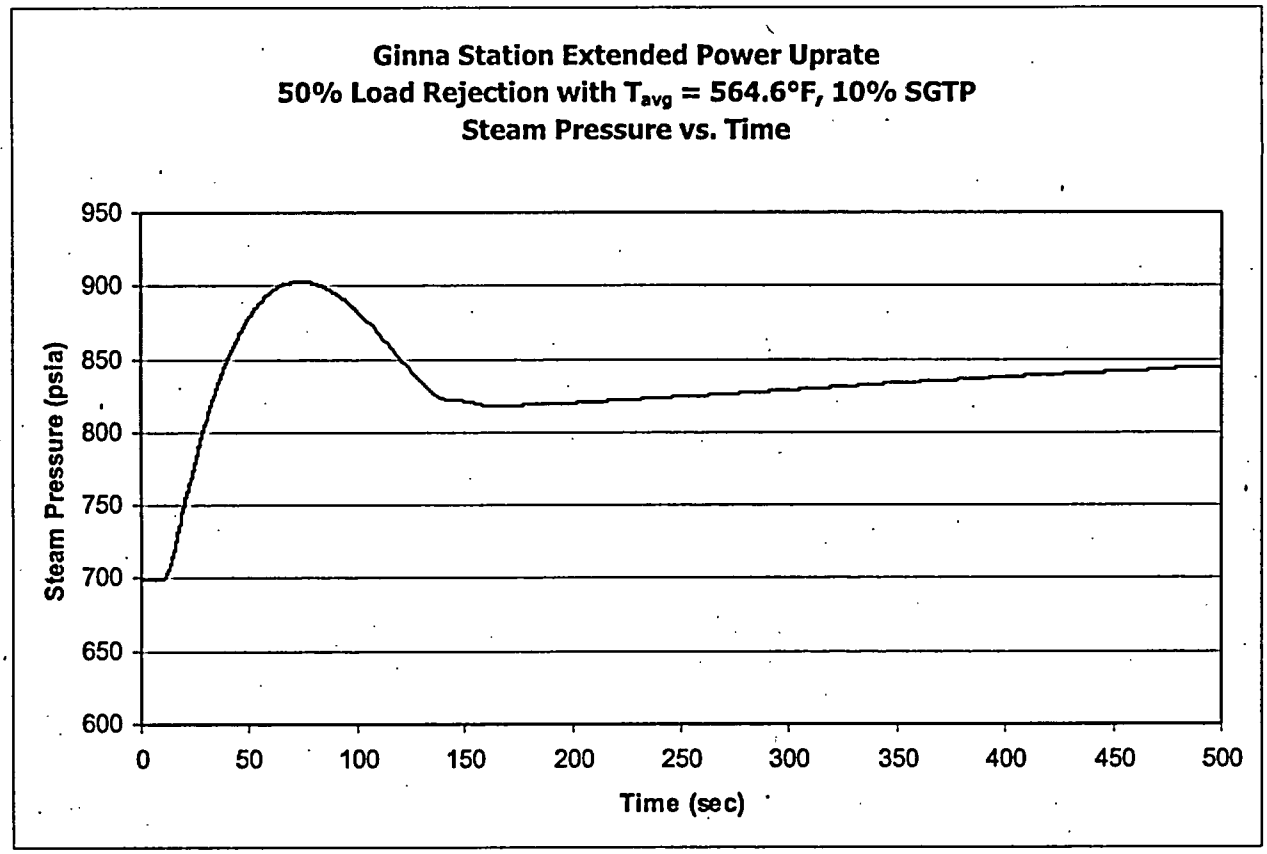


Figure 2.4.2-5

50% Load Rejection, Steam Pressure vs. Time

2.4.3 Pressurizer Component Sizing

2.4.3.1 Regulatory Evaluation

The pressurizer pressure control system (consisting of the pressurizer heaters, spray, and power operated relief valves (PORVs)) provides the means of controlling the pressurizer pressure to less than the design basis setpoint value during steady-state operation and to minimize the pressurizer pressure excursions during design basis operational transients. Ginna Nuclear Power Plant, LLC (Ginna) conducted a review of the pressurizer pressure control system for the EPU to ensure that the system, and any changes necessary for the EPU, are adequately designed so that they continue to meet their design basis operational functions. The acceptance criteria related to the quality of design of the pressurizer pressure control systems are based on:

- 10CFR50.55a(a)(1), insofar as it requires that safety-related structures, systems, and components (SSCs) be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.

- 10CFR50.55a(h), insofar as it requires that protection systems must be consistent with the plant's current licensing basis.

- GDC-1, insofar as it requires that safety-related structures, systems, and components (SSCs) are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

- GDC-19, insofar as it requires that a control room be provided from which actions can be taken to operate the nuclear unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs).

- GDC-24, insofar as it requires that the protection system be separated from the control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection systems. Interconnection of the protection and control systems will be limited so as to ensure that safety is not significantly impaired.

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Ginna Current Licensing Basis

As noted in Ginna UFSAR Section 3.1 the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station safety-related SSCs will continue to be protected against the failure of the pressurizer pressure control systems and components consistent with the following:

- 10CFR50.55a(a)(1) is described in UFSAR section 3.2.1, "Classification of Structures, Components, and Systems – Introduction." As part of SEP Topic III-1, the original codes and standards used in the design of the Ginna Station were compared with later licensing criteria, including 10CFR50.55a. The objective was to assess the capability of Ginna Station SSCs to perform their safety functions as judged by the later standards. Although several areas were identified where requirements had changed, all areas were satisfactorily resolved, and SEP Topic III-1 was closed. This SEP review established a plant licensing basis for conformance with 10CFR50.55a(h).

- GDC-1 is described in Ginna UFSAR section 3.1.2.1.1, "GDC-1 – Quality Standards and Records." As described in this UFSAR section, safety-related SSCs are designed, fabricated, inspected and erected, and the materials selected to the applicable provisions of the then recognized codes, good nuclear practice, and to quality standards that reflected their importance. Additionally, as described in UFSAR section 5.1.3.1, the reactor coolant system (RCS), which includes the pressurizer pressure control components (pressurizer heaters, spray valves, and power operated relief valves (PORVs)), conform to the applicable provisions of approved codes and good nuclear practice.

- GDC-13 is described in Ginna UFSAR section 3.1.2.2.4, "GDC-13 – Instrumentation and Control". As described in this UFSAR section, instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain various process variables including RCS pressure within prescribed operating ranges. Additionally, as described in UFSAR section 7.7.1.3.1, RCS pressure is maintained at constant value through operation of the pressurizer pressure control system. The components of the pressurizer pressure control system operate in conjunction with the pressurizer pressure control I&C logic to

control the pressurizer pressure during plant steady-state operation and during design basis transients.

- GDC-19 is described in Ginna UFSAR section 3.1.2.2.10, "GDC-19 – Control Room. Instrumentation and Control". As described in this UFSAR section, Ginna Station has a control room which contains controls and instrumentation necessary for the operation of the reactor and turbine generator under normal and accident conditions. However, the pressurizer pressure control system and its components are not safety related, as indicated by its description being included in UFSAR section 7.7, "Control Systems Not Required For Safety," (see UFSAR sections 7.7.1.1.5 and 7.7.1.3.1).

- GDC-24 is described in Ginna UFSAR section 3.1.2.3.5, "GDC-24 – Separation of Protection and Control Systems". As noted in this section, the protection system shall be separated from the control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection systems. Interconnection of the protection and control systems will be limited so as to ensure that safety is not significantly impaired. This is met by having the Reactor Trip System (RTS) physically and electrically separate from the control system such that the failure of any single control component or channel, or its removal from service, still leaves intact a system satisfying the reliability, redundancy, and independence requirements of the Reactor Trip System, as noted in UFSAR section 7.2.5, "Interaction of Control and Protection Systems." This UFSAR section describes the NRC evaluation of the Ginna Station RTS that was performed as part of the SEP, Topic VII-1A. The NRC concluded that the RTS is adequately isolated from non-safety systems (such as the pressurizer pressure control system) and satisfies the criteria set forth in GDC-24.

In addition to the evaluations described in the Ginna UFSAR, the aging effects requiring management were assessed for plant license renewal for the pressurizer, spray valves, and PORVs. The adequacy of these components for license renewal is documented in the License Renewal SER, NUREG-1786, under the overall pressurizer evaluation described in SER section 2.3.1.4, "Pressurizer". The impact of EPU on the License Renewal evaluation for the pressurizer, spray valves, and PORVs is discussed in LR section 2.2.2, "Plant Level Scoping Results - Evaluation."

2.4.3.2 Technical Evaluation

2.4.3.2.1 Introduction

As part of the EPU, the following pressurizer components were evaluated to ensure that the nuclear steam supply system (NSSS) pressure control system is adequate for the increased pressures and temperatures for the uprate conditions shown in LR section 1.1, "Nuclear Steam Supply System Parameters."

- Pressurizer power-operated relief valves (PORVs)

- Pressurizer spray valves
- Pressurizer heaters

To support the Ginna EPU, Westinghouse recommended new rod control and steam dump settings described in LR section 2.4.1, "Reactor Protection, Safety Features Actuation, and Control Systems." The recommended new setpoints for rod control were used in this pressurizer pressure control component sizing analysis. For conservatism, the current steam dump settings were used in this analysis since the recommended new steam dump settings cause the steam dump valves to open sooner resulting in lower pressurizer pressures.

The components comprising the pressurizer pressure control system are described in the Ginna UFSAR sections 5.3.2.5, 5.4.7.1, 5.4.10.1, and 7.7.1.3.1. Based on these descriptions, the following bases were used to evaluate the sizing acceptability of the various components of the pressurizer pressure control system.

Pressurizer PORVs

The sizing basis for the pressurizer PORVs is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis large step-load decrease with steam dump transient. This design basis load rejection is defined as a 50% step-load decrease from 100% to 50% power.

Pressurizer Spray Valves

The sizing basis for the pressurizer spray valves is to prevent challenges to the pressurizer PORVs for a 10% step-load decrease transient. For load decreases up to 10% power, the spray valves are the prime means of controlling pressure without actuating the pressurizer PORVs when in automatic pressure control mode.

Pressurizer Heaters

The pressurizer heaters are sized to be able to heat up the pressurizer liquid at a maximum rate of a 55°F per hr during the initial plant heatup phase from cold shutdown. In addition, they are intended to assist the plant in controlling the pressurizer pressure decrease that would occur during design basis transients that result in pressurizer outsurge events. These include the initial part of a 10% step-load increase transient, a 5% per minute plant unloading transient, or events resulting in a reactor trip. The design basis pressurizer heater capacity to meet this capacity is 1 kW/ft^3 of heater capacity per cubic foot of pressurizer volume.

The originally installed heater capacity was 800 kW, 400 kW from the proportional heaters plus 400 kW from the backup heaters. The pressurizer internal volume is 800 ft^3, therefore, the sizing basis of 1 kW/ft^3 was met (800 kW/800 ft^3 = 1 kW/ ft^3). Subsequent to the initial pressurizer heater installation, the operating heater capacity was reduced to approximately 760 kW. The installed heater capacity is still acceptably close to the design basis sizing requirement of 1 kW/ft^3 and has shown at the current conditions to be sufficient to maintain the pressurizer pressure at its setpoint during steady-state operation and to minimize pressure excursions during design basis operational transients.

2.4.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Pressurizer PORVs

The pressurizer PORV sizing analysis was performed at the Ginna Station EPU operating conditions shown in LR section 1.1, "Nuclear Steam Supply System Parameters." The analysis was performed to envelope the window of operating conditions, full power T_{avg} of 564.6°F to 576°F and zero to maximum steam generator tube plugging (SGTP) levels.

With the NSSS power uprate to 1817 MWt (1811 MWt reactor power), the demand on the pressurizer PORVs would tend to increase. Therefore, the pressurizer PORV sizing analysis was performed to ensure acceptability. The analysis was performed following the general guidelines and methodology presently in use. This included the following key input parameters and assumptions listed below:

- The transient is conservatively modeled as a 50% step load reduction from 100% to 50% power. 100% power corresponds to 1811 MWt reactor power.
- A maximum steam pressure condition will provide the maximum pressurizer insurge, therefore, a 0% steam generator tube plugging (SGTP) level is used in the analysis.
- The plant is initially at nominal high T_{avg} plus a 4.0°F uncertainty.
- The initial pressurizer pressure is at nominal pressure of 2250 psia.
- The initial pressurizer water level is at the nominal setpoint applicable to the high T_{avg} operating conditions.
- The steam generator heat transfer coefficient is increased to the maximum credible value (0% fouling and 0% tube plugging).
- The pressurizer PORV installed capacity is 179,000-lb/hr saturated steam per valve at 2335 psig. There are a total of two valves.
- The NSSS control systems actuation logic and corresponding setpoints for rod control, steam dump control, pressurizer pressure and level control, and steam generator level control systems were used. A conservative value of 24.3% of the rated steam flow for the steam dump capacity was used in the analysis.
- Best-estimate nuclear design parameters (moderator temperature coefficient, Doppler power defect, control rod worth, and startup data) at conservative beginning-of-life (BOL) conditions were assumed.

The acceptance criterion was that the installed pressurizer PORV capacity should be sufficient to limit the peak pressurizer pressure to a value below the installed high pressurizer pressure reactor trip setpoint of 2377 psig during the design basis load rejection with steam dump transient.

Pressurizer Spray Valves

The pressurizer Spray Valve sizing analysis was performed at the Ginna Station EPU operating conditions discussed in LR section 1.1, "Nuclear Steam Supply System Parameters." With the uprating, the demand on the pressurizer spray valves would tend to increase. Therefore, the pressurizer spray sizing analysis was performed to ensure acceptability at the uprated conditions. The analysis was performed following the general guidelines and methodology presently in use. This included the following key input parameters and assumptions listed below:

- The transient is conservatively modeled as a 10% step-load reduction from 100% to 90% power.
- The plant is initially at nominal high T_{avg} plus a 4.0°F uncertainty.
- The initial pressurizer pressure is at nominal pressure of 2250 psia.
- The initial pressurizer water level is at the nominal setpoint applicable for the high T_{avg} operating conditions.
- The steam generator heat transfer coefficient is increased to the maximum credible value (0% fouling and 0% tube plugging).
- The installed pressurizer spray valves capacity was analyzed at 200 gpm per valve. There are two valves for a total capacity of 400 gpm.
- The NSSS control systems actuation logic and corresponding setpoints for rod control, pressurizer pressure and level control, and steam generator level control systems are included. Steam dump is not actuated for a 10% step-load decrease transient; therefore, steam dump is not credited for this analysis.
- Best-estimate nuclear design parameters (moderator temperature coefficient, Doppler power defect, control rod worth, and startup data) at conservative BOL conditions are assumed.

The acceptance criterion was that the total installed capacity (400-gpm total) of the pressurizer spray valves should be adequate to limit the peak pressurizer pressure to less than the pressurizer PORV actuation setpoint of 2335 psig on a 10% step-load decrease transient.

Pressurizer Heaters

The general sizing basis for the pressurizer heaters is that the heater capacity is 1 kW/ft^3 of pressurizer total volume. The current heater capacity of 760 KW is acceptably close to the 1 kW/ft^3 and was used in the analyses. However, the acceptance criterion in the sizing of the pressurizer heaters is that the low pressurizer pressure reactor trip is not actuated during the design basis operational transient which is assessed in LR section 2.4.2, Plant Operability.

2.4.3.2.3 Description of Analyses and Evaluations

Pressurizer PORVs

A 50% step-load decrease with steam dump transient was analyzed using the configured version of the LOFTRAN computer code. This computer code is a system-level program code and models the overall NSSS including the detailed modeling for control and protection systems. A LOFTRAN computer model was developed for the Ginna Station two-loop plant. The 50% step-load decrease transient is loop-symmetric; the single-loop version of the LOFTRAN code was used for this analysis. The analysis was performed for the limiting case for this sizing analysis and corresponds to the plant operating at the upper limit T_{avg} of 576°F, since it was an RCS heatup transient.

Pressurizer Spray Valves

A 10% step-load decrease from full-power transient was analyzed using the configured version of the LOFTRAN computer code. A LOFTRAN computer model was developed for the Ginna Station two-loop plant. The method of analysis was similar to the standard sizing procedure for pressurizer spray valves used in the original sizing calculations. The limiting case was for plant operation at the upper limit T_{avg} of 576°F.

Pressurizer Heaters

The required heater capacity was not affected by the EPU conditions. Design basis transients resulting in pressurizer insurges/outsurges such as loadings/unloadings, load rejections, and reactor trips showed pressurizer pressure changes that were too rapid for the pressurizer heaters to influence. Generic analyses (as well as plant operating experience) on Westinghouse plants have shown that the pressurizer heater capacity is not a strong influence on the minimum pressure noted during the above operational events, or during reactor trips. The minimum pressure is controlled by the outsurge that results during the transient. In addition, analyses done on other plants have shown very little difference in the maximum/minimum pressurizer pressure when it was assumed that a certain percentage of the pressurizer heaters were out of service. Analyses have been performed where the pressurizer heater capacity has been reduced by as much as 20% and no major difference has been observed in the analysis results. The heatup time from cold shutdown to hot standby is not affected by the uprate. The heatup maneuver would be essentially the same as that which the Ginna Station presently experiences.

The general sizing basis for the pressurizer heaters is that the heater capacity is 1 kW/ft^3 of pressurizer total volume. However, the acceptance criterion in the sizing of the pressurizer heaters is that the low pressurizer pressure reactor trip is not actuated during the design basis operational transients. This is assessed in LR section 2.4.2, "Plant Operability," and shown to be satisfied.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The components of the pressurizer pressure control system were evaluated for aging effects requiring management for license renewal purposes. The adequacy of these components for license renewal is documented in the License Renewal SER, NUREG-1786, section 2.3.1.4, "Pressurizer." The system components are subject to existing aging management programs which are described in the License Renewal SER section 3.2, "Reactor Coolant Systems." EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operation of components of the pressurizer pressure control systems at EPU conditions are instrument setpoint changes that do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus, no new aging effects requiring management are identified.

2.4.3.3 Results

Pressurizer PORVs

The maximum pressurizer pressure resulting from the analysis of the transient was 2352 psia or 2337 psig, which is less than the high pressurizer pressure reactor trip setpoint of 2377 psig.

The pressurizer PORVs had sufficient relief capacity to avoid a reactor trip on high pressurizer pressure for the design basis load rejection for the Ginna Station EPU conditions. The PORVs were adequately sized for the EPU conditions and no limitations to the plant operating conditions are required.

Pressurizer PORVs are not subjected to water-solid conditions during non-LOCA transients. LR section 2.8.5.2.2, Loss of Non-emergency AC Power, and LR section 2.8.5.2.3, Loss of Normal Feedwater show that water-solid conditions in the pressurizer do not occur under those transients. LR section 2.8.5.0 addresses the ability of the RETRAN computer code that was used for these analyses to accurately determine the occurrence of water-solid conditions. LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems describes the ability of the pressurizer level control system to maintain the level low enough that a sufficient steam volume is maintained to ensure the pressurizer does not go solid during non-LOCA transient conditions.

Pressurizer Spray Valves

For the limiting case analyzed (high T_{avg}, 0% SGTP level), the results showed a maximum peak pressurizer pressure of 2332 psia or 2317 psig, which is less than the pressurizer PORV actuation setpoint of 2335 psig.

Since the peak pressurizer pressure was less than the PORV actuation setpoint of 2335 psig, the total installed capacity of 400 gpm is adequate to avoid actuation of the PORVs during a 10% step-load decrease transient for the uprated conditions.

Pressurizer Heaters

The plant operability analyses described in LR section 2.4.2, "Plant Operability," show that the pressurizer pressure is able to be maintained above the low pressurizer pressure reactor trip setpoint during the design basis operational transients. Therefore, for EPU, the current heater capacity remains sufficient to maintain the pressurizer pressure at its setpoint during steady-state operation and to minimize pressure excursions during design basis operational transients.

2.4.3.4 Conclusion

The Ginna staff has evaluated the effects of the proposed EPU on the functional design of the NSSS pressurizer pressure control systems. The Ginna staff concludes that the evaluation adequately addresses the effects of the proposed EPU on these systems and that the other changes that are necessary to the reactor control systems to achieve the proposed EPU, are consistent with the pressurizer pressure control systems' design basis. The Ginna staff further concludes that the pressurizer pressure control systems will continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR50.55a(a)(1) and 10CFR50.55(a)(h), and GDC-1, GDC-13, GDC-19, and GDC-24. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the pressurizer pressure control systems.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

2.5.1.1.1.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC's (Ginna) review in the area of flood protection to ensure that safety-related structures, systems, and components are protected from flooding covered flooding of safety-related structures, systems, and components from internal sources, such as those caused by failures of tanks and vessels. The review focused on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on General Design Criterion 2. Specific review criteria are contained in the Standard Review Plan Section 3.4.1.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to General Design Criterion 2 is addressed in Ginna UFSAR Section 3.1.2.1.2, "General Design Criterion 2 – Design Bases for Protection Against Natural Phenomena." As noted in that section, flood protection measures are discussed in Ginna UFSAR Section 3.4.

Internal flooding from sources other than high energy line breaks and moderate energy line cracks, including internal flooding due to failure of tanks and process equipment, is addressed in the following Ginna UFSAR sections and correspondence with the NRC:

- UFSAR section 3.4.2, Flooding Due to Failure of Tanks

- UFSAR section 3.11.3.2.4, Flooding

- UFSAR section 3.11.3.6, Diesel Generator Rooms
- UFSAR section 3.11.3.8, Auxiliary Building Annex

- Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, "SEP Topic III-6, Seismic Qualification of Tanks," June 16, 1983.

- Letter from D. M. Crutchfield, NRC to J. E. Maier, RG&E, "Integrated Plant Safety Assessment Report (IPSAR), Section 4.25.3, Flooding Due to Failure of Tanks," July 8, 1983.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

During plant license renewal evaluations, tanks and pipes, which were not already in-scope pursuant to 10CFR54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (criterion (a)(2)). Components which met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

2.5.1.1.1.2 Technical Evaluation

2.5.1.1.1.2.1 Introduction

This section addresses protection from internal flooding outside containment from sources other than from high energy line breaks and moderate energy line cracks, including internal flooding due to failure of tanks, vessels, and process equipment. Internal flooding is also addressed in the following sections:

- Internal flooding due to high energy line breaks and moderate energy line cracks in the intermediate building and turbine building is addressed in LR section 2.5.1.3, "Pipe Failures."

- Submergence inside containment is addressed in LR section 2.3.1, "Environmental Qualification of Electrical Equipment."

- Protection of the control building from flooding due to a break / leakage in the circulating water system, and protection from internal flooding in the turbine building and screen house, are addressed in LR section 2.5.1.1.3, "Circulating Water System."

- Internal flooding is also addressed in LR section 2.5.1.1.2, "Equipment and Floor Drains."

Internal Flooding Due to Failure of Tanks, Vessels, and Process Equipment

The plant areas in which safety-related components are potentially affected by flooding of nonseismic tanks are the auxiliary building and the auxiliary building annex. These areas are addressed below.

Auxiliary Building

As addressed in Ginna UFSAR section 3.4.2 and correspondence with the NRC (Reference: Letter from J. E. Maier, Rochester Gas & Electric Corp. to D. M. Crutchfield, NRC, "SEP Topic III-6, Seismic Qualification of Tanks," June 16, 1983), the three chemical & volume control system holdup tanks and the waste holdup tank were qualified to Seismic Category I requirements. Seismic qualification was performed in order to ensure that the amount of flooding in the auxiliary building which would result from failure of non-seismic tanks would not adversely affect required safety-related equipment in the lower levels of the building. This issue was addressed in SEP Topic IX-3. The maximum water level from the failure of the non-qualified tanks is 8 inches, which is below the elevation of the bottom of the safety injection system pump motor of 20 inches. With the seismic qualification of the three chemical & volume control system holdup tanks, the NRC determined that the issue of internal flooding due to potential failure of tanks was adequately resolved (Reference: Letter from D. M. Crutchfield, NRC to J. E. Maier, RG&E, "Integrated Plant Safety Assessment Report (IPSAR), Section 4.25.3, Flooding Due to Failure of Tanks," July 8, 1983).

Ginna UFSAR Section 3.4.2 also states that failure of the demineralization system in the Auxiliary Building would result in a maximum water level increase of 0.2 inches. Therefore, the maximum water level from failure of the non-qualified tanks and the demineralization system is 8.2 inches. However, this level is below the bottom elevation of the SI pump motor, identified above.

Auxiliary Building Annex

The auxiliary building annex houses the standby auxiliary feedwater system. As addressed in Ginna UFSAR section 3.11.3.8, flooding is not a concern in this structure, since all safety-related equipment associated with the standby auxiliary feedwater system is elevated, so that a complete failure of the condensate test tank located in the structure would not cause submergence of this equipment.

Internal Flooding Due to Failure of Systems / Components Other Than Tanks

Auxiliary Building

As addressed in Ginna UFSAR section 3.11.3.2.4, a review of potential equipment failures was conducted as part of the Appendix R fire protection review, as well as Systematic Evaluation Program Topic III-6, Seismic Design Considerations. It was determined that actuation of the fire protection sprinklers would not flood required safety-related equipment in the auxiliary building.

Diesel Generator Rooms

As addressed in Ginna UFSAR section 3.11.3.6, installation of the "superwall" at the diesel generator annex / turbine building interface provides protection against flooding of the diesel generator rooms from a circulating water system line break.

2.5.1.1.1.2.2 Acceptance Criteria

Ginna Station's current licensing basis addresses design of systems and components important to safety to withstand the effects of natural phenomena (GDC-2). Review guidance in Standard Review Plan Section 3.4.1 addresses (1) determination if liquid-carrying systems could produce flooding, and an evaluation of measures taken to protect safety-related equipment and (2) review of the effects of potential flooding of systems and components due to postulated failure of non-seismic Category I and non-tornado protected tanks, vessels, and other process equipment.

2.5.1.1.1.2.3 Description of Analyses and Evaluations

Evaluation of the impact of the EPU on protection from internal flooding outside containment is as follows:

Internal Flooding Due to Failure of Tanks

For the EPU, there are no new tanks and no changes in the size of existing non-seismic tanks located in the auxiliary building or in the amount of fluid in these tanks that could lead to flooding due to failure of the tanks. The EPU does not affect the demineralization system in the auxiliary building. The EPU does not affect the volume of water in the condensate test tank located in the auxiliary building annex. Therefore, the EPU does not affect the results of existing analyses of flooding due to failure of tanks and process equipment in the auxiliary building and auxiliary building annex.

Internal Flooding Due to Failure of Components Other Than Tanks

The EPU does not affect the fire suppression systems in the auxiliary building, and therefore does not affect the conclusion that actuation of the fire protection sprinklers will not flood required safety-related equipment in the auxiliary building.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

In addition to the evaluations described in the Ginna UFSAR, the barriers and equipment used to mitigate floods was evaluated for the Ginna License Renewal. The evaluations are documented in the License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Section 2.4.2 of the License Renewal SER addresses aging management review of flood barriers in plant structures. As addressed the Licenser Renewal SER, sections 2.3.4.1 and 2.3.4.2, these barriers are evaluated within the structure that contains them. Since the EPU does not add any new structures/ components used to resist the effects of flooding, it does not affect the evaluation of these structures in the SER. Aging management of these components is addressed in SER section 3.5.

2.5.1.1.1.3 Results

Previous evaluations have demonstrated that the failure of non-seismic tanks and the demineralizer system will not result in internal flooding that would adversely affect safety-related equipment in the auxiliary building. Internal flooding in the auxiliary building annex is not a concern because safety-related equipment is elevated higher that any anticiapated flood level. The EDG rooms are protected against internal flooding consequences of of circulating water system pipe break.

EPU does not add any new non-seismic tanks or increase the capacity of existing non-seismic tanks. EPU does not affect the previously analyzed fire suppression system with respect to internal flooding. Therefore, the previously analyzed and accepted internal flooding considerations remain valid for operation at EPU conditions.

2.5.1.1.1.4 Conclusion

Internal flooding from sources other than high energy line breaks and moderate energy line cracks, including flooding from tanks and vessels which could potentially affect safety-related components, has been reviewed by the Ginna staff for impacts of the proposed EPU. The Ginna staff concludes that safety-related structures, systems, and components will continue to be protected from flooding and will continue to meet the Ginna Station current licensing basis with respect to the requirements of General Design Criterion 2 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

2.5.1.1.2.1 Regulatory Evaluation

The equipment and floor drainage system ensures that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The equipment and floor drainage system is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The Ginna Nuclear Power Plant, LLC's (Ginna) review of the equipment and floor drainage system included the collection and disposal of liquid effluents outside containment. The review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. The NRC's acceptance criteria for the equipment and floor drainage system are based on General Design Criterion 2 and General Design Criterion 4 insofar as they require the equipment and floor drainage system to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). Specific review criteria are contained in Standard Review Plan Section 9.3.3.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to:

- General Design Criterion 2 is addressed in Ginna UFSAR Section 3.1.2.1.2, "General Design Criterion 2 – Design Bases for Protection Against Natural Phenomena."

- General Design Criterion 4 is addressed in Ginna UFSAR Section 3.1.2.1.4, "General Design Criterion 4 – Environmental and Missile Design Bases."

Functions and features of the equipment and floor drains systems regarding internal flooding are addressed in the following Ginna UFSAR sections and correspondence with the NRC:

- UFSAR section 9.3.3, Equipment and Floor Drains
- UFSAR section 9.5.1.2.4.5, Floor Drains and Curbs
- UFSAR section 11.2.2, Liquid Waste Management System Description
- Letter from D. M. Cutchfield, NRC to J. E. Maier, RG&E, "SEP Topic III-5.B, Pipe Break Outside Containment," September 4, 1981.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

During plant license renewal evaluations, tanks and pipes which were not already in-scope pursuant to 10CFR54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (criterion (a)(2)). Components which met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

2.5.1.1.2.2 Technical Evaluation

2.5.1.1.2.2.1 Introduction

This section addresses functions of the equipment and floor drains systems, including routing and control of leakage, and prevention of backflow of water / contaminated fluids to areas of the plant containing safety-related equipment. Flooding in the intermediate building caused by a high energy line break (feedwater line rupture), including discussion of impact on drainage for EPU conditions, is addressed in LR section 2.5.1.3, "Pipe Failures."

As addressed in Ginna UFSAR section 9.3.3, the equipment and floor drain systems serve to route leakage from equipment and compartments in order to provide proper control of leakage, prevent uncontrolled communication between areas as necessary,

and allow monitoring of leakage prior to disposition. The equipment and floor drains are included in the liquid waste disposal system.

As addressed in Ginna UFSAR section 9.5.1.2.4.5, safety-related equipment is mounted on pedestals and floor drains provided in these areas are adequate to carry off fire-water and prevent safety-related equipment from being flooded with standing water. In areas such as the control room, where floor drains are not provided, fire water will be drained out through door openings. Curbs are provided in the screen house to prevent water or flammable liquid from flowing into the basement, where both divisions of safety-related cables are routed. Additional curbs are provided around the diesel-driven fire pump area in the screen house.

As documented in the NRC evaluation of SEP Topic III-5.B (Reference: Letter from D. M. Cutchfield, NRC to J. E. Maier, RG&E, "SEP Topic III-5.B, Pipe Break Outside Containment," September 4, 1981), drainage via floor drains in several plant areas assists in the prevention of flooding:

- Flooding on the auxiliary building operating floor due to a line break in the service water system, component cooling water system, or fire water system is prevented by sufficient drainage through building drains and open stairways.

As discussed in Ginna UFSAR section 9.5.1.2.4.5, where drains from safety-related areas are tied into drains from areas which contain a large quantity of flammable liquid, backflow protection is provided to prevent possible spread of a liquid fire via the drain system. As addressed in Ginna UFSAR section 9.3.3, the installed backflow prevention devices also serve to prevent internal flooding from affecting the following areas:

- Air handling room (control building)
- Battery rooms 1A and 1B
- Diesel generator rooms 1A and 1B

2.5.1.1.2.2.2 Acceptance Criteria

Ginna Station's licensing basis addressed design of safety-related systems and components to withstand the effects of natural phenomena (GDC-2). Guidance in Standard Review Plan Section 9.3.3 addresses inundation of safety-related areas due to drain backflow.

Ginna Station's licensing basis addresses design of systems to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (GDC-4). Guidance in Standard Review Plan 9.3.3 addresses acceptability of the system based on design to prevent flooding which could result in adverse affects on essential systems or components.

2.5.1.1.2.2.3 Description of Analyses and Evaluations

Liquids leaking from process systems and liquids from maintenance activities enter the equipment and floor drains system during all plant operating modes. The only pumps that will be modified to provide increased capacity in support of the EPU are the main feedwater pumps and the condensate booster pumps. Although the impellers for these pumps will be changed for the EPU, the pump shaft seals will not be changed, and the existing seals will still be within their design parameters. Accordingly, the EPU does not impact the quantities of liquids that enter the equipment and floor drains systems from these sources.

As addressed in LR section 2.5.1.1.1.2, the EPU does not affect size or volume of fluid in the non-seismic tanks in plant areas where flooding from these tanks could affect safety-related components. Therefore, there is no additional leakage from these sources which would affect the equipment and floor drains systems.

The EPU does not affect the operating flow rates and pressures of the service water system, component cooling water system, fire water system, or residual heat removal system. Therefore, the EPU does not affect the capability of the floor drains systems to assist in prevention of flooding due to line breaks in these systems in applicable areas.

The function of the installed backflow prevention devices to prevent flooding of safety-related areas via backflow through floor drains is not affected by the EPU. No new areas requiring backflow prevention are required at EPU conditions.

As addressed in LR section 2.5.1.1.3.2, in analyzing the water buildup in the turbine building due to a break or leakage in the circulating water system, the conservative assumption is made that no water escapes from the building via floor drains. Drainage of water from the building via the floor drains system would assist in lowering the height of water in the building due to circulating water system flooding.

As noted above, flooding in the intermediate building caused by a high energy line break (feedwater line rupture), including discussion of impact on drainage for EPU conditions, is addressed in LR section 2.5.1.3, "Pipe Failures."

Refer to LR section 2.5.6.2 for a discussion of the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

In addition to the evaluations described in the Ginna UFSAR, the floor drains and equipment used to mitigate floods was evaluated for the Ginna License Renewal. The evaluations are documented in the License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Section 2.4.2 of the License Renewal SER addresses aging management review of flood barriers in plant structures. As addressed the Licenser Renewal SER, sections 2.3.4.1 and 2.3.4.2, these barriers are evaluated within the structure that contains them. Floor drains are evaluated in SER section 2.3.3.12.1, Treated water. Since the EPU does not add any new structures/ components used to resist the effect of flooding, it does not affect the evaluation of these structures in the SER. Aging management of these components is addressed in SER section 3.5.

2.5.1.1.2.3 Results

The EPU does not impact the quantities of liquids that enter the equipment and floor drains systems from maintenance activities.

The EPU does not affect size or volume of fluid in the nonseismic tanks in plant areas where flooding from these tanks could affect safety-related components.

The EPU does not affect the operating flow rates and pressures of the service water system, component cooling water system, fire water system, or residual heat removal system.

No new areas requiring backflow prevention are required at EPU conditions.

The potential water buildup in the turbine building due to a break or leakage in the circulating water system is unaffected by the EPU.

As discussed in LR section 2.5.1.3, the water level in the basement of the intermediate building due to flooding is unaffected by the EPU. Since alternate shutdown methods are available in the event of high/ moderate energy pipe failures in the screen house, the limiting environment in the screen house continues to be normal ambient conditions. The high energy and moderate energy systems / lines in the auxiliary building are not impacted by the EPU. Environmental conditions in the control building are not affected by pipe failures in the turbine building. As discussed in LR section 2.5.1.1.3, the analyses and design features related to internal flooding in the turbine building due to leakage or a break in the circulating water system are unaffected by the EPU.

2.5.1.1.2.4 Conclusion

The Ginna staff has reviewed the effects of the proposed EPU on the equipment and floor drainage system and concludes that the assessment has adequately accounted for the plant changes resulting in larger capacity pumps The Ginna staff concludes that the equipment and floor drain system has sufficient capacity to (1) handle the expected leakage resulting from plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to non-contaminated drainage systems. Based on this, the Ginna staff concludes that the equipment and floor drainage system will continue to meet the Ginna Station current licensing basis with respect to the requirements of General Design Criterion 2 and

General Design Criterion 4 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the equipment and floor drainage system.

2.5.1.1.3 Circulating Water System

2.5.1.1.3.1 Regulatory Evaluation

The circulating water system provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The review of the circulating water system focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed EPU. The NRC's acceptance criteria for the circulating water system are based on General Design Criterion 4 for the effects of flooding of safety-related areas due to leakage from the circulating water system and the effects of malfunction or failure of a component or piping of the circulating water system on the functional performance capabilities of safety-related structures, systems, and components. Specific review criteria are contained in Standard Review Plan Section 10.4.5.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to General Design Criterion 4 is addressed in Ginna UFSAR Section 3.1.2.1.4, "General Design Criterion 4 – Environmental and Missile Design Bases."

Analyses / design features related to internal flooding due to leakage or a break in the circulating water system are addressed in the following Ginna UFSAR sections and correspondence with the NRC:

- UFSAR section 3.6.2.4.8.2, Screen House and Turbine Building Flooding
- UFSAR section 10.6.2.9, Flooding Protection
- Letter from K. W. Amish, RG&E, to D. J. Skovholt, AEC, "Flooding of Critical Equipment," May 31, 1973

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

During plant license renewal evaluations, tanks and pipes which were not already in-scope pursuant to criterion 10CFR54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (criterion (a)(2)). Components which met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

2.5.1.1.3.2 Technical Evaluation

2.5.1.1.3.2.1 Introduction

Protection of safety-related equipment from flooding due to a break or leakage in the circulating water system, is discussed in Ginna UFSAR Sections 3.6.2.4.8.2 and 10.6.2.9, and in a letter to the NRC (Reference: Letter from K. W. Amish, RG&E, to D. J. Skovholt, AEC, "Flooding of Critical Equipment," May 31, 1973). This protection consists of (1) tripping the circulating water pumps by level switches installed in the circulating water pump pit in the screen house and in the condenser pit in the turbine building, (2) a permanently installed, non-movable Seismic Category I dike in the screen house, and (3) elevated doorways between the turbine building and the control building, which have been built to contain the water that may escape from the circulating water system.

The level switch circuitry for tripping the circulating water pumps consists of float switches with redundant two-out-of-three logic receiving level information from the circulating water pump pit in the screen house and in the condenser pit in the turbine building.

The maximum height of water in the turbine building due to circulating water system flooding has been determined to be 8.92 inches. Protection of the control building due to potential flooding in the turbine building is provided by doorways to the control building being elevated 18 inches above the turbine building floor.

In the screen house, based on the conservative assumption that the two circulating water pumps do not trip, the maximum water level that could occur would be 10.8 inches. The dikes in the screen house are 30 inches high, and are situated to prevent water from reaching safety-related equipment.

In analyzing the water buildup in the turbine building, the conservative assumption was made that no water escapes from the turbine building through open doors or floor drains. If this water were to escape from the floor, but be restricted from draining into the circulating water discharge canal, the water level would reach a steady-state height of 4 inches around the screen house. Wave action due to a Safe Shutdown Earthquake would add 4 inches, for a total of 8 inches. In order to

prevent this water from draining into the basement of the screen house, 9-inch curbs were installed around entrances to this area.

2.5.1.1.3.2.2 Acceptance Criteria

Ginna Station's licensing basis addresses design of systems to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (GDC-4). As addressed in Standard Review Plan Section 10.4.5, compliance with GDC-4 is based on meeting the following:

- Providing means to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system / component will not be precluded due to leakage from the circulating water system.

Ensuring that malfunction of a component or piping of the circulating water system will not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.

2.5.1.1.3.2.3 Description of Analyses and Evaluations

Evaluation of the impact of the EPU on analyses and design features related to internal flooding due to leakage or a break in the circulating water system is as follows: As discussed in LR section 2.5.8.1, the circulating water system flow rate and operating pressures do not change at EPU conditions. There are no modifications to the circulating water system resulting from the EPU.

Therefore, for the turbine building, control building, and the screen house, the analyses and design features related to internal flooding due to leakage or a break in the circulating water system for current plant conditions are unaffected by the EPU; protection of safety-related equipment continues to be provided.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

In addition to the evaluations described in the Ginna UFSAR, the barriers and equipment used to mitigate floods was evaluated for the Ginna License Renewal. The evaluations are documented in the License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Section 2.4.2 of the License Renewal SER addresses aging management review of flood barriers in plant structures. As addressed the Licenser Renewal SER, sections 2.3.4.1 and 2.3.4.2, these barriers are evaluated within the structure that contains them. Since the EPU does not add any new structures/ components used to resist the effects of flooding, it does not affect the evaluation of these structures in the SER. Aging management of these components is addressed in SER section 3.5.

2.5.1.1.3.3 Results

As discussed in LR section 2.5.8.1, the circulating water system flow rate and operating pressures do not change at EPU conditions. There are no modifications to the circulating water system resulting from the EPU. The analyses and design features related to internal flooding due to leakage or a break in the circulating water system for current plant conditions are unaffected by the EPU.

2.5.1.1.3.4 Conclusion

The Ginna staff has reviewed the protection of safety-related equipment from flooding due to a break or leakage in the circulating water system. Affected areas reviewed include the turbine building, control building, and the screen house. The Ginna staff concludes that, consistent with the Ginna Station will continue to meet the current licensing basis with respect to the requirements of General Design Criterion 4, since the circulating water system flow and operating pressures will remain unchanged for the proposed EPU, and there are no modifications to the circulating water system resulting from the proposed EPU. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to the flooding from the circulating water system.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

2.5.1.2.1.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff's review concerns missiles that could result from in-plant component overspeed failures and high pressure system ruptures. The Ginna staff's review of potential missile sources covered pressurized components and systems, and high-speed rotating equipment. The Ginna review was conducted to ensure that safety-related systems, structures, and components (SSC's) are adequately protected from internally generated missiles. In addition, for cases where safety-related SSC's are located in areas containing non-safety related SSC's, the Ginna staff reviewed the non-safety related SSC's to ensure that their failure will not preclude the intended safety function of the safety-related SSC's. The Ginna staff's review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected.

The NRC's acceptance criteria for the protection of safety-related SSC's against the effects of internally generated missiles that may result from equipment failures are based on

- GDC-4 insofar as SSC's important-to-safety are required to be protected against the effects of internally generated missiles that may result from equipment failures in order to maintain their essential safety functions.

Specific review criteria are contained in SRP sections 3.5.1.1 and 3.5.1.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR Section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station safety-related SSC's being adequately protected from internally generated missiles relative to conformance to:

- GDC – 4 is described in Ginna UFSAR Section 3.1.2.1.4, General Design Criterion 4 – Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review of postulated pipe breaks both inside and outside containment, including the dynamic effects of such pipe breaks, as part of the Systematic Evaluation Program (SEP) Topic III-4.C, Internally Generated Missiles. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant.

Conformance to the requirements of GDC-4 ensuring that safety-related SSC's are adequately protected from internally generated missiles is addressed in Ginna UFSAR Section 3.5.1.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's missile barrier components were evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage the aging effects are documented in:

- License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the equipment and components credited with mitigating the effect of missiles is described in section 2.4 and the programs credited with managing that equipment aging is described in sections 3.5.2.2.1 and 3.5.2.4.2.

2.5.1.2.1.2 Technical Evaluation

2.5.1.2.1.2.1 Introduction

Safety-related SSC's at Ginna Station have been evaluated to determine which need protection from internally generated missiles. These missiles are generated by failures in high energy systems and the overspeeding of rotating components. Ginna UFSAR section 3.5.1 discusses the measures taken to protect the safety related SSC's at Ginna Station against internally generated missiles.

2.5.1.2.1.2.2 Description of Analyses and Evaluations

Missiles which are generated internally to the reactor facility (inside or outside containment) may cause damage to SSC's that are necessary for the safe shutdown of the reactor or for accident mitigation or may cause damage to the SSC's whose failure could result in a significant release of radioactivity. The potential sources of such missiles are valve bonnets and hardware retaining bolts, relief valve parts, instrument wells, pressure containing equipment (such as accumulators and high pressure bottles), high speed rotating machinery, and rotating components such as impellers and fan blades.

Ginna UFSAR section 3.5.1 provides a detailed discussion of the measures taken to protect safety related SSC's against internally generated missiles.

The Ginna EPU review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. EPU does not change the characteristics of the previously evaluated potential missile sources nor, with one exception, add new potential high energy missile sources. . Changing the Feedwater Isolation Valves to automatic actuation introduces new high pressure air accumulators (as discussed in the license amendment request dated April 29, 2005). However, the effects of postulated failure of a new air accumulator are bounded by the effects of previously evaluated block wall failures. Notwithstanding the acceptability of postulated failure of an air accumulator, the modification design will locate and orient them for the least effect.

The Ginna EPU does not adversely impact the system pressures for the systems that could generate missiles, based on the evaluations contained in the specific licensing report system evaluation sections. As such, the existing missile protection measures remain adequate for EPU conditions. For plant areas containing safety-related SSC's, the proposed EPU will not result in any changes to existing missile sources or add any new components that could become a new potential missile source. The planned impeller modifications to the feedwater and condensate booster pumps (Refer to LR section 2.5.5.4, "Condensate and Feedwater") will not result in any missile concerns, as these pumps are not located near any safety related SSC's. The proposed EPU will also not result in any system configuration changes that would impact any existing missile barrier considerations.

Refer to LR section 2.5.1.2.2, "Turbine Missiles," for evaluations of the impact of turbine missiles.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's missile barrier components were evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage the aging effects are documented in License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the equipment and components credited with mitigating the effect of missiles are described in section 2.4 and the programs credited with managing that equipments aging are described in section 3.5.

2.5.1.2.1.3 Results

Since the Ginna EPU does not adversely impact the pressures for the systems that could generate missiles, the existing missile protection measures remain adequate for the proposed EPU conditions. For plant areas containing safety-related SSC's, the proposed EPU will not result in any changes to existing missile sources or add any new components that could become a new potential missile source. The proposed EPU will also not result in any system configuration changes that would impact any existing missile barrier considerations.

The results of the evaluations demonstrate that the Ginna EPU will not adversely impact safety-related SSC's with respect to internally generated missile concerns and will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4.

The Ginna EPU does not add new missile barrier components or modify any existing components that would change the license renewal evaluation boundaries. Therefore, no new aging effects requiring management are identified.

2.5.1.2.1.4 Conclusion

The Ginna staff reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that safety-related SSC's will continue to be protected from internally generated missiles and will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to internally generated missiles.

2.5.1.2.2 Turbine Generator

2.5.1.2.2.1 Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and the extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The Ginna Nuclear Power Plant, LLC (Ginna) staff's review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely.

The NRC's acceptance criteria for the turbine generator are based on:

- GDC-4, insofar as it relates to the protection of SCCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles.

Specific review criteria are contained in NRC Standard Review Plan (SRP) section 10.2.

Ginna Current Licensing Basis

As noted in Ginna USFAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in USFAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear plants, in order to reconfirm and document their safety. The results of the SEP review of Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of the Ginna Station turbine overspeed protection features relative to conformance to:

- GDC-4 is described in Ginna UFSAR, Section 3.1.2.1.4, "GDC-4, Environmental and Missile Design Bases". As described in this section of the UFSAR, Ginna Station received post-construction review as part of the SEP (SEP Topic III-4.B). The results of this review are documented in NUREG-0821. Conformance to the requirements of GDC-4, with respect to overspeed protection, is also described in UFSAR Section 3.5.1.2, "Turbine Missiles".

In addition to the evaluations described in the UFSAR, Ginna Station systems were evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, the Turbine Generator is not within the scope of License Renewal. However, the programs used to manage the aging effect associated with the steam and power conversion systems are discussed in section 3.4 for the above SER.

2.5.1.2.2.2 Technical Evaluation

2.5.1.2.2.2.1 Introduction

The Turbine Generator and Control System is described in Ginna UFSAR Section 10.2.

The main turbine train is made up of one high-pressure turbine and two low pressure turbines, all mounted on a common shaft. The steam flow path is first through the high-pressure turbine, through the moisture separators, then in a parallel path through the two low-pressure turbines. The main turbine operates at a design speed of 1800 rpm. High-pressure steam is admitted to the high-pressure turbine through two 24" lines. Each line has one stop valve and two control valves. These valves are controlled by the electro-hydraulic control (EHC) system. The stop valves are designed to shut off the steam flow to the turbine in the event the unit overspeeds beyond the setting of the overspeed trip. Following the steam flow through the moisture separator reheaters, steam enters the low-pressure turbines through four reheater stop valve and four reheater intercept valves. These valves are also controlled by the EHC system. The purpose of the reheater stop and intercept valves is to control the steam flow to the low-pressure turbines in the event of an overspeed condition. Installed in each extraction steam line from the high-pressure turbine are two counter-weighted air-operated non-return valves designed to prevent overspeeding of the turbine due to backflow of steam from the feedwater heaters following a turbine trip.

The overspeed protection system for the main turbine includes a mechanical overspeed trip mechanism. This device is an eccentric weight mounted on the turbine shaft rotor extension shaft. It is designed to trip the main turbine unit to ensure the turbine speed remains less than 120% of the design speed (2160 rpm). There is also an Overspeed Protection Controller incorporated into the EHC system. This includes a load drop anticipator and an auxiliary governor function. The load drop anticipator logic will rapidly close all control and intercept valves on a complete loss of load, and rapidly close the intercept valves on a partial loss of load. If the auxiliary governor senses an overspeed condition at 103%, the system will close the reheater intercept valves and modulate close the control valves until the overspeed condition clears.

The HP Turbine steam path and rotor are being replaced utilizing the existing HP turbine casing. Included in the replacement is elimination of the control nozzle block and converting the turbine control valve operation from a partial arc control system (sequential valve operation during turbine loading) to a full arc (4 valve operation during turbine loading) to increase the full load turbine efficiency. With respect to controlling turbine overspeed, changing to the full arc turbine control scheme does not impact the operation of the auxiliary governor or the mechanical overspeed trip functions.

Instrumentation changes in the EHC control cabinet for the control valve program, valve feedback circuits, and the load drop anticipator circuit inputs are described in LR section 2.4.1, Reactor Protection, Safety Features Actuation. Other than minor piping to accommodate the control valve modifications identified below, there are no physical changes required to the EHC system to support EPU.

2.5.1.2.2.2.2 Description of Analyses and Evaluations

In the event of a loss of electrical load on the turbine generator unit, the restraining torque on the turbine rotor unit is lost. The turbine control system is designed to close the inlet and extraction valves of both the high pressure turbine and the low pressure turbines. However, the steam energy entrapped in the turbine unit will cause the rotor to accelerate, potentially causing an overspeed condition.

The EPU increases the unit maximum power and the amount of entrapped energy. This results in an increase in expected peak overspeed.

The analysis was performed to demonstrate that the increase in power and entrapped steam energy at EPU conditions will not cause the turbine rotor to overspeed beyond the current design limit.

The following considerations were applied to this overspeed analysis:

- The normal operating turbine generator rotor "running" speed of 1800 rpm will not change as the result of EPU.
- The existing overspeed trip setpoint of 110% was assumed.
- The Turbine design overspeed limit of 120 % of design speed will not change as the result of EPU.
- The electrical separation of the turbine generator from the grid was assumed to be at full EPU load..
- The Ginna EPU only involves the replacement of the high pressure turbine. The existing low pressure turbines are to remain. The high pressure turbine rotor weight will increase. Therefore, the corresponding increase in inertia will tend to mitigate the EPU effects of increased power and energy. The increase in inertia was considered in the overspeed

analysis. However, the weight of the high pressure turbine rotor represents only a small contribution to total turbine generator weight. Therefore, the total increase in inertia due to the high pressure turbine replacement had only a minor impact upon the overspeed analysis results.

- In addition to the steam volumes of the turbine block components (valves, cylinders and cross under/cross over piping) the contribution of the steam and water volumes associated with the extraction lines and feedwater heaters were included in the overspeed analysis.

- The Ginna EPU will incorporate a modification to the four (4) high pressure control valves. Essentially, the control valves will be converted to a high lift design. The modifications will result in reduced valve velocities and corresponding decrease in pressure drop. This modification will increase the valve stroke significantly. The overspeed analysis is based upon the closure time of the slowest valve in the system. For Ginna this represents the high pressure turbine stop valves at a closure time of 300 ms maximum. The modified control valves are specified to have a maximum closure time of 200 ms.

Presently, the turbine mechanical overspeed trip allowable setpoint is less than 110% of rated speed (1980 rpm). The evaluation at EPU conditions determined that the allowable overspeed setpoint needs to be reduced to less than 109.3% of rated design speed (1967.4 rpm). Results from the overspeed tests performed between 1997 through 2005 indicates the current average setting to be 108.81% +/- 0.2%. Since the current setting is less than the new allowable setpoint, the current mechanical overspeed setting is acceptable for EPU. The reduced mechanical overspeed setpoint of 109.3% will ensure that the design overspeed of 120% will not be exceeded.

The only modification associated with the turbine inlet valves involves converting the four (4) high-pressure control valves to a high lift (increase stroke) design. Each high pressure turbine steam inlet path will still contain two (2) valves (stop valve and control valve) in series. Each high-pressure extraction line will still contain two (2) non-return valves in series. The overspeed protection system will still include a mechanical overspeed trip mechanism and auxiliary governor controller associated with the EHC system. Therefore, the EPU will not impact the redundancy aspects of the turbine overspeed control system.

The operability and reliability of the turbine overspeed protection system is verified via the performance of routine turbine control valve testing. At each turbine overhaul and each refueling outage, the turbine speed is increased to the overspeed trip setpoint during the power descent to verify closure of the stop, control, reheater stop and reheater intercept valves. When the turbine unit is brought up to speed the stop, control, reheater stop and reheater intercept valves are tested as part of the plant start-up. On an annual basis the stop, control, reheater stop and reheater interceptor valves are stroked tested. Neither the test methodology and nor test intervals of the turbine valves

will be modified under the implementation of the EPU. Therefore, the continued operability and reliability provided via valve performance testing at EPU conditions will be maintained.

The probability of turbine missile generation is analyzed via an evaluation of internal flaws (cracks) present in the low-pressure turbine discs and potential growth of the postulated flaws by a fatigue mechanism resulting from the speed cycling of the rotor. The purpose of these analyses is to determine an acceptable low-pressure turbine disc inspection interval, thereby limiting the postulated flaw growth below the established failure threshold based upon a critical flaw size.

The following considerations were applied to the turbine disc inspection interval analysis:

- The analysis is limited to the evaluation of the low-pressure turbine discs. The high-pressure turbine rotor is not considered in the disc inspection interval analysis. This is consistent with the original turbine disc integrity analysis performed. The basis for this position is that the high-pressure turbine is considered to be of an "integral" construction (single monoblock forging with axial or tangential entry blades). In addition, the material properties and the enhanced forging process associated with the replacement high-pressure rotor have provided an increase in the material toughness and therefore, a decrease in the likelihood of crack generation and growth.

- The low-pressure turbine discs are not being modified as part of the EPU implementation.

- The normal operating turbine generator rotor "running" speed of 1800 rpm will not change as the result of EPU. The design overspeed of 120% will not change as the result of EPU.

- The methodology of the low-pressure disc integrity analysis is consistent with the original methodology, previously accepted by NRC.

- The parameter that most directly impacts the inspection interval for the low-pressure turbine rotors is disc temperature. Slightly higher temperatures for some discs are expected under EPU conditions.

The low-pressure disc inspection interval decreased from 103.8 months at the original power rating to 99.7 months at EPU conditions. These inspection intervals are based upon the most limiting disc. Since the decrease in disc inspection interval still provides considerable margin above the current Ginna low-pressure turbine disc inspection interval of 78 months, the results are acceptable.

Refer to LR section 2.13.1, "Risk Evaluation of EPU," for the evaluation of the impact of EPU on the Probabilistic Risk Assessment (PRA) related to turbine missiles.

Evaluation of Impact of Renewed Plant Operating License Evaluations and License Renewal Programs

The Turbine Generator is not within the scope of License Renewal. However, the programs used to manage the aging effect associated with the steam and power conversion systems are discussed in License Renewal SER section 3.4. The changes associated with operating the turbine generator at EPU conditions do not add any new or previously unevaluated materials or components to the system. In addition, changes being made to the high pressure turbine and the high pressure control valves do not introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. The turbine generator overspeed control system will continue to operate within its current design limit. Therefore, no new aging effects requiring management are identified.

2.5.1.2.2.2.3 Turbine Generator Results

Continued compliance with the turbine generator overspeed protection requirements was demonstrated at the EPU conditions with no plant changes. The EPU overspeed analyses results are as follows:

- The allowable mechanical overspeed trip setpoint will be reduced from 110% to 109.3%. This will ensure that the turbine design overspeed limit of 120% will not be exceeded.
- Each turbine steam inlet path and high-pressure turbine extraction line will still be configured with two (2) isolation valves in series. Therefore, the redundancy aspects of the turbine overspeed control system will not be impacted.
- There will be no modifications to the performance testing methodology or inspection intervals of the turbine inlet valves. Therefore, there will be no impact upon the turbine overspeed protection system reliability associated with valve performance verification testing.
- The low-pressure disc inspection interval will decrease from 103.8 months to 99.7 months. Since the current inspection interval is 78 months, this decrease in calculated inspection interval is acceptable.

2.5.1.2.2.2.4 Conclusion

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the turbine generator and concludes that Ginna has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The Ginna staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-4 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

2.5.1.3.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff conducted a review of the plant design for protection from piping failures outside containment to ensure (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The Ginna staff review of pipe failures included high and moderate energy fluid system piping located outside containment. The review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of post accident operations where the consequences are not bounded by previous analysis.

The NRC's acceptance criteria for pipe failures are based on GDC-4, which requires, in part, that structures, systems and components (SSCs) important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids.

Specific review criteria are contained in Standard Review Plan 3.6.1.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station design relative to:

- General Design Criterion 4 is addressed in Ginna UFSAR section 3.1.2.1.4, "General Design Criterion 4 – Environmental and Missile Design Bases." As noted in Ginna UFSAR section 3.1.2.1.4, a review of postulated pipe breaks outside containment was conducted as part of the (SEP), including dynamic effects such as pipe whip and jet impingement.

Other Ginna UFSAR sections, which discuss pipe breaks outside containment, include sections 3.6.2 and 3.11.

Ginna UFSAR section 3.6.2 addresses the following topics:

- Postulated piping failures in fluid systems outside containment and provides the plant changes and measures to cope with piping failures as evaluated to the original NRC criteria for operating units.

- The Augmented Inservice Inspection Program. Another measure that was taken was the installation of the standby auxiliary feedwater system in a separate building not subject to the pipe failures. Plant modifications were made for installation of pipe whip restraints, jet impingement shields, relocation of certain piping and electrical equipment, and sealing certain wall penetrations.

- Summary of the results of the SEP review and identification of the plant measures to resolve SEP issues.

- The methodology for postulating break and crack locations, dynamic effects analysis methodology, blowdown analysis methodology, compartment pressurization analysis, and flooding analysis.

- The results of the pipe failures causing building pressurization. The limiting pipe failure for pressurization of the intermediate building is a break in a 6 inch main steam branch line. The pipe failures considered for pressurization of the turbine building are the 12 inch main steam turbine by-pass line and the 20 inch main feedwater supply line downstream of the number 5 feedwater heaters.

- Pipe failures causing flooding, and the features provided to protect safety-related equipment from flooding.

- Main steam safety and atmospheric relief valves that are subject to damage from cracks in feedwater piping in the intermediate building.

Ginna UFSAR section 3.11 provides supplemental information regarding the reviews and analyses performed for high energy and moderate energy pipe failures. This UFSAR section documents the environmental service conditions in plant buildings containing safety-related equipment, for both normal and accident conditions, including pipe failures.

In addition to the evaluations described in the Ginna UFSAR, the protective barriers and license renewal boundaries associated with high energy line breaks were evaluated for the Ginna Station License Renewal. The evaluations are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Section 2.4.2 of the License Renewal SER addresses aging management review of high energy line break barriers in plant structures. As addressed in the License Renewal SER, sections 2.3.4.1.2 and 2.3.4.2.1, the license renewal evaluation boundaries for the main steam and main feedwater systems were defined consistent with the high energy line break analysis as defined in the UFSAR. Within the SER, the structures and structural components used to resist the effects of pipe breaks are evaluated within the structure that contains them. Aging management of these components is described in the SER section 3.5.

2.5.1.3.2 Technical Evaluation

2.5.1.3.2.1 Introduction

As noted above, piping failures are discussed in Ginna UFSAR sections 3.1.2.1.4, 'General Design Criterion 4 – Environmental and Missile Design Bases;" 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping;" and 3.11, "Environmental Design of Mechanical and Electrical Equipment."

The high energy line break (HELB) analysis identifies high and moderate energy piping system lines subject to failure and the plant safety-related equipment potentially impacted by piping failures, determines the environmental effects resulting from the piping failures, and identifies the protection measures required to mitigate the effects of the piping failures. The environmental conditions resulting from this analysis are provided as input into the environmental qualification program. (Refer to LR section 2.3.1, "Equipment Qualification of Electrical Equipment." Refer also to LR section 2.2.1, "Pipe Rupture Locations and Associated Dynamic Effects," for the discussion of the impact of EPU on pipe break locations.)

The evaluation of pipe breaks outside containment considered the zones within the plant which contain systems required for safe shutdown and / or systems required to mitigate the effects of postulated pipe breaks.

2.5.1.3.2.2 Description of Analyses and Evaluations

2.5.1.3.2.2.1 High Energy / Moderate Energy Lines

The identification of the high energy and moderate energy lines does not change as a result of the EPU. The changes to system process conditions will not add or delete systems from the high energy or moderate energy category. The evaluations for EPU conditions do not create any new or revised pipe break locations from those identified in the UFSAR. Refer to LR section 2.2.1, "Pipe Rupture Locations and Associated Dynamic Effects." Because the high and moderate energy boundary definitions have not changed and no new equipment has been added that requires protection from the effects of pipe break, the existing high and moderate energy pipe break locations are not affected by EPU operating conditions.

The 30 inch and 36 inch main steam lines and 14 inch main feedwater lines in the intermediate building, and the 36 inch and 24 inch main steam lines and certain locations in the 20 inch main feedwater lines in the turbine building are excluded from the postulation of breaks because of the augmented in-service inspection program. The main steam and main feedwater system operating temperatures for EPU operating conditions are comparable to the existing temperatures considered in the design basis piping evaluations. The design pressures for the main steam and main feedwater systems do not change due to the EPU. Accordingly, the augmented in-service inspection program credited with ensuring the integrity of the large diameter pipes described above does not require boundary expansion as a result of EPU.

2.5.1.3.2.2.2 Pipe Whip and Jet Impingement

The design of jet impingement shields and pipe rupture restraint protection features are based on the pipe break dynamic effects at design pressures. The proposed EPU does not affect system design pressures. Therefore, the jet impingement shields and pipe restraints installed as protection for the effects of piping failures remain acceptable for the EPU.

2.5.1.3.2.2.3 Evaluation of Piping Failures

Main Steam System

The bounding high energy line break for the intermediate building pressure and temperature is a 6 inch branch line off the main steam line. The environmental conditions resulting from this break for current plant conditions are based on main steam operating pressure at full power. Although the full power operating pressure increases slightly for EPU, there is no adverse impact of this high energy line break at EPU conditions for the following reason:

- The only equipment required to operate in the environment created by this high energy line break is the reactor trip breakers which operate very early following the event. Equipment required to operate for mitigation of this event in the long term (e.g., standby auxiliary feedwater system) is located remote from the intermediate building.
- The peak intermediate building pressure for the existing operating condition is 0.25 psi as reported in UFSAR Table 3.11-1. This pressure is well below the 0.85 psi design limit of the limiting building structural component discussed in UFSAR Section 3.6.2.5.1.2

As discussed in UFSAR Section 3.11.3.5, the control building is protected against high energy line breaks which could occur outside of the control building envelope. For a 6" steam line break in the intermediate building, the east end of the cable tunnel was sealed to prevent leakage of steam into the control building rooms from creating a harsh environment. Since the EPU operating conditions do not significantly impact the

pressurization results associated with a 6" steam line break in the intermediate building, the control building environmental conditions are unaffected by the EPU.

The limiting main steam high energy line break in the turbine building is a 12 inch steam line break associated with the turbine by-pass branch lines off of the 36 inch main steam header. The larger 24 inch and 36 inch main steam lines are excluded from high energy line break considerations due to the augmented in-service inspection program performed on these lines. As discussed in UFSAR Section 3.6.2.5.1.4, the pressurization results from a 12 inch steam line break are bounded by the more severe pressurization results obtained from the most limiting main feedwater line break

Main Feedwater System

Cracks in the feedwater pipes were evaluated for jet impingement on the main steam safety valves and atmospheric relief valves. Jet impingement loads are based on the feedwater design pressure, which has not changed as a result of EPU. The EPU does not affect the protection measures currently in place for mitigation of these breaks.

Since augmented in-service inspection is performed on the 14 inch main feedwater piping in the intermediate building, the most limiting main feedwater break that could be postulated is from 1 inch branch lines. The resulting building pressurization results for this break size are bounded by the result from the more limiting 6 inch main steam branch line pipe break. The potential flooding consequences of a 1 inch main feedwater line break are accommodated by the design features of the intermediate building to protect against flooding. As discussed in UFSAR Section 3.6.2.5.1.2, sufficient drainage area is provided in the intermediate building to prevent any appreciable build-up of water in any floor of the intermediate building excluding the sub-basement level. Any water released as a result of a pipe break in the intermediate building would flow via open stairwells and hatch gratings to the sub-basement level which does not contain any equipment necessary for safe shutdown of the plant. Therefore, there are no flooding or environmental concerns associated with main feedwater piping in the intermediate building due to the EPU.

For the turbine building the most limiting high energy line break that needs to be postulated is in the 20 inch main feedwater piping. For the existing operating condition a postulated break in the 20 inch feedwater line downstream of the number 5 feedwater heaters in the turbine building is the worst case break for turbine building pressurization and environmental conditions for equipment. The results of this break envelope the results of a break in a 12 inch steam line in the turbine building. The impact of EPU process conditions on the turbine building pressurization was evaluated by comparing the mass/energy/steam addition rates under the pre-EPU and EPU conditions. For mass release rates, the modified Zaloudek critical flow correlation (for subcooled conditions) as well as the Moody critical flow correlation (for saturated conditions) were applied since the conditions upstream of the break are expected to change from subcooled to saturated conditions during blowdown. The maximum increase in mass release rate under EPU conditions is estimated as 3.26% for saturated conditions. The

maximum increase in enthalpy of the break effluent under EPU conditions is estimated as 1.8%. Therefore, the maximum increase in energy release rate (i.e., mass rate x enthalpy) under EPU conditions is estimated as 5.13%. Since the vapor (i.e., steam portion) of the break effluent was determined based on the total pressure of 14.7 psia (i.e., pressure flash option), the maximum increase in steam addition rate (i.e., mass rate x flashed fraction) under EPU conditions is estimated to be approximately 6.6%.

Based on the above results, the turbine building pressurization under EPU conditions is conservatively estimated by applying the multiplier of 1.066, which is based on the maximum steam addition rate increase, to the existing peak pressure loads. The peak pressure for the basement and mezzanine levels of the turbine building increases from the pre-EPU value of approximately 0.85 psig to an EPU value of 0.91 psig, which is below the maximum environmental design basis value of 1.14 psig. Similarly, the operating level pressure increases from 0.46 psig to 0.49 psig, which is below the design pressure of 0.7 psig. Therefore, EPU conditions are bounded by the pre-EPU environmental parameters in the turbine building of 220°F, 1.14 psig on the mezzanine and basement levels, and 0.7 psig on the operating floor. The design pressure for the walls protecting the control room is 0.7 psig and 1.14 psig for the relay room, battery room and diesel generator rooms. Therefore, EPU operation does not impact the design of these structures.

As is the case with the current analyses, the pressure resulting from a 20-inch main feedwater or 12-inch main steam line break in the turbine building is sufficient to cause failure of the turbine building / intermediate building concrete block walls. However, the EPU does not affect the protection measures currently in place for mitigation of these breaks as indicated in the Ginna UFSAR section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping."

Preferred Auxiliary Feedwater System

Process and design conditions for the preferred auxiliary feedwater system are unchanged by the EPU. (Refer to LR section 2.5.4.5, "Auxiliary Feedwater System.") Loss of this system due to a high energy line break is mitigated by use of the standby auxiliary feedwater system. This is not changed because of the EPU.

Service Water System

A service water line break in the sub-basement of the intermediate building or the mechanical equipment room of the control building creates a flooding condition. As addressed in LR section 2.5.4.2, "Service Water System," the EPU does not affect the flow rate of the service water system. Accordingly, the analyses and protection measures currently in place for mitigation of these breaks are not affected by the EPU.

A service water system line crack in the screen house could result in the loss of the service water system, including loss of service water cooling to the diesel generators. For this event, if the turbine-driven auxiliary feedwater pump is assumed to fail, the

steam generators can remove decay heat for approximately 35 minutes at EPU conditions before they boil dry. This time period would be used to align alternate cooling to a diesel generator per existing plant procedures, to start the diesel generator and to load a motor-driven auxiliary feedwater or standby auxiliary feedwater pump to the operating diesel generator.

A service water system line break in diesel generator rooms 1A or 1B could result in spray and/or flooding effects. As addressed in LR section 2.5.4.2, "Service Water System," the EPU does not affect the flow rate of the service water system. The EPU does not affect the protection measures currently in place for mitigation of this line break, e.g., use of redundant diesel generator.

Steam Generator Blowdown System

As addressed in LR section 2.1.10, "Steam Generator Blowdown System," the steam generator blowdown system design conditions are not affected by the EPU.

Chemical and Volume Control System Charging and Letdown Lines

As addressed in LR section 2.2.1, "Pipe Rupture Locations and Associated Dynamic Effects," the chemical and volume control system piping evaluations concluded that the EPU does not result in any new or revised break locations. The EPU does not affect the design or operating conditions in the charging and letdown lines. Accordingly, the EPU does not affect the protection measures currently in place for mitigation of breaks in these lines.

Plant Heating Steam

The EPU does not affect the process conditions in this system. The protection measures currently in place for mitigation of postulated breaks in this system are not changed by the EPU.

The effects of failure of a plant heating steam line in the screen house are similar to those of a service water system line crack in the screen house. Refer to the discussion under the service water system above.

Circulating Water System

A circulating water line break in the screen house or turbine building creates a flooding condition in these buildings. Refer to LR section 2.5.1.1.3, "Circulation Water System," for a discussion of this break, the protection measures for mitigation of the break, and impact of the EPU on these measures.

Fire Water System

The EPU does not affect the process conditions in this system.

The effects of failure of a fire water line in the screen house are similar to those of a service water system line crack in the screen house. Refer to the discussion under the service water system above.

The effects of failure of a fire water line in the diesel generator rooms are similar to those of a service water system line break in the diesel generator room. Refer to the discussion under the service water system above.

2.5.1.3.2.2.4 Summary of EPU Impact on Building Environments

Intermediate Building

The bounding high energy line break for the intermediate building pressure and temperature is a 6 inch branch line off the main steam line. However, there is no adverse impact of this high energy line break at EPU conditions. The only equipment required to operate is the reactor trip breakers whose operation occurs early in the event.

The water level in the basement of the building due to flooding is unaffected by the EPU. The existing design features of the intermediate building causes water to flow to the intermediate building sub-basement and prevent any appreciable accumulation of water at other floors of the building. The only equipment required to operate is the reactor trip breakers whose operation occurs early in the event prior to being impacted by the EPU conditions.

Since the EPU does not affect the flowrate of the service water system, the effects of a service water line break in the sub-basement of the intermediate building are not affected by the EPU.

Turbine Building

A postulated break in the 20-inch feedwater line downstream of the number 5 feedwater heaters is the worst case break for building pressurization and environmental conditions for equipment. The 20-inch feedwater break for EPU operation will result in a small increase the building pressures (i.e., ~0.02 to 0.04 psi), but the pressures remain bounded by the environmental parameters in the turbine building at current conditions.

Flooding in the turbine building due to a circulating water system line break is addressed in LR section 2.5.1.1.3, "Circulating Water System."

Auxiliary Building

The high energy and moderate energy systems / lines in the auxiliary building are not impacted by the EPU. The environmental conditions for current plant conditions apply for EPU conditions.

Screen House

Since alternate shutdown methods are available, no protection measures are needed for the service water system in the event of high / moderate energy pipe failures in the screen house. This is not changed as a result of the EPU. The limiting environment in the screen house continues to be normal ambient conditions.

Control Building (Control Room, Relay Room, Battery Rooms, Mechanical Equipment Room)

Environmental conditions in the control building are not affected by pipe failures in the turbine building. The pressure-shielding steel barrier wall between the turbine building and the control building protects the above-identified rooms from the turbine building environment. As indicated above, the EPU does not change the maximum environmental parameters in the turbine building used for the design of the walls. The sealing of the east end of the cable tunnel protects the control building from any harsh environmental conditions associated with breaks in the intermediate building.

Diesel Generator Building

Environmental conditions in the diesel generator building are not affected by pipe failures in the turbine building. The pressure-shielding steel barrier wall between the turbine building and the diesel generator building protects the diesel generator rooms from the turbine building environment. As indicated above, the EPU does not change the maximum environmental parameters in the turbine building.

Evaluation of Impact of Renewed Plant Operating License Evaluations and License Renewal Programs

In addition to the evaluations described in the Ginna UFSAR, the protective barriers associated with high energy line breaks were evaluated for the Ginna Station License Renewal. The evaluations are documented in License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Section 2.4.2 of the License Renewal SER addresses aging management review of high energy line break barriers in plant structures. As addressed in the License Renewal SER, sections 2.3.4.1 and 2.3.4.2, the license renewal evaluation boundaries for the main steam and main feedwater systems were defined consistent with the high energy line break analysis as defined in the UFSAR. Within the SER, the structures and structural components used to resist the effects of pipe breaks are evaluated within the

structure that contains them. Since the EPU does not add any new structures / components or require the modification of existing structures / components used to resist the effects of pipe breaks, it does not affect the evaluations of these structures in the SER. Aging management of these components is described in the SER section 3.5.

2.5.1.3.3 Results

The changes to system process conditions will not add or delete systems from the high energy or moderate energy category. The evaluations for EPU conditions do not create any new or revised pipe break locations from those identified in the UFSAR. The existing high and moderate energy pipe break locations are not affected by EPU operating conditions. The proposed EPU does not affect system design pressures. The jet impingement shields and pipe restraints installed as protection for the effects of piping failures remain acceptable for the EPU. Therefore, the current analyses remain valid for operation at EPU conditions.

2.5.1.3.4 Conclusions

The Ginna staff has reviewed the changes that are necessary for the proposed EPU and the proposed operation of the plant and concludes that safety-related SSCs will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the Ginna Station current licensing basis with respect to the requirements of General Design Criterion 4 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to protection against piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

2.5.1.4.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The Ginna Nuclear Power Plant, LLC (Ginna) staff review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire.

The NRC's acceptance criteria for the fire protection program are based on:

- 10CFR50.48 and associated Appendix R to 10CFR50, insofar as they require the development of a fire protection program to ensure, among other things, the capability to safely shut down the plant

- GDC-3, insofar as it requires that:

  a) Structures, systems, and components important-to-safety be designed and located to minimize the probability and effect of fires,

  b) Noncombustible and heat resistant materials be used, and

  c) Fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important-to-safety .

- GDC-5, insofar as it requires that SSCs important-to-safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Specific review criteria are contained in Standard Review Plan Section 9.5.1, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety

Assessment Report (IPSAR), completed August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station design relative to conformance to:

- GDC - 3 is described in Ginna UFSAR section 9.5.1, "Fire Protection Systems." This UFSAR section discusses conformance to GDC-3 of 10CFR50, Appendix A; Branch Technical Position (BTP) 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants;" Atomic Industrial Forum (AIF) GDC-3; 10CFR50.48, "Fire Protection;" and 10CFR50, Appendix R, "Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979." The discussion in UFSAR section 9.5.1 provides a consolidation of the information provided in UFSAR section 3.1.1.1.3 for AIF GDC-3 and UFSAR section 3.1.2.1.3 for NRC GDC-3 as presented in 10CFR50, Appendix A. As described in UFSAR section 9.5.1, fire detection and fighting systems of appropriate capacity and capability are provided to minimize the adverse effects of fire on safety-related SSCs . Fire prevention in all areas of the plant is provided by structure and component design, which optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Sensing devices include both ionization chambers (smoke detectors) and temperature detectors. Fire-fighting equipment includes fixed (automatic water suppression) in appropriate areas. In addition, automatically initiated Halon 1301 provides a total flooding system in the relay room and computer room. Appropriate hoses and portable fire-fighting equipment are placed throughout the plant. The fire protection system is designed in accordance with the standards of the National Fire Protection Association and is based generally on the recommendations of the Nuclear Energy Property Insurance Association.

- GDC – 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna is a single unit installation. Therefore, there are no shared SSCs.

As addressed in Ginna UFSAR section 9.5.1.1.1, the design criterion used during the licensing of Ginna Station was General Design Criterion 3, included in the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967. The design of the fire protection system was reviewed in 1972 on the basis of GDC 3 of Appendix A to 10CFR50, which was promulgated after the licensing of Ginna Station. It was determined that the requirements of GDC 3 were appropriately met by the plant design.

As addressed in Ginna UFSAR section 9.5.1.1.2, in May 1976 NRC Branch Technical Position (BTP) 9.5-1 was published for comment as Regulatory Guide 1.120, and in August 1976, Appendix A to BTP 9.5-1 was published for use by plants docketed prior to July 1, 1976. The design of the Ginna Station fire protection system was reviewed against the criteria of BTP 9.5-1 in a submittal to the NRC in February 1977. The submittal included a fire hazards analysis and several proposed design modifications in compliance with the regulatory guidance. A safety evaluation report was issued by the NRC in February 1979, with supplements in December 1980, February 1981, and June 1981.

As addressed in Ginna UFSAR section 9.5.1.4:

- The requirements for protecting safe shutdown systems and their respective components and associated circuits are specified in 10CFR50, Appendix R and NRC Generic Letter 81-12, "Fire Protection Rule." Ginna submitted an evaluation report in January 1984 describing alternative safe shutdown capability in accordance with Appendix R, section III.G. The report was revised in October 1984 and in January 1985. The revisions included a request for specific exemptions from the retrofit requirements of Appendix R, section III.G. In safety evaluation reports (SERs) of February 1985 and March 1985, the NRC accepted the alternative safe shutdown proposals and granted the requested exemptions. Thus, all areas of the plant either meet the retrofit requirements of Appendix R (as exempted), or are provided with acceptable alternative safe shutdown capability. In 1998, Ginna notified the NRC that one of the exemptions granted in the safety evaluation report of March 1985 was no longer required.

- Subsequent to the implementation of the Appendix R modifications during the 1986 refueling outage, the Ginna Station alternative safe shutdown report was revised (March 1986) to incorporate deviations from the original design and compliance methods. The updated safe shutdown report is included in the Ginna Station Fire Protection Program Report.

The Ginna Station Fire Protection Program Report summarizes the licensing basis for the program, as follows: The program has been developed to comply with and is based upon the requirements of GDC 3 of 10CFR50, Appendix A; 10CFR50.48(a); and Ginna's commitment to implement 10CFR50,Appendix R, sections III.G (Fire Protection of Safe Shutdown Capability), III.J (Emergency Lighting), and III.O (Reactor Coolant Pump Oil Collection System), and BTP 9.5-1, Appendix A. The requirements contained in 10CFR50, Appendix R, section III.L (Alternative and Dedicated Shutdown Capability) are applicable to areas where alternate shutdown capability is selected.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station Fire Protection Program was evaluated for plant license renewal. The evaluation is documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The fire protection program / systems are addressed in sections 2.3.3.6 and 3.3.2.3.2 of the SER. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them.

2.5.1.4.2 Technical Evaluation

2.5.1.4.2.1 Introduction

Fire Protection Program Report

The Ginna Station Fire Protection Program Report consolidates a detailed summary of the Ginna Station regulatory-required Fire Protection Program into a single document, and, as such, embodies the fire protection program. The Report documents Ginna's Fire Protection Plan, Fire Hazards Analysis, and Appendix R Safe Shutdown Analysis (herein referred to as the "Safe Shutdown Analysis"). It also includes a summary of Ginna's conformance to the BTP 9.5-1, Appendix A, a summary of the commitments made in demonstrating conformance to 10CFR50, Appendix R, a summary of Ginna's fire protection review and actions relative to industry operating experience, a review of compliance with selected NFPA codes, and a summary of fire protection engineering evaluations.

The program description / evaluation in this section addresses:

- "Fire Protection," which includes elements of the Fire Protection Program associated with the Fire Protection Plan and the Fire Hazards Analysis.

- "Safe Shutdown Analysis," which includes program elements associated with the Appendix R Safe Shutdown Analysis.

Fire Protection

The Fire Protection Plan describes the controls associated with the Ginna Station Fire Protection Program. The plan describes the features necessary to implement the Fire Protection Program, including administrative controls, personnel requirements for fire prevention and manual fire suppression activities, and fire protection systems and features, including fire detection and automatic and manually operated suppression systems. The administrative controls include controls to minimize the amounts of combustibles to which a safety-related / safe shutdown area may be exposed, control of hot work, impairment monitoring, etc.

The Fire Hazards Analysis includes evaluations of the fire areas / fire zones at Ginna, physical characteristics of required fire barriers, and combustible loading and fire severity in each fire area / fire zone.

Safe Shutdown Analysis

The Safe Shutdown Analysis identifies the analysis methodology used to demonstrate compliance with 10CFR50, Appendix R, sections III.G, III.J, III.O, and portions of III.L. Included is a description of the approved Appendix R exemptions for those plant areas where literal compliance with Appendix R, section III.G was not required.

The following Safe Shutdown Analysis topics are addressed below:

- Safe shutdown systems / components
- Alternative shutdown capability
- Appendix R compliance strategies
- Time critical tasks
- Other supporting analyses / evaluations
- Safe shutdown procedures

Safe Shutdown Systems / Components

The Safe Shutdown Analysis identifies the specific systems credited with achieving safe and/or alternative shutdown (e.g., auxiliary feedwater (AFW) system, residual heat removal (RHR) system). The safe shutdown components, which represent the minimum set of components used to achieve Appendix R safe / alternative shutdown (e.g., main steam atmospheric relief valves), are identified in a plant database.

Alternative Shutdown Capability

For the following Ginna fires areas, compliance with the provisions of 10CFR50, Appendix R, section III.G.2 cannot be achieved, and therefore an alternative shutdown capability is provided for the following plant areas:

- Control complex
- Cable tunnel
- Auxiliary building basement / mezzanine
- Battery rooms 1A / 1B
- Emergency diesel generator 1B area
- Screenhouse

The alternative shutdown methods provide the capability to achieve cold shutdown conditions within 72 hours and maintain them thereafter. The systems and equipment comprising the alternative shutdown methods are capable of being powered by either on-site or off-site electrical power sources.

Alternative shutdown methods include the following:

- Local control of one charging pump to provide makeup / increase in reactor coolant system pressure.
- Local control of turbine-driven AFW pump lube oil pump and turbine.
- Local control of one emergency diesel generator.
- Local connection between standby AFW system and underground yard fire water hydrant using fire hose to provide alternative AFW in event of loss of service water system.
- Local connection between underground yard fire water hydrant and emergency diesel generator using fire hose to provide alternative diesel generator cooling in event of loss of the service water system.

Appendix R Compliance Strategies

The Safe Shutdown Analysis provides a description of the safe shutdown methods for each plant fire area. Although other equipment may be used to bring the plant to a safe shutdown condition following a fire, the Safe Shutdown Analysis shows that at least one train of safe shutdown equipment is available to achieve safe shutdown given a fire in the defined fires areas at Ginna Station.

In the event of a fire in the auxiliary building basement / mezzanine, auxiliary building operating floor / intermediate level, or the screenhouse, the RHR system or its associated support systems may be damaged by fire. If the RHR system is rendered inoperable, water solid steam generator operation can be employed for cooldown to Cold Shutdown conditions. A design analysis shows that, for current plant conditions, the plant can be cooled from 260°F to less than 200°F in 12 hours using water solid steam generator operation.

In the event of a fire in the control complex, alternative shutdown capability is provided by local operation of the turbine-driven AFW pump. A time validation performed for the fire procedure for this area showed that the operators were controlling flow to the steam generators using the turbine-driven pump discharge control valve at 20 minutes after reactor shutdown. Analysis of steam generator dryout time for current plant conditions shows that it would take greater than 50 minutes for both steam generators to boil dry assuming no feedwater addition.

Time-Critical Tasks

The Safe Shutdown Analysis identifies three "time-critical" tasks. A description of these tasks and their basis follows:

- Closure of the RHR pump suction valve from the refueling water storage tank (RWST) after reactor shutdown for an Appendix R fire.

  A design analysis determines the time to drain inventory in the RWST for the scenario in which spurious operation of one of the RHR pump suction valves causes flow into containment. Closure of the RHR pump suction valve from the RWST within 28 minutes after reactor shutdown terminates the drainage of the RWST and assures that makeup to the RWST is not required for an Appendix R fire in which the pressurizer power operated relief valves (PORVs) are not available for primary system depressurization.

- Restoration of charging flow after reactor shutdown for an Appendix R fire.

  A design analysis determines the time for pressurizer level to decrease from the level at hot zero power (35%) to the bottom level tap, due to total reactor coolant system leakage, including reactor coolant pump leakage. Restoration of charging flow within 36 minutes after reactor shutdown assures that pressurizer level will not decrease below the level indicating range.

- Stopping charging flow after reactor shutdown for an Appendix R fire.

  A design analysis documents that the worst case for a loss of charging pumps suction would be a fire event which caused a spurious closure of the volume control tank suction air-operated valve, with the RWST suction air-operated valve remaining closed. The analysis determines that tripping the charging pumps from the control room within 1 minute after reactor shutdown will prevent any pump degradation due to loss of pump suction.

As addressed in the Safe Shutdown Analysis, as part of the development of the current fire procedures, these time critical tasks have been validated in the plant, utilizing plant operators.

Other Supporting Analyses / Evaluations

Supporting analyses / evaluations include the following:

- A design analysis documents the capability of the underground yard (city water) fire water loop to provide an alternate supply to the service water header. The limiting conditions in this analysis include a fire in the screen house which renders the motor and diesel driven fire pumps, along with the service water pumps, inoperable. The analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby auxiliary feedwater pumps, and two standby auxiliary feedwater room coolers.

- An effectiveness review of the reactor coolant pump lube oil collection system concludes that the system is adequately designed to contain the lube oil spills from the various leak sites located on the reactor coolant pump motors.

Safe Shutdown Procedures

Various procedures have been developed to guide the operators during alternative shutdown operations. These include operational and repair procedures to achieve cold shutdown. Operational procedures include alternative shutdown procedures for the fire areas requiring alternative shutdown capability, identified above. Operational procedures also address alternate water supplies to the auxiliary feedwater pumps and alternate cooling for the emergency diesel generators.

Acceptance Criteria

The overall acceptability of the Fire Protection Program at Ginna Station is based upon a defense-in-depth approach to demonstrate that in case of fire, the plant can be safely shutdown and maintained in a safe shutdown condition. The key attributes to this defense-in-depth approach are to provide

- Fire protection by structure and component design and location and by the application of appropriate administrative controls to minimize the probability and effects of fires,

- Fire detection capability by appropriate fire sensing devices, and

- Fire fighting and suppression capability through personnel training and provision of suitable portable and fixed fire fighting equipment of appropriate capacity and capability.

2.5.1.4.2.2 Description of Analyses and Evaluations

Fire Protection

Apart from plant modifications, the EPU does not affect the following elements of the fire protection program:

- Addition of new combustible material.
- Fire barriers, penetrations, doors, or the plant radio system.
- Ventilation air flow patterns.
- Plant fire programs or the Fire Protection Program Report.
- Fire wrap and fire coatings on structural steel.
- Fire protection suppression or fire detection system components.
- Safety-related components within an area protected by the fire suppression system.

Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Fire Protection Program requirements.

The EPU does not affect the elements of the fire protection program related to administrative controls and fire protection responsibilities of plant personnel. Evaluation of the impact of the EPU on the probability of increased radiological release resulting from a fire is addressed in LR section 2.13, Risk Evaluation.

Safe Shutdown Analysis

Safe Shutdown Systems / Components

The EPU does not affect the minimum set of pre EPU safe shutdown systems / components, including cables, credited with achieving safe and/or alternative shutdown. However, additional equipment is added to the list to account for the effects of increased decay heat. Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

Alternative Shutdown Capability

The EPU does not affect the alternative shutdown methods, discussed in LR section 2.5.1.4.2.1. The EPU does not modify the function of any mechanical component in the alternative safe shutdown flow paths or introduce any plant equipment failure modes which will impact the ability to achieve any of the alternative shutdown functions. The EPU does not adversely affect any components or circuits that provide power, control, or indication to components required for alternative safe shutdown. Enhancements will be made to selected components or control circuits to improve alternative shut down capability. As stated above, plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

Analyses were performed to demonstrate that the plant can be cooled down from normal operating temperature to Cold Shutdown at EPU conditions for the following cases: (1)

residual heat removal system (and its supporting systems) is available, and (2) RHR system is not available. For the case where the RHR system is not available, water solid steam generator operation is used for the final phase of cooldown to Cold Shutdown.

- Case 1: RHR system is available:
  - An analysis shows that the plant can be cooled from normal operating temperature to residual heat removal system initiation conditions with one steam generator within 60 hours after reactor shutdown.
  - An analysis shows that the plant can be cooled from RHR system initiation conditions to Cold Shutdown within 72 hours after reactor shutdown, assuming single train cooldown and a cooldown start time of 60 hours after reactor shutdown.
- Case 2: RHR system is not available:
  - An analysis provides time vs. temperature data for plant cooldown, using two steam generators, from normal operating temperature to conditions for initiation of water solid steam generator operation. For this analysis, at least one pressurizer PORV is available for depressurization of the primary system during the cooldown.

    For the three fire areas which may require water solid steam generator operation (Auxiliary Building basement / mezzanine, Auxiliary Building operating floor / intermediate level, and Screenhouse), in the Safe shutdown Analysis, only one train of equipment (e.g., one steam generator atmospheric relief valve) is currently credited for cooldown to the temperature at which water solid steam generator operation is initiated. Since two steam generators are needed in order to meet the Appendix R 72-hour cooldown requirement for this case at EPU conditions, actions will be taken, including update of the Safe Shutdown Analysis and applicable fire procedures, to address this change. Note that this scenario does not require that an additional single failure be postulated.
  - An analysis shows that the plant can be cooled from conditions for initiation of water solid steam generator operation to Cold Shutdown within 72 hours after shutdown, assuming initiation of water solid steam generator operation 50 hours after reactor shutdown. To facilitate drainage of water from the steam generators during water solid operation, a modification will be implemented to install drain piping which will direct drainage from each main steam header to the steam generator blowdown tank via the blowdown tank steam header.

Based on the above analyses, the requirement to achieve cold shutdown conditions within 72 hours after reactor shutdown continues to be met for EPU conditions.

Appendix R Compliance Strategies

As discussed in Section 2.5.1.4.2.1 above, in the event of a fire in the control complex, local operation of the turbine-driven auxiliary feedwater pump is used for decay heat removal, and a time validation showed that the operators were controlling auxiliary feedwater flow to the steam generators at 20 minutes after reactor shutdown. Analysis of steam generator dryout time for EPU plant conditions shows that it would take 35 minutes for both steam generators to boil dry assuming no feedwater addition . Thus, there continues to be adequate time for the operator to supply feedwater to the steam generators at EPU conditions. Actions will be taken to enhance local control of auxiliary feedwater flow to the steam generators.

The Safe Shutdown Analysis shows that at least one train of safe shutdown equipment is available to achieve safe shutdown following a fire in the defined fire areas at Ginna Station for current plant conditions. As stated above, modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

For a fire in the auxiliary building basement / mezzanine, the current Shutdown Analysis states that (1) depressurization of the RCS by ambient heat losses from the pressurizer without use of a PORV may be required, and (2) water solid steam generator cooldown to provide cold shutdown capability may be required For current plant conditions, an RCS cooldown rate based on a pressurizer cooldown rate of 3.75°F per hour is used for depressurization of the RCS in the temperature range of 550°F to 350°F. Based on results of the EPU analysis for cooldown using two ARVs and water solid steam generator cooldown analysis, a pressurizer cooldown rate of greater than 3.75°F per hour will be required at EPU conditions. Therefore, to ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available.

Time-Critical Tasks

The impact of the EPU on the three "time-critical" tasks described in LR section 2.5.1.4.2.1 follows:

- Closure of the RHR pump suction valve from the RWST after reactor shutdown for an Appendix R fire.

  The analysis for this task uses as input the limiting case for total reactor coolant system leakage for a 72 hour Appendix R event. The cooldown scenario applicable to this limiting case at current plant conditions includes RCS cooldown based on a pressurizer cooldown rate of 3.75°F per hour by ambient heat loss (PORVs not available), and water solid SG cooldown at 5°F per hour. As shown in the analysis, the RCS cooldown rate is essentially the same as the pressurizer cooldown rate in the temperature range where the RCS cooldown follows RCS depressurization by ambient heat loss from the pressurizer.

As indicated under "Appendix R Compliance Strategies" above, to ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available. Since the pressurizer cooldown rate at EPU conditions will exceed 3.75°F per hour, the limiting RCS leakage determined in the current analysis will remain bounding for EPU conditions. Therefore, the calculated time for closure of the RHR pump suction valve after an Appendix R fire (28 minutes) at current conditions, which prevents drain-down of the RWST to a level that requires makeup when PORVs are not available, remains bounding for EPU conditions.

- Restoration of charging flow after reactor shutdown for an Appendix R fire.

  An analysis determines that the time for pressurizer level to decrease from the level at hot zero power (35%) to the bottom level tap, due to total reactor coolant system leakage at current plant conditions, is 36 minutes. At EPU conditions, the pressurizer level low limit at hot zero power will be changed to 20% of span. The time required for pressurizer level to decrease from the low limit at EPU conditions to the bottom level tap is determined to be 23.9 minutes. Therefore, charging flow will need to be restored within 23 minutes after reactor shutdown to assure that pressurizer level will not decrease below the indicating range at EPU conditions. Modifications in support of restoring charging flow within 23 minutes are as follows:

  - Relocation of the "A" charging pump control power transfer switch from bus 14 to the charging pump room. This will reduce the time required to transfer control power for local operation of the charging pump.

  - Installation of a bank of high pressure air cylinders with a permanent connection to the charging pump speed control system normal air supply. This will ensure a backup air supply in the event of loss of normal air supply to the speed control system.

- Stopping charging flow after reactor shutdown for an Appendix R fire.

  The EPU does not affect the analysis for this task. Therefore, this time critical task remains unchanged for EPU conditions.

Other Supporting Analyses / Evaluations

The impact of the EPU on the supporting analyses / evaluations described in LR section 2.5.1.4.2.1 follows:

- An analysis shows that the city yard fire loop has the capability to simultaneously supply water to:

    - One emergency diesel generator
    - Two standby AFW pumps
    - Two standby AFW room coolers

The emergency diesel generator flow rate is based on the minimum service water flow rate to the diesel generator coolers during the LOCA injection phase, plus margin. As addressed in LR section 2.3.3, "AC Onsite Power System," the EPU does not affect the maximum steady state loading on the emergency diesel generators during the LOCA injection phase . Therefore, the emergency diesel generator flow rate for this case, 320 gpm, is not affected by the EPU.

Note: Since some equipment used for the LOCA injection phase is not required for Appendix R cooldown (e.g., containment fans, a containment spray pump), the minimum service water flow rate to the diesel generator coolers during the LOCA injection phase would envelope the minimum flow rate needed for Appendix R cooldown.

The standby AFW pump flow rate (225 gpm per pump) is the flow rate used in the water solid steam generator cooldown analysis for current plant conditions. In the plant cooldown analysis using water solid steam generator operation for EPU conditions, a standby AFW pump flow rate of 250 gpm per pump is used. A design analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby AFW pumps (250 gpm to each steam generator), and two standby AFW room coolers at EPU conditions.

The EPU does not affect the design flow rate of cooling water to the standby auxiliary feedwater room coolers.

- The EPU does not affect the reactor coolant pump lube oil collection system, and therefore the EPU does not affect the conclusions of the system effectiveness review.

Safe Shutdown Procedures

As addressed above, for the three fire areas which may require water solid steam generator operation (Auxiliary Building basement / mezzanine level, Auxiliary Building operating floor / intermediate level, and Screenhouse), actions required to implement the use of two steam generators in order to meet the Appendix R 72-hour cooldown requirement at EPU conditions include the need to update applicable fire procedures.

Implementation of a contingency activity to use auxiliary spray to reduce RCS pressure for a fire in the Auxiliary Building basement / mezzanine will require development of a new procedure and revision of the alternate shutdown fire procedure for a fire in the Auxiliary Building basement / mezzanine.

In order to accommodate changes in times to perform specific actions at EPU conditions, procedural activities may need to be re-ordered to ensure actions will be performed in the required time. As addressed above, at EPU conditions charging flow will be restored within 23 minutes after reactor shutdown to assure that pressurizer level will not decrease below the indicating range (change from 36 minutes at current conditions). Applicable fire procedures will be updated as required.

Evaluation of Impact of Renewed Plant Operating License Evaluations and License Renewal Programs

The fire protection program attributes and system components that are within the scope of license renewal are addressed in License Renewal SER section 2.3.3.6. SER section 3.3.2.3.2 addresses aging management of the fire water system and associated components. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them.

The License Renewal SER notes that Ginna stated that the Fire Protection Program is consistent with, but includes exceptions to NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." The SER addresses exceptions taken by Ginna to the GALL Report in the following areas:

- Testing frequency of the halon system
- Visual inspection of fire doors
- Periodic flow testing of infrequently used fire water system loops
- Visual inspection frequency of yard fire hydrants
- Frequency of fire hydrant flow tests

The NRC concluded that for those portions of the program which Ginna stated were consistent with the GALL program are consistent with GALL, and that, with regard to exceptions taken to the GALL program, Ginna had demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained.

Based on the uprate evaluation of elements of the fire protection program in this section, the EPU does not affect the evaluation / conclusions in the License Renewal SER regarding the fire protection program, and no new aging effects requiring management are identified.

2.5.1.4.3 Results

Apart from plant modifications, the EPU does not affect the fire protection program. Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Fire Protection Program requirements.

The EPU does not affect the elements of the fire protection program related to administrative controls and fire protection responsibilities of plant personnel.

The EPU does not affect the minimum set of pre EPU safe shutdown systems / components, including cables, credited with achieving safe and/or alternative shutdown. However, additional equipment is added to the list to account for the effects of increased decay heat.

The EPU does not affect the alternative shutdown methods. The EPU does not modify the function of any mechanical component in the alternative safe shutdown flow paths or introduce any plant equipment failure modes which will impact the ability to achieve any of the alternative shutdown functions. The EPU does not adversely affect any components or circuits that provide power, control, or indication to components required for alternative safe shutdown. Enhancements will be made to selected components or control circuits to improve alternative shut down capability.

The requirement to achieve cold shutdown conditions within 72 hours after reactor shutdown continues to be met for EPU conditions.

To ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available.

A time-critical task for restoration of charging flow after reactor shutdown will be modified to assure that pressurizer level will not decrease below the indicating range at EPU conditions.

An analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby AFW pumps, and two standby AFW room coolers.

The EPU does not affect the reactor coolant pump lube oil collection system.

Actions required to implement the use of two steam generators for three fire areas in order to meet the Appendix R 72-hour cooldown requirement at EPU conditions include the need to update applicable fire procedures.

2.5.1.4.4 Conclusion

The Ginna staff has reviewed the fire-related safe shutdown assessment and concludes that Ginna has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The Ginna staff further concludes that the Ginna Station Fire Protection Program will continue to meet the requirements of 10CFR50.48, Appendix R to 10CFR50, and will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-3 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to fire protection.

2.5.2 Pressurizer Relief Tank

2.5.2.1 Regulatory Evaluation

The pressurizer relief tank (PRT) is a pressure vessel that condenses and cools the discharge from the pressurizer safety and relief valves. The tank is designed with a capacity to accept discharge fluid from the pressurizer relief valve during a specified step-load decrease. The PRT is not safety-related and is not designed to accept a continuous discharge from the pressurizer. Ginna Nuclear Power Plant, LLC (Ginna) conducted a review of the PRT to ensure that operation of the tank at EPU conditions is consistent with transient analyses of related systems, and that failure or malfunction of the PRT will not adversely affect safety-related structures, systems, and components (SSCs). Ginna's review focused on any design changes related to the PRT and connected piping, and changes related to operational assumptions that are necessary in support of the EPU that are not bounded by previous analyses. In general, the steam condensing capacity of the tank and the tank rupture disk relief capacity should be adequate, taking into consideration the capacity of the pressurizer PORVs and safety valves, the piping to the tank should be adequately sized, and systems inside containment should be adequately protected from the effects of high-energy line breaks and moderate-energy line cracks in the pressurizer relief system.

The NRC's acceptance criteria for the PRT are based on:

- GDC-2, insofar as it requires that structures, systems, and components important-to-safety be designed to withstand the effects of earthquakes

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles

Specific review criteria are contained in the SRP section 5.4.11.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), section 3.1, the general design criteria used during the licensing of the Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s, the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983.

The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station safety-related structures, systems, and components will continue to be protected against the failure of the PRT consistent with the following:

- GDC-2 which is described in the Ginna UFSAR section 3.1.2.1.2, "GDC-2 – Design Bases for Protection against Natural Phenomena."

  As described in UFSAR section 5.4.8.2, in response to NUREG 0737, section II.D.1, and the NRC plant-specific piping evaluation, Westinghouse performed an analysis of the Ginna pressurizer safety and relief valve discharge piping system (UFSAR section 3.9.2.1.4). It was determined that the operability and structural integrity of the system were ensured for all applicable loadings and load combinations.

- GDC-4 which is described in Ginna UFSAR, section 3.1.2.1.4, "GDC-4 – Environmental and Missile Design Bases. As described in this UFSAR section, the Ginna Station received post-construction review as part of the SEP. The results of this review are documented in NUREG-0821. Conformance to the requirements of GDC-4 is also described in the following:

  — "Environmental Design of Mechanical and Electrical Equipment" (Ginna UFSAR section 3.11)

  — "Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping (Ginna UFSAR section 3.6)

    - "Pipe Breaks Inside Containment" (SEP, Topic III-5.A)
    - "Pipe Breaks Outside Containment" (SEP, Topic III-5.B)

  — "Missile Protection" (Ginna UFSAR section 3.5)

In addition to the evaluations described in the Ginna UFSAR, the PRT was evaluated for plant license renewal. System and system component materials of construction, operating history, and programs used to manage aging effects are documented in the license renewal SER, NUREG-1786. That SER describes Non-Class 1 reactor coolant system (RCS) components including the PRT in section 2.3.1.6, "Reactor Coolant System (Non-Class 1)." The PRT itself is not within the scope of License Renewal. The programs used to manage the aging effects associated with Non-Class 1 RCS components are discussed in section 3.1 of the Ginna Licensing Renewal submittal.

2.5.2.2 Technical Evaluation

2.5.2.2.1 Introduction

The PRT is described in the Ginna UFSAR section 5.4.8.1. The pressurizer safety and pressurizer power-operated relief valves (PORVs) described in the Ginna UFSAR section 5.4.10, discharge to the PRT. Principal design parameters of the PRT are given in the Ginna UFSAR Table 5.4-8, and a diagram of the tank is shown in the Ginna UFSAR Figure 5.4-9.

The pressurizer safety valves are required to have adequate capacity to ensure that the RCS pressure does not exceed 110% of system design pressure. This is the maximum pressure allowed by the ASME Code (section III, NB-7300 and NC-7300) for the worst-case loss-of-heat-sink event, that is, the loss-of-external-electrical load. The design of the surge line, safety valve inlet piping, and safety valve discharge piping (including the PRT sparger pipe) are also based on the safety valve design capacity.

The PORVs are required to have adequate capacity to prevent pressurizer pressure from reaching the high-pressure reactor trip set point for an external load reduction of up to 50% of rated electrical load.

The PRT design (including the tank level set points) is also based on the total safety valve capacity and conservatively sized to condense and cool a discharge of pressurizer steam equal to 110% of the steam volume above the full-power pressurizer water level set point. This sizing basis was selected to ensure the tank could accept the discharge from the pressurizer safety valves following the worst case loss-of-external load transient. The PRT is equipped with a rupture disk that has a relief capacity in excess of the combined capacity of the pressurizer safety valves.

The tank normally contains water in a predominantly nitrogen atmosphere. The volume of nitrogen gas in the tank is selected to limit the maximum pressure to 50 psig following a design discharge. The volume of water in the tank is selected to limit the maximum temperature to 200°F following a design discharge. The PRT level set points ensure adequate coolant is

maintained in the tank to condense and cool the design bases discharge, and to preclude the tank temperature and pressure from exceeding 200°F and 50 psig, respectively.

2.5.2.2.2 Description of Analyses and Evaluations

The PRT was evaluated based on the results of the loss of external load analysis described in LR section 2.8.5.2.1, Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum. The analysis was performed for the range of NSSS design parameters approved for EPU listed in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. The evaluation was conservatively performed for an analyzed NSSS thermal power of 1817 MWt.

The results of the loss of electrical load analysis confirmed that the installed capacity of the pressurizer safety valves is adequate to preclude RCS overpressurization at EPU conditions. Since the design of the surge line, safety valve inlet piping, safety valve discharge piping, PRT, PRT rupture disk, and sparger pipe are based on the capacity of the pressurizer safety valve capacity, it can be concluded these components are also adequate for EPU conditions.

In addition, the loss of external electrical load transient analysis for EPU determined that the mass and energy of the steam discharged from the pressurizer into the PRT is less than the design bases discharge. Since the current PRT level set points ensure adequate coolant is maintained in the tank to condense and cool the design bases discharge these setpoints remain adequate to preclude the tank temperature and pressure from exceeding 200°F and 50 psig, respectively at EPU conditions. In fact, the mass of coolant currently maintained in the PRT exceeds by more than 100% the mass of coolant required for the worst case loss of external load transient at EPU conditions.

The PORVs are required to have adequate capacity to prevent pressurizer pressure from reaching the high-pressure reactor trip set point for an external load reduction of up to 50% of rated electrical load. A margin to trip analysis was performed based on the range of NSSS design parameters for EPU listed in LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. The results of this analysis are described in LR section 2.4.2, Plant Operability. These analyses confirmed that the installed capacity of the PORVs is adequate to preclude a high-pressurizer pressure reactor trip at EPU conditions. Based on these results, it can also be concluded that the design of the inlet and discharge piping of the PORVs is adequate at EPU condition, since the design of this piping is based on the design capacity of the PORVs. The mass and energy addition to the PRT during load rejection is not limiting with respect to the design and operating set points for the PRT, since this transient discharge is less severe than the loss of external electrical load transient discharge.

Since the current design basis for the PRT bounds the EPU loss of external load analysis mass and energy addition, without any changes in the PRT set points, it can also be concluded that the current design basis for the PRT interface support functions are not impacted by the EPU.

These support functions include reactor makeup for cooling, nitrogen for pressure control, gas analyzer connection for periodic sampling, and means to vent and drain the tank.

LR section 2.2.2.2, Non-Class 1 Piping and Piping Supports, and LR section 2.5.1.3, Pipe Failures, evaluated the piping and supports for the PRT relative to meeting the Ginna Station current licensing basis requirements with respect to GDC-2 and GDC-4. These GDC address protection of structures, systems and components following design basis events that may result in failure of the non-safety grade PRT. Since the original design bases for the PRT and associated piping remain bounding at EPU conditions, the current licensing basis requirements with respect to GDC-2 and GDC-4 remain satisfied for EPU.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The PRT is not within the scope of License Renewal.

2.5.2.3 Results

Based on the results of the evaluations described in LR section 2.8.5.2.1 and LR section 2.4.3, the current design basis for the PRT, PRT rupture disk, PRT sparger, surge line, PORV and safety valve inlet piping, and PORV and safety valve discharge piping remains bounding for a loss of electrical load at EPU conditions. In addition, the current PRT design bounds the EPU loss of external load analysis mass and energy addition, such that following implementation of EPU the PRT continues to meet its design basis mass and energy addition, without any changes in the PRT level or pressure set points. The mass of coolant currently maintained in the PRT exceeds by more than 100% the mass of coolant required for the worst case loss of external load transient at EPU conditions.

The evaluations described in LR section 2.2.2.2 and LR section 2.5.1.3, determined the piping and supports associated with the PRT remain adequate for EPU conditions relative to satisfying the Ginna Station current licensing basis requirements with respect to GDC 2 and GDC 4.

2.5.2.4 Conclusion

Ginna has reviewed the pressurizer discharge to the PRT as a result of the EPU and concludes that the PRT will continue to operate in a manner consistent with transient analyses of related systems, and structures, systems, and components will continue to be protected against the failure of the PRT consistent with GDC-2 and GDC-4 with respect to the current Ginna licensing basis. Therefore, Ginna finds the EPU acceptable with respect to the design of the PRT.

2.5.3 Fission Product Control

2.5.3.1 Fission Product Control Systems and Structures

Regulatory Evaluation

Ginna Nuclear Power Plant, LLC (Ginna) review for fission product control systems and structures covered the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. Ginna's review primarily focused on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on

- GDC-41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents.

Specific review criteria are contained in SRP Section 6.5.3.

Current Licensing Bases

GDC 41 is addressed in Ginna UFSAR Sections 3.1.2.4.12, 6.5.1, and 6.5.2.

- The containment recirculation fan cooler (CRFC) system consists of four air handling systems, each including motor, fan, cooling coils, moisture separators and HEPA filters, duct distribution system, and instrumentation and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. Two of the four air handling systems are equipped with activated charcoal filter units, normally isolated from the main air recirculation stream, through which the air-steam mixture is bypassed to remove volatile iodine following an accident. The filter units are located on a platform above the operating floor. The charcoal filter units are not credited in the control room, exclusion area boundary (EAB) or low population zone (LPZ) dose calculations (see LR section 2.9.2, Radiological Consequences Analyses Using Alternative Source Terms).

- The containment spray system includes the injection of sodium hydroxide solution into the spray into the containment to remove elemental iodine. The system consists of redundant active components each supplied from separate electrical buses. No single active failure will cause both subsystems to fail to operate.

Not yet included in the Ginna UFSAR is the approved submittal for the new Control Room Emergency Air Treatment System (CREATS) and Alternate Source Term (AST) per References 1 and 2. The EPU dose calculations use the methodology contained in these submittals.

In addition to the evaluations described in the Ginna UFSAR, the above systems were evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

The SER discusses the systems in section 2.3.2.2, Containment Spray and Section 2.3.3.9, Containment Ventilation.

2.5.3.1.1 Technical Evaluation

The CRFCS is designed to remove fission products, from the containment atmosphere following a LOCA. The CRFCS consists of four units, each includes high efficiency particulate air (HEPA) filters, and 2 units include charcoal adsorbers. Two of the four units are required during the post-accident period. Each unit has 30,000 cfm flow capacity. During normal plant operation, the charcoal filters are by-passed.

In the event of a LOCA, the air flow can be directed through the charcoal adsorbers. However, the adsorbers are not credited for evaluating potential radiological consequences (References 1 and 2). Two CRFCS units recirculate 12,000 cfm within the lower (unsprayed) containment volume, and 48,000 cfm is assumed to mix the sprayed and unsprayed volumes. The offsite and control room dose analyses, presented in LR section 2.9.2, demonstrate the effectiveness of the CRFCS to minimize the release of radioactivity to the environment following a LBLOCA.

The CSS, in conjunction with the CRFCS, is designed to remove fission products from the containment atmosphere following the postulated LBLOCA. The CSS consists of two trains. Each train consists of a pump, two spray headers, and associated valves. Each train of CSS is independently capable of delivering 1,300 gpm of borated water from the RWST into the containment atmosphere. The spray pumps are automatically started following the coincidence of two sets of two-out-of-three high-high containment pressure signals. After 52 minutes into the accident, spray pump operation is terminated. No containment spray recirculation phase is assumed. The offsite and control room dose analyses, presented in LR section 2.9.2, demonstrate the effectiveness of the CSS to minimize the release of radioactivity to the environment following a LBLOCA.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the fission product control systems are within the scope of License Renewal. Aging management programs are addressed in the License Renewal SER section 3.2.2.4.2, Containment Spray System, and SER section 3.3, Auxiliary Systems. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Operating at EPU conditions does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.3.1.2 Results

The effect of the EPU is an increase in source term of approximately 20 percent, which is considered in the new LOCA dose analysis discussed in LR section 2.9.2. A review of this section indicates that the CRFCs and CSS, in conjunction with other SSCs, are effective in limiting both Control Room and

off-site dose to within regulatory guidelines.

2.5.3.1.3 References:

1. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), "R.E. Ginna Nuclear Power Plant – Modification of the Control Room Emergency Air Treatment System and Change to Dose Calculation Methodology to Alternate Source Term (TAC No. MB9123)," dated February 25, 2005.

2. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), R.E. Ginna Nuclear Power Plant – Correction to Amendment No. 87 Re: Modification of the Control Room Emergency Air Treatment System (TAC MB9123)," dated May 18, 2005.

2.5.3.1.4 Conclusion

The Ginna staff has performed an assessment of the effects of the proposed EPU on fission product control systems and structures. The Ginna staff has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU. The Ginna staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in post accident environments following implementation of the proposed EPU. Based on this, the Ginna staff also concludes that the fission product control systems and structures will continue to meet the current licensing basis with respect to the requirements of GDC-41. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

2.5.3.2 Main Condenser Evacuation System

2.5.3.2.1 Regulatory Evaluation

The main condenser evacuation system consists of two subsystems:

- The condenser air removal and priming ejectors (hoggers) that initially establish main condenser vacuum
- The condenser air removal steam jet air ejectors to maintain condenser vacuum once it has been established

The Ginna Nuclear Power Plant, LLC (Ginna) review focused on the effects of the proposed EPU on the system's capability to maintain condenser vacuum, modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the main condenser evacuation system are based on:

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.
- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Specific review criteria related to these GDC are contained in SRP section 10.4.2.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the later 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821,Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna's main condenser evacuation system design relative to conformance to:

- GDC-60 is described in the Ginna UFSAR section 3.1.2.6.1, General Design Criterion 60 – Control of Release of Radioactive Materials to the Environment. As described in this UFSAR section, the handling, control, and release of radioactive materials during Modes 1 and 2 are in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

- GDC-64 as described in the Ginna UFSAR section 3.1.2.6.5, General Design Criterion 64 – Monitoring Radioactive Release.

Radioactivity levels contained in the facility effluent discharge paths are continually monitored during normal and accident conditions by the station process radiation monitoring system as described in the Ginna UFSAR section 11.5. Implementation of the Ginna Radiation Protection Programs are intended to meet As Low As Is Reasonably Achievable (ALARA) concepts as described in UFSAR section 12.5.

In addition to the evaluations described in the UFSAR, the main condenser evacuation system was evaluated in the Ginna Station License Renewal Application. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

The above SER discusses the main condenser evacuation system (i.e., air ejectors) in section 2.3.4.4, Turbine Generator and Supporting Systems. Aging effects and the programs used to manage aging effects associated with main condenser evacuation system are discussed in SER section 3.4 of the Ginna License Renewal Application.

2.5.3.2.2 Technical Evaluation

2.5.3.2.2.1 Introduction

The condenser air removal system is discussed in the Ginna UFSAR section 10.7.6.2. The condenser air removal system removes non-condensable gases from the condenser to draw a vacuum for start up and subsequently maintain condenser vacuum during operation. The air removal system consists of two stage steam air ejectors and hogging air ejector. The two condensers share a common connection to the four first stage ejectors and two second stage ejectors. Two priming ejectors (hoggers) are provided to evacuate non-condensable gases from the condensate during plant start up.

2.5.3.2.2.2 Description of Analyses and Evaluations

The condenser air removal system must be capable of removing non-condensable gases and air in-leakage from the condenser shell (steam space) to maintain vacuum. Air in-leakage will not be adversely affected by the EPU since air in-leakage is entirely related to the physical design of the condenser and its state of integrity. In addition, any existing air in-leakage may be slightly reduced due to the higher condenser backpressure at EPU. Therefore, the air removal system is evaluated by comparing its removal capability with the expected increase in non-condensables resulting from the increased low pressure turbine exhaust flow rate at EPU conditions. Refer to LR section 2.5.5.2, Main Condenser, for additional discussion related to the condenser.

The four element set of two-stage steam jet air ejectors have a total rated capacity of 30 cfm (7.5 cfm each) at 70°F free dry air for both condensers. The air removal capability was compared to the Heat Exchange Institute Standards for Steam Surface Condensers recommended removal capacity and actual measurements of air ejector flow rate.

Refer to LR section 2.10.1, Occupational and Public Radiation Doses, for the evaluation of plant radioactive monitoring and control of releases of radioactive materials to the environment in compliance of GDC-60 and GDC-64. For gaseous radioactive material handling refer to LR section 2.5.6.1, Gaseous Waste Management System.

Since there is no potential for explosive gas mixtures in the condenser, it was not included in the evaluation.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the main condenser evacuation system are within the scope of License Renewal. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Operating the main condenser evacuation system at EPU conditions does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.3.2.3 Results

The estimated condenser evacuation rate will increase from 2.5 to approximately 3 scfm at EPU conditions. The higher EPU evacuation rate remains less than the steam jet air ejectors design capacity. Also, the condenser air removal system is normally operated with four first stage ejectors and one second stage ejector to maintain vacuum in the condenser. Therefore, the existing steam jet air ejectors are adequate for EPU without modifications.

The two priming ejectors (hoggers) are provided to evacuate non-condensable gases from the condenser during startup. Since startup conditions do not change due to EPU operation. The priming ejectors are adequate at EPU conditions.

The design of the main condenser evacuation system does not change following the implementation of the EPU. Therefore, the EPU does not impact the ability of Ginna Station to control radioactive material or the monitoring of radioactive material releases in accordance with GDC-60 and GDC-64, respectively. The impact of EPU on radiological effluent releases from Ginna Station and compliance with 10CFR50, Appendix I, is discussed in LR section 2.10.1, Occupational and Public Radiation Doses.

2.5.3.2.4 Conclusion

The Ginna staff has assessed the effects of the proposed EPU on the main condenser evacuation system. The Ginna staff concludes that the evaluation adequately accounts for the effects of the proposed EPU on the system's capability to remove non-condensable gases from the condenser during start up and normal operation. Based on this, the Ginna staff concludes that the main condenser evacuation system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-60 and GDC-64. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the main condenser evacuation system.

2.5.3.3 Turbine Gland Sealing System

2.5.3.3.1 Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The Ginna Nuclear Power Plant, LLC (Ginna) reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths).

The NRC's acceptance criteria for the turbine gland sealing system are based on:

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents
- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Specific review criteria are contained in SRP Section 10.4.3.

Ginna Current Licensing Basis

The turbine gland sealing system is described in Ginna UFSAR section 10.7.6.1. The turbine gland sealing system prevents air leakage into the turbine casing that could increase turbine windage losses and reduce condenser vacuum. It also prevents steam leakage from the turbine casing into the turbine building.

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station turbine gland sealing system design relative to conformance to:

- GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. The requirements of GDC 60 related to the turbine gland sealing system require that the plant design include means to control suitably the release of radioactive materials and that sufficient holdup capacity be provided. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

- GDC 64 is addressed in Ginna UFSAR section 3.1.2.6.5. The requirements of GDC 64 related to the turbine gland sealing system require that means shall be provided for monitoring the effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents. Ginna UFSAR section 3.1.2.6.5 states that radioactivity levels contained in the effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for Ginna Station as described in Ginna UFSAR sections 11.5 and 12.5.

Other Ginna UFSAR sections that address the design features and functions of the turbine gland sealing system include:

- Ginna UFSAR section 10.3.2, Main Steam System Description, which addresses the steam supply to the gland steam condenser

- Ginna UFSAR section 10.4.2, Feedwater and Condensate System Description, which describes the cooling water supply to the gland steam condenser

- Ginna UFSAR section 10.7.8, Erosion/Corrosion Monitoring Program, which addresses the fluid flow conditions in piping systems and the potential for erosion / corrosion.

In addition to the evaluations described in the Ginna UFSAR, the turbine gland sealing system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the turbine gland sealing system is described in sections 2.3.4.4, Turbine Generator and Supporting Systems. Aging effects, and the programs credited with managing those effects, are described in section 3.4.2.4.4, Turbine Generator and Supporting Systems.

2.5.3.3.2 Technical Evaluation

2.5.3.3.2.1 Introduction

The turbine gland sealing system prevents air leakage into the turbine casing and prevents steam leakage from the turbine casing into the turbine building. The turbine rotor is designed with labyrinth type glands / seals which provide a high resistance to steam or air flow along the shaft. Gland sealing steam is provided to the gland seal chamber to maintain a positive pressure of about 3 psig under all operating conditions. Excess steam leaks off from the gland and is collected in the gland steam condenser. Condensed steam drains from the gland steam condenser to the main condenser.

For plant startup, sealing steam is initially supplied from an external source, main steam. As the turbine load is increased, the turbine steam pressure increases and leakage from the high-pressure turbine glands and steam from the regulator valve supplies the steam sealing requirements for the low pressure turbine glands.

The gland steam condenser maintains a pressure slightly below atmospheric in the gland leakoff system to prevent the escape of steam from the glands to the turbine building. The gland steam condenser also condenses the steam vapor to recover its energy. The condensed steam is drained to the main condenser. Cooling of the gland steam condenser is provided by the condensate system. The entrained air and other noncondensible vapors leaving the gland steam condenser are discharged through an atmospheric vent by an air exhauster. A radiation monitor is provided on the discharge vent pipe to monitor the effluent being released to the environment.

2.5.3.3.2.2 Description of Analyses and Evaluations

The turbine gland sealing system was evaluated to ensure that the system design will continue to control the release of radioactive material from steam in the turbine to the environment. It will also continue to provide sufficient sealing steam to the high pressure and low pressure turbine glands from plant start-up through full power operation at the EPU. The evaluation determined whether changes are required to the existing design of the system and its components in order to meet their design functions during EPU conditions and whether such changes affect the system's ability to control radioactive releases.

Other evaluations of the turbine gland sealing system, piping and components are addressed in the following LR section:

- Erosion / corrosion issues – LR section 2.1.8, Flow Accelerated Corrosion
- Cooling water to the gland steam condenser - LR section 2.5.5.4, Condensate and Feedwater Systems
- External steam supply for turbine gland sealing - LR section 2.5.5.1, Main Steam

Evaluation of Impact on Renewed Plant Operating License Evaluations and Licensing Renewal Programs

In regard to the aging programs and aging influences described in the License Renewal SER NUREG 1786, there are no effects due to the EPU operating conditions. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the turbine gland sealing system at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.3.3.2.3 Results

The gland sealing steam flow and condensate cooling water flow to the gland steam condenser both increase at normal EPU plant operating conditions due to the turbine steam conditions. The increase in steam flow to the gland condenser due to the EPU is small compared to the existing steam flow and results primarily from the increase in the high pressure turbine control valve stem leak-off flows due to the new high pressure turbine. The increased steam flow will not affect the operation of the gland steam condenser. The increase of the gland steam condenser cooling water flow is adequately supplied by the condensate system as described in LR section 2.5.5.4, Condensate and Feedwater Systems.

There is no increase in the required external steam supply flow from the main steam system since it is primarily used to supply sealing steam for turbine / plant startup and at reduced power operation. As plant power increases the amount of main steam flow required for gland sealing decreases. Therefore it is unaffected by the EPU.

Since the steam flow to the gland steam condenser is not expected to increase significantly due to the EPU, there is no significant change in the air and non-condensible exhaust flows exiting the gland steam condenser due to the EPU. The small increase in exhaust flow due to EPU is within the capability of the existing exhaust radiation monitor setpoints as described in LR section 2.10.1, Occupational and Public Radiation Doses.

The EPU has no impact on the gland sealing steam requirements for the low pressure turbines. The major impact of the EPU on the turbine gland sealing system is to increase both the high pressure turbine control valve steam leak-off flows and the high pressure turbine gland steam leak-off flows. The control valve stem leak-off flow increase is due to going from partial arc admission with the existing high pressure turbine to full arc admission with the new high pressure turbine. Full arc admission will cause all four turbine control valves to be throttled at full power conditions. In addition, all four turbine control valves will be modified to a "high lift" design to reduce the pressure drop across them. These modifications will result in an additional 5% per valve increase in stem leakoff flow. These flow rates are small and result in a slight increase in steam flow to the gland steam condenser as previously discussed.

The high pressure turbine gland steam leak-off flow increases due to the increased high pressure turbine exhaust pressure associated with the EPU. This increased leak-off flow is then used to supply sealing steam to the low pressure turbines. Since the low pressure turbine sealing steam requirements do not increase due to the EPU, the increase in the high pressure turbine gland leak-off flow may require the installation of a spillover line in the gland supply to the low pressure turbine glands. This spillover line would direct any excess steam flow from the high pressure turbine glands to the main condenser. The need for a spillover line addition to the gland sealing system due to EPU is presently under evaluation by Ginna. Other than the possible addition of a spillover line, no additional physical changes are required to the turbine gland sealing components and piping due to the changes in steam and water flow associated with the EPU.

The increase in sealing steam flow and cooling water flow raises the velocity in the piping. The potential for increased erosion / corrosion has been evaluated in LR section 2.1.8, Flow Accelerated Corrosion.

The evaluation of the turbine gland sealing system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, as described in the Offsite Dose Calculation Manual.

The evaluation of the turbine gland sealing system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. This design capability remains unchanged by the EPU. Radioactivity levels contained in the effluent discharge paths in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for Ginna Station. Refer to LR section 2.10.1, Occupational and Public Radiation Doses.

2.5.3.3.3 Conclusion

The Ginna staff has reviewed the assessment of the required changes to the turbine gland sealing system and concludes that the assessment has adequately evaluated these changes. The Ginna staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with the current licensing basis with respect to the requirements of GDC-60 and GDC-64. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

2.5.4 Component Cooling and Decay Heat Removal

2.5.4.1 Spent Fuel Pool Cooling and Cleanup System

2.5.4.1.1 Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The Ginna Nuclear Power Plant, LLC (Ginna) review for the proposed EPU focused on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions.

The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on:

- GDC-5, insofar as it requires that structures, systems, and components (SSCs) important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from important-to-safety SSCs to a heat sink under both normal operating and accident conditions be provided

- GDC-61, insofar as it requires that fuel storage systems be designed with residual heat removal (RHR) capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss-of-fuel-storage-coolant inventory under accident conditions

Specific review criteria are contained in SRP section 9.1.3, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of section 2.1 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station spent fuel pool cooling system design relative to conformance to:

- GDC–5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna Station is a single unit installation so there are no shared SSCs.

- GDC-44 is described in Ginna UFSAR section 3.1.2.4.15, General Design Criterion 44 – Cooling Water. GDC 44 addresses provision of a system to transfer heat from safety-related SSCs to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating conditions. Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling and service water design features to transfer heat to the ultimate heat sink. Note that the spent fuel pool cooling system is not specifically addressed; however, the system does provide the heat removal design functions considered under this GDC.

- GDC-61 is described in Ginna UFSAR section 3.1.2.6.2, General Design Criterion 61 - Fuel Storage and Handling and Radioactivity Control. GDC 61 addresses the fuel storage and systems which contain radioactivity and shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety-related components, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal (RHR) capability having reliability and testability that reflects the importance to safety of decay heat and other RHR, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. Ginna UFSAR section 3.1.2.6.2 states that the spent fuel pool cooling system, which contains radioactivity, is designed to ensure adequate safety under normal and postulated accident conditions.

  A. Components are designed and located such that appropriate periodic inspection and testing may be performed.

  B. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy.

  C. Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

  D. The spent fuel pool cooling system provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed such that, in addition to permanently installed equipment, temporary connections and equipment can also be utilized.

  E. The spent fuel pool is designed such that no postulated accident could cause significant loss of coolant inventory.

The spent fuel pool cooling system is described in Ginna UFSAR section 9.1.3, Spent Fuel Pool Cooling System. The Ginna Station licensing bases regarding the spent fuel pool cooling system design features, operating modes, cooling capabilities, pool temperatures and failure modes are described in this UFSAR section. As stated in Ginna UFSAR section 9.1.3, the system is designed to remove decay from fuel assemblies stored in the spent fuel pool. The system also purifies and maintains water clarity in the spent fuel pool. Borated water in the spent fuel pool provides radioactive shielding. The spent fuel pool cooling and purification piping is arranged so that failure of any line does not drain the spent fuel pool. The heat from the spent fuel pool is rejected to the service water system. The maximum design temperature of the service water is a lake temperature of 85°F, which is the Ginna Station design basis temperature used for safety related evaluations based on lake historical data.

Full Core Off-Load vs. Lake Temperature

The required time after shutdown prior to initiating a full core off-load is strongly dependent upon the available lake temperature. Use of an 85°F lake temperature for assessing full core off-loads during a typical fall or spring outage associated with an eighteen month fuel cycle may require delaying core off-load for an extended period. As stated in Ginna UFSAR section 9.1.3.4.1.2, actual lake temperatures greater than 80°F are very infrequent. Therefore, the EPU analyses of the spent fuel pool cooling system for typical full core off-loads were performed with lake temperatures of 40°F, 60°F, and 80°F. It is noted that for any full core off-load, each off-load scenario is conservatively evaluated on a cycle specific basis to identify the minimum required time after shutdown as required by Ginna Technical Requirements Manual section 3.9.4. This would include defining a bounding lake temperature to be used in calculating available spent fuel pool cooling capability.

Single Failure Consideration

Consideration of a single passive failure for spent fuel pool cooling is not required by NRC SRP 9.1.3 and is not a part of the licensing basis for the Ginna Station spent fuel pool cooling system. As discussed in SRP 9.1.3, for the "maximum normal heat load" (of the spent fuel pool) there should be suitable redundancy of components so that safety functions can be performed assuming a single active failure of a component coincident with the loss of offsite power. For the "abnormal maximum heat load" (defined in the SRP as full core unload), a single active failure need not be considered; however, Ginna Technical Requirements Manual section 3.9.4 conservatively requires that two spent fuel pool cooling systems shall be operable, each commensurate with the spent fuel pool cooling heat load. NRCSEP Topic IX-1 evaluated Ginna's conformance to the SRP and its conformance to SRP section 9.1.3 has been documented.

Fuel Pool Re-Racking

In 1998, NRC approval was received for re-racking portions of the spent fuel pool. This included replacement of the three region 1 flux trap racks with two types of high density fixed neutron absorber type racks. In addition, the approval allowed attachment of similar high-density fixed neutron absorber type racks to the north and south faces of the Boraflex neutron absorber racks that constituted region 2 prior to the approval. The installation of the new high-

density racks was planned in two phases. In 1998, phase 1 added five rack modules to create region 1 for storage of fresh fuel and two additional rack modules to augment region 2. This modification increased the number of usable cells from 1015 to 1320 (one additional cell is capped). Although currently licensed for the phase 2 attachment of the remaining six high-density rack modules (net addition of 48 storage cells) to the Boraflex racks in region 2, Ginna expects that future implementation of on-site dry cask storage will make this unnecessary.

Other Design Features

Other Ginna UFSAR sections that address the design features related to the spent fuel pool cooling system include:

- Ginna UFSAR section 3.5.1.3.2, Systems Whose Failures May Result in Radioactivity Release, which describes the plant evaluations and design features which protect the spent fuel pool cooling system from internally generated missiles.

- Ginna UFSAR section 9.2.1, Service Water System, which describes the cooling water provided to the spent fuel pool heat exchangers.

- Ginna UFSAR section 9.4.4, Spent Fuel Pool Area Ventilation, which describes the ventilation system design and airborne activity control features provided for the spent fuel pool area.

In addition to the evaluations described in the UFSAR, the Ginna Station's spent fuel pool cooling system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

With respect to the above SER, the spent fuel pool cooling system is described in section 2.3.3.3, Spent Fuel Pool Cooling and Fuel Storage. Aging effects, and the programs used to manage the aging effects associated with spent fuel pool cooling system, are discussed in section 3.3.2.4.3, Spent Fuel Pool Cooling and Fuel Storage.

2.5.4.1.2 Technical Evaluations

2.5.4.1.2.1 Introduction

The spent fuel pool cooling system consists of three individual loops, each with pumps, heat exchangers, and associated piping, valves, and hoses. The primary cooling path is loop "B," permanently installed, including spent fuel pool pump B, spent fuel pool heat exchanger B, and piping. This loop is safety-related and seismically qualified and functions as the preferred loop for ensuring adequate cooling in the spent fuel pool. The backup loops include loop "A," permanently installed, including spent fuel pool pump A, spent fuel pool heat exchanger A, and

piping, and the standby loop, skid-mounted, including the spent fuel pool standby pump, heat exchanger, and hoses. The skid mounted pump is permanently installed while the skid mounted heat exchanger is not permanently installed. Since the combined heat capacity of the "A" loop and standby loop exceeds the heat capacity of the "B" loop (see Table 2.5.4.1-2), they provide 100 percent redundancy to the "B" loop. All three heat exchangers use service water (lake water) to cool the spent fuel pool water.

There is also the ability to align pump A and the standby pump individually or in parallel to supply spent fuel pool water to spent fuel pool heat exchanger B. In addition, there is also the ability to provide fire water for cooling of heat exchanger A and the standby heat exchanger.

The heat removal criteria of the spent fuel pool cooling system as described in UFSAR section 9.1.3.1 are that the system should be capable of maintaining the spent fuel pool temperature less than or equal to 120°F during normal plant operation and normal refueling operations and less than or equal to 150°F during full core discharge situations. The 120°F is not a safety requirement but is an administrative limit set for operator comfort during normal plant operation and a normal refueling operation. For structural integrity reasons, the pool water temperature is not to exceed 180°F. In order to provide sufficient time to take corrective action in the event of spent fuel pool cooling system failure, the pool temperature limit is not to exceed 150°F for all modes of operation including a full core discharge. Normal refueling operations are conducted approximately every 18 months and are defined for the purpose of this evaluation as removing approximately one-third of the core (approximately 44 fuel assemblies) from the core and placing them in the spent fuel pool. Full core discharge occurs when all the fuel in the reactor (121 fuel assemblies) is placed in the spent fuel pool. The full core will be discharged once every 10 years for in-service inspection. Full core discharge may also occur on other occasions when it is deemed necessary to support planned refueling outage activities. Therefore, full core discharges may occur several times during a 10 year inservice inspection interval.

2.5.4.1.2.2 Description of Analyses and Evaluations

The spent fuel pool cooling system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations were conservatively performed for an analyzed NSSS core power of 1811 MWt. It is noted that no changes to system flows, pressures or temperatures are required due to EPU. The major impact of EPU is the potential increase in shutdown time required before core off-loads can be initiated due to the increased decay heat associated with EPU operation. The evaluations determined whether the existing design parameters of the spent fuel pool cooling system and components meet the EPU conditions for the following design aspects:

- Design pressure / temperature of piping and components
- Flow velocities
- Cooling capacity – normal refueling
- Cooling capacity – full core off-load
- Loss of cooling
- Concrete wall temperature

- Purification subsystem

Other evaluations related to the spent fuel pool cooling system and components are addressed in the following LR section:

- Piping / component supports – LR section 2.2.2.2, BOP (All Non-Class 1)
- Spent fuel pool instrumentation - LR section 2.4.1, Additional Review Areas (Instrumentation and Controls) (BOP)
- Protection of the spent fuel pool cooling system from internally generated missiles – LR section 2.5.1.2, Missile Protection
- Service water cooling for spent fuel pool heat exchangers – LR section 2.5.4.2, Service Water System
- Protection against dynamic effects of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Fuel pool ventilation and control of airborne radioactivity - LR section 2.7.4, Spent Fuel Pool Area Ventilation
- Fuel pool criticality, fuel movement and storage; evaluation of the Boraflex storage racks - LR section 2.8.6.2, Spent Fuel Storage

System / Component Design Parameters

No modifications to the spent fuel pool cooling system are required for EPU to comply with Ginna's existing regulatory commitments.

The current spent fuel pool cooling system design pressure and temperature are 150 psig and 200°F, respectively. The maximum operating conditions at EPU, including the maximum spent fuel pool temperature during cooling system operation of 150°F, do not change. The only time the 150°F temperature could be exceeded is during a complete loss of all spent fuel pool cooling. Therefore, the existing design pressure and temperature of the system components; heat exchangers, pumps, valves, demineralizers, strainers, and filters are acceptable at EPU.

The current spent fuel pool cooling flow rate provides for acceptable heat removal in the spent fuel pool heat exchangers. Therefore, no changes are required to the fuel pool cooling pumps / motors and the system piping velocities are unchanged at EPU conditions.

Cooling Capacity – Normal Refueling

A normal refueling outage occurs approximately every 18 months and is based upon removal of approximately one-third of the 121 reactor core fuel assemblies. For the EPU operating conditions, the number of fuel assemblies to be discharged is expected to alternate between 44 and 45 fuel assemblies per outage, with the exception of the 2006 and 2008 refuelings, which are expected to discharge 49 fuel assemblies based on the transition from the existing 1520 MWt full power rating to the EPU rated power. Presently, the actual number of spent fuel rack storage positions installed in the SFP is 1321. Due to the present inventory of stored fuel

assemblies, Ginna plans to implement on-site dry cask storage around 2009 to accommodate the off-loads out through 2029. Therefore, to maximize the pool residual heat load from the existing 1321 storage positions a bounding full core off-load in 2029 was assumed. The off-load analysis assumed completely filling all 1321 available storage locations. The previously discharged fuel was assumed to be the most recent spent fuel available to completely fill the pool. This assumes that fuel assemblies being stored in dry casks are the oldest fuel assemblies with the lowest decay heat rates. Based on these assumptions the residual spent fuel pool heat load in 2029 prior to an off-load was calculated to be 3.9 MBTU/hr for the 1200 fuel assemblies previously discharged to the pool.

This residual pool heat load was obtained using decay heat loads determined with the ORIGEN2 computer code. ORIGEN2 has previously been used for the Ginna Re-racking Licensing Amendment Request in 1998. For normal refueling in 2029, the decay heat was calculated for 45 fuel assemblies irradiated for 18 months at a power level of 1811 MWt and was added to residual heat load in the spent fuel pool. Spent fuel pool cooling lineups consisting of Loop B and either Pump A or the Standby pump cross connected to Heat Exchanger B were evaluated with service water temperatures of 40°F, 60°F, and 80°F. Table 2.5.4.1-1 shows the offload delay time after shutdown required to not exceed the 120°F administrative limit for spent fuel pool temperature. The table results show that for an 80°F lake temperature required shutdown time would exceed 100 hrs to maintain a pool temperature of 120°F. If a partial offload at 100 hours was desired with an 80°F service water temperature then the maximum spent fuel pool temperature would be 141°F using Loop B and 147°F using either Pump A or the Standby pump cross connected to Heat Exchanger B.

Table 2.5.4.1 – 1
Normal Off-load

	Lake Water Temperature °F	SFP Heat Exchanger Capacity Mbtu/hr	Decay Heat Load Mbtu/hr	Delay Time Required for 120°F Admin. Limit Temp. hours
SFP Loop B (normal)	40	17.9	12.9	100[1]
	60	13.5	12.9	100[1]
	80	9.0	9.0	376
SFP Pump A or SFP Standby Pump with HX B (cross-connect)	40	16.2	12.9	100[1]
	60	12.1	12.1	124
	80	8.1	8.1	605

[1] – Minimum time based on delay time used for radiological releases from a fuel handling accident.

Cooling Capacity – Full Core Off-load

As stated in Ginna UFSAR section 9.1.3.4.1.8, when performing full core off-loads, each off-load scenario is conservatively evaluated on a case by case basis to identify the minimum required time after shutdown when fuel off-load can commence in order to maintain spent fuel pool bulk temperature at or below 150°F. Ginna previously committed to performing these cycle specific analyses for full core off-loads in its Re-racking Licensing Amendment Request in 1998.

For the purpose of the evaluations included in this report, a full core off-load of all 121 fuel assemblies was assumed to occur in the year 2029 and results in a total of 1321 fuel assemblies stored in the spent fuel pool. Though the current Ginna Technical Specifications permit storage in the spent fuel pool of 1879 assemblies, 1321 fuel assemblies was used as this is current number of storage locations available as noted in UFSAR section 9.1.2.2. Storing of 1879 fuel assemblies in the existing pool would require performing fuel rod consolidation so that the fuel rods from two fuel assemblies can be stored in a single canister. Although this is possible, it is not expected to be implemented over the remaining licensed life of Ginna. In lieu of fuel rod consolidation, Ginna plans to implement on-site dry cask storage in 2009. Therefore, the total number of fuel assemblies that would be expected to be cooled by the spent fuel pool cooling system is limited to the 1321 storage positions in racks presently installed in the spent fuel pool at Ginna. The analysis of the spent fuel pool heat load assumes that the oldest fuel assemblies with the least amount of decay heat would be removed from the pool and stored in on-site dry casks.

For full core off-load in 2029, the decay heat was calculated using ORIGEN2 for 1/3 of the core with 1.5 years burnup, 1/3 of the core with 3 years burnup, and 1/3 of the core with 4.5 years burnup at a power level of 1811 MWt and was added to residual heat load in the spent fuel pool. Table 2.5.4.1-2 shows the offload delay time after shutdown required to maintain the spent fuel pool at 150 °F with service water temperatures of 40°F, 60°F, and 80°F with the various system lineups currently utilized at the Ginna Station.

Table 2.5.4.1 – 2
Full Core Off-load

	Lake Water Temperature (°F)	SFP Heat Exchanger Capacity (Mbtu/hr)	Decay Heat Load (Mbtu/hr)	Delay Time Required for 150°F Tech. Spec. Limit Temp. (hours)
SFP Loop B (normal) OR SFP Pump A and SFP Standby Pump with HX B (cross-connect)	40	23.4	23.4	131
	60	19.2	19.2	230
	80	14.9	14.9	481
SFP Loop A and SFP Standby Loop	40	27.0	25.7	100[1]
	60	22.1	22.1	156
	80	17.2	17.2	319

[1] - 100 hour time after shutdown is based upon doses from a fuel handling accident

The results in Table 2.5.4.1-2 are conservative in assuming design flow rates and design heat exchanger heat transfer coefficients. Plant data show that flow rates greater than design are achievable and that the heat exchangers are fouled less than design, thereby realistically resulting in better heat transfer performance such that a shorter offload delay time after shutdown is possible. Therefore, cycle specific analyses will continue to be performed prior to each full core off-load, and the requirement to have two spent fuel pool cooling systems operational, each commensurate with the spent fuel pool cooling heat load, will be maintained consistent with Ginna's current Technical Requirements Manual requirements.

Loss of Cooling

The spent fuel pool cooling system was evaluated for a complete loss of B loop cooling. With the spent fuel pool at an initial temperature of 150°F, the time to reach both180°F and 212°F are shown in Table 2.5.4.1-3. The heat-up times listed in Table 2.5.4.1-3 increase with increasing service water (lake) temperature. This is because the delay time for offloading of spent fuel to the pool is determined by the service water temperature at the time of offload. The increased spent fuel pool cooling system heat removal capability associated with a colder lake temperature allows shorter offload delays. Therefore, if cooling is lost at completion of offloading with the service water at lower temperatures, the heat load in the pool is higher, which results in faster pool heat-up rates. In addition, Table 2.5.4.1-3 lists make up water requirements to accommodate boil off at a pool temperature of 212°F. The heat-up analysis conservatively neglected any cooling associated with heat transfer to the spent fuel pool concrete walls, convective cooling to the ambient air and any evaporative cooling from the pool surface. The analysis took credit for the thermal inertia of the spent fuel pool water as well as

the thermal inertia associated with the spent fuel pool racks, stored fuel assemblies and the spent fuel pool steel liner

In actuality, following a loss of cooling from the B loop, the A loop heat exchanger can be made operational within forty-five minutes as discussed in UFSAR section 9.1.3.4.3. Following initiation of cooling by heat exchanger A the pool heat up rate would be decreased by approximately 50%. The resulting times to heat up the pool to 180°F would be 5.2, 6.6, and 8.7 hours for 40, 60, and 80°F SW temperatures, respectively. Since the spent fuel pool standby heat exchanger can be put in operation within three hours the maximum pool temperature would not be expected to reach 180°F under the conditions conservatively assumed.

A makeup water flow rate of 60 gpm can be made available from the refueling water storage tank in less than 15 minutes. As an alternative, 50 gpm of water from the chemical and volume control system hold up tanks can also be made available in approximately 15 minutes, which bounds all but one case presented in Table 2.5.4.1-3. For this case where the alternate makeup source is insufficient, the off-load time can be delayed until the requirement is met. Therefore, sufficient makeup is available for the spent fuel pool cooling system in the event of a loss of normal cooling capability.

Table 2.5.4.1 – 3
Full Core Off-load – Loss of Cooling

Operating Cooling Loop	Service Water Temperature (°F)	Decay Heat Load (MBtu/hr)	Time to 180°F in Fuel Pool (hours)	Time to 212°F in Fuel Pool (hours)	Required Make-up Rate (gpm)
SFP Loop B (normal) OR SFP Pump A and SFP Standby Pump cross-connected To SFP HX B	40	23.4	2.6	5.3	48.5
	60	19.2	3.1	6.5	39.8
	80	14.9	4.0	8.4	31.1
SFP Loop A (normal) AND SFP standby Loop (normal)	40	27.0	2.4	4.9	52.8
	60	22.1	2.7	5.7	45.6
	80	17.2	3.5	7.3	35.5

Concrete Temperature

The preceding analyses of normal operation and planned fuel offloading scenarios show that the 180° structural design limit is not reached. Sufficient time is available following a loss of all cooling to restore alternate cooling of the pool prior to reaching 180°F. Therefore, the maximum concrete temperature is not expected to exceed 180°F.

Purification Subsystem

The EPU has no impact on the hydraulic portions of the purification subsystem. The current purification flow rate is adequate for EPU conditions. No equipment changes in the purification loop are required to support the uprate. The purification subsystem may experience a slight increase in the frequency of demineralizer resin replacement due to higher levels of fission

products in the pool. However, any significant increase in fission product inventory in the primary coolant system due to the EPU will be mitigated by RCS cleanup systems prior to transmission to the spent fuel pool.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the spent fuel pool cooling system are within the scope of License Renewal as identified in the License Renewal Safety Evaluation Report, NUREG-1786, Section 2.3.3.3. Aging effects, and the programs used to manage the aging effects associated with spent fuel pool cooling system, are discussed in SER Section 3.3.2.4.3, Spent Fuel Pool Cooling and Fuel Storage. EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The EPU conditions do not add any new or previously unevaluated materials to the spent fuel pool cooling system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

Results

The following subsections evaluate the specific spent fuel pool cooling system and component licensing, design and performance capabilities while at EPU conditions.

The evaluation of the spent fuel pool cooling system capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in Ginna UFSAR section 3.1.2.4.15. Although the spent fuel pool cooling system is not specifically addressed by this GDC, the system does provide for heat removal from the fuel pool and transfers the heat ultimately to the environment. The spent fuel pool cooling system provides this capability under both normal operating and accident conditions. The effect of a single failure on the system's capability to achieving this function is evaluated in accordance with the Ginna current licensing basis described in UFSAR section 9.1.3, Spent Fuel Pool Cooling System. The implementation of EPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described below.

The evaluation of the spent fuel pool cooling system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate periodic inspection and testing features, with suitable shielding for radiation protection, with appropriate containment, confinement, and filtering systems, with appropriate residual heat removal capability and with design features to prevent significant reduction in fuel storage coolant inventory under accident conditions. These design capabilities remain unchanged by the EPU. Refer to LR section 2.7.4, Spent Fuel Pool Area Ventilation, for additional details.

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

2.5.4.1.3 Conclusions

The Ginna staff has reviewed the assessment related to spent fuel pool cooling and cleanup system and concludes that the assessment has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. Based on this review, the Ginna staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-5, GDC-44, and GDC-61. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.4.2 Service Water System

2.5.4.2.1 Regulatory Evaluation

The service water system provides essential cooling to safety-related equipment and also provides cooling to non-safety-related auxiliary components that are used for normal plant operation. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the characteristics of the service water system components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with loss-of-offsite power). The Ginna review focused on the additional heat load that would result from the EPU.

The NRC's acceptance criteria are based on:

- GDC-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from important-to-safety SSCs to a heat sink under both normal operating and accident conditions be provided

Specific review criteria are contained in NRC Standard Review Plan (SRP), Section 9.2.1, as supplemented by NRC Generic Letter (GL) 89-13 and GL 96-06 as provided in Matrix 5 of RS-001, Revision 0, "Station Service Water System."

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station Service Water System design relative to conformance to:

- GDC 4 is described in the Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the SEP. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant.

  - Environmental Design Of Mechanical And Electrical Equipment (UFSAR Section 3.11)
  - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (UFSAR Section 3.6)
    - Pipe Breaks Inside Containment (SEP Topic III-5.A)
    - Pipe Breaks Outside Containment (SEP Topic III-5.B)

- GDC 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna Station is a single unit installation so there are no shared SSCs.

- GDC 44 is described in Ginna UFSAR section 3.1.2.4.15, General Design Criterion 44 – Cooling Water. GDC 44 addresses the provision of a system to transfer heat from SSCs important to safety to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

  Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling and service water design features to transfer heat to the ultimate heat sink. The systems provided to transfer heat from safety-related components to the ultimate heat sink of Lake Ontario consist of the service water and the component cooling water systems described in Ginna UFSAR sections 9.2.1 and 9.2.2. Service water is supplied by four pumps, two being fed power from one safeguards bus, the other two from another safeguards bus. Only one pump is needed during safe shutdown operation or during the injection phase of a postulated loss-of-coolant accident. Evaluation of service water flows during post-LOCA recirculation phase operation has demonstrated that two service water pumps are required during the recirculation phase for EPU, at elevated service water temperatures. A commitment to modify the Technical Specification Bases for EPU is thus required. The system is operable from offsite power or from emergency onsite power (from the diesel generators). No single active failure results in system loss of function for those functions important to safety.

Other Ginna UFSAR sections that address the design features and functions of the service water system include:

- Ginna UFSAR section 3.7.3.7.3.10, Service Water System, which describes the portions of the service water system included in the seismic piping upgrade program.

- Ginna UFSAR section 6.2.4, Containment Isolation system, which describes the design features of the service water system provided for containment isolation.

- Ginna UFSAR section 6.5.1.3, NRC Generic Letter 96-06 Requirements, which addresses overpressurization of isolated piping inside containment and boiling / flow blockage / water hammer effects in service water piping to the containment recirculation fan coolers.

- Ginna UFSAR section 9.2.1.2.6, Service Water Fouling, which addresses service water fouling in heat exchangers (NRC Generic Letter 89-13).

- Ginna UFSAR sections 11.5.2.2.9, Containment Service Water Monitor and 11.5.2.2.13, Spent Fuel Pool Heat Exchanger Service Water Monitors, which describe the radiation monitoring of service water.

In addition to the evaluations described in the UFSAR, the service water system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, the service water system is described in section 2.3.3, Auxiliary Systems. Aging effects, and the programs used to manage the aging effects associated with service water, are discussed in section 3.3 of the SER.

2.5.4.2.2 Technical Evaluation

2.5.4.2.2.1 Introduction

The service water system is described in Ginna USFAR section 9.2.1. The service water system takes suction from Lake Ontario via the screen house and supplies cooling water to various turbine plant loads as well as auxiliary reactor plant loads. Service water is the normal supply to the standby auxiliary feedwater system and an alternate supply to the preferred auxiliary feedwater system. The system is designed to provide adequate cooling to critical and non-critical loads during Modes 1 and 2 and to critical loads during abnormal and accident conditions. The system normally discharges back into Lake Ontario via the discharge canal. An alternate discharge line to Deer Creek is available for selected auxiliary building service water system loads.

The service water supply temperature varies with the Lake Ontario water temperature. The maximum service water temperature used for turbine cycle equipment design and performance analyses is 80°F. The maximum design temperature used for safety related plant evaluations is 85°F.

The service water system consists of four service water pumps, a single loop supply header, isolation valves, and a normal and standby discharge header. The physical design of the service water system is such that one service water pump from each class 1E electrical train is arranged on each of the two discharge headers which then supplies the service water loop header. All portions of the service water system (pumps, piping, etc.) serving safeguards equipment are designed as Seismic Category I. All other portions of the service water system serving nonsafety loads are designated as nonseismic and are capable of being isolated from the Seismic Category I portion.

2.5.4.2.2.2 Description of Analyses and Evaluation

The service water systems and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations compared the existing design parameters of the systems/components with the EPU conditions for the following design aspects:

- Service water flow and heat removal requirements
- Design pressure / temperature of piping and components
- Overpressurization of isolated piping inside containment and boiling / flow blockage / water hammer effects in service water piping to the containment recirculation fan coolers (NRC Generic Letter 96-06)
- Service water fouling in heat exchangers cooled by service water (NRC Generic Letter 89-13)

Other evaluations of service water system and components are addressed in the following LR section:

- Piping / component supports – LR section 2.2.2.2, Balance Of Plant (Non-Class 1)
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip, discharging fluids and flooding effects - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Service water instrumentation - LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems
- Environmental qualification – LR section 2.3.1, Environmental Qualification

- Safety related valve and pump testing and valve closure, including containment isolation requirements – LR section 2.2.4, Safety Related Valves and Pumps

- Protection against turbine missiles and internal missiles - LR section 2.5.1.2, Missile Protection

- Evaluation of heat exchangers cooled by service water is provided in the following:

  LR section 2.5.4.1, Spent Fuel Pool Cooling System
  LR section 2.5.4.3, Component Cooling Water System
  LR section 2.5.4.5, Auxiliary Feedwater Systems
  LR section 2.5.5.4, Condensate and Feedwater Systems
  LR section 2.6.5, Containment Heat Removal

- Service water to the auxiliary feedwater system for long term heat removal from the primary system - LR section 2.5.4.5, Auxiliary Feedwater Systems

- Post-accident heat removal requirements – LR section 2.6.1, Primary Containment Functional Design

- Reactor cooldown requirements – LR section 2.8.4.4, Residual Heat Removal System

- Control of radioactive material and the monitoring of releases - LR section 2.10.1, Occupational and Public Radiation Dose

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The service water system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report, NUREG-1786. Portions of the service water system are within the scope of License Renewal. The service water system is described in SER section 2.3.3, Auxiliary Systems. Aging effects, and the programs used to manage the aging effects associated with service water, are discussed in section 3.3 of the SER. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the service water system at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.4.2.3 Results

The following subsections evaluate the specific service water system and component licensing, design and performance capabilities while at EPU conditions.

General Design Criteria

The evaluation of the service water system capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-4. The system is protected from the dynamic effects of pipe break, including missiles, pipe whip, discharging fluids and flooding, as described in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures. Safety related equipment is environmentally qualified for the worst case environments as discussed in LR section 2.3.1, Environmental Qualification.

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

The evaluation of the service water system capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-44. The service water system provides heat removal from the reactor and transfers the heat ultimately to the environment. The service water system provides this capability under both normal operating and accident conditions and is capable of achieving this function considering a single failure. The implementation of EPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described below and by the analysis results discussed in LR section 2.6.1, Primary Containment Functional Design, LR section 2.6.5, Containment Heat Removal, and LR section 2.8.4.4, Residual Heat Removal System using the service water system during the postulated cooldown and accident scenarios.

Service Water Flow and Heat Removal from Cooled Components

The service water system supplies cooling water to the following components:

- Component cooling water heat exchangers
- Spent fuel pool heat exchangers
- Diesel generator jacket water heat exchangers
- Diesel generator lube oil cooers
- Generator bus duct coolers
- Turbine lube oil coolers
- Generator exciter cooler
- Generator hydrogen and air side seal oil coolers
- Turbine electro-hydraulic control system oil coolers
- Containment recirculation fan coolers
- Containment recirculation fan motor cooling coils

- Reactor compartment coolers
- Residual heat removal pump area coolers
- Charging pump area coolers
- Containment penetration cooler
- Standby auxiliary feedwater pump room cooling units
- Instrument air compressors
- Air conditioning water chillers
- Sample coolers and chiller
- Feedwater pumps lube oil coolers
- Safety injection pumps bearing oil coolers
- Motor driven auxiliary feedwater pumps lube oil coolers
- Turbine driven auxiliary feedwater pumps lube oil coolers
- Administrative computer room a/c unit
- Telephone equipment room a/c unit
- Condensate pumps motor bearing oil coolers
- Heater drain pumps motor bearing oil coolers
- Battery room air conditioning unit

The existing service water supply flow rates to these cooled components are capable of removing the required EPU heat loads from each component. The EPU evaluation of these cooled components is performed in the system-related LR section referenced above.

The evaluation demonstrated that, during the recirculation phase following postulated loss of coolant accidents (LOCAs), the operation of two service water pumps are required to provide sufficient flow to remove the required heat via the containment recirculation fan coolers, component cooling water heat exchangers, emergency diesel generator coolers and miscellaneous safety related pumps and fan coolers at EPU conditions. A change to the Technical Specification Bases is being made redefining the number of operable service water pumps required for a service water train to be operable from one pump to two pumps. This will ensure a minimum of 2 service water pumps are operable following a LOCA.

The majority of the cooled components are unaffected by EPU conditions since their functions and heat removal requirements are unrelated to the reactor power level or turbine cycle performance. The components significantly affected by EPU include the following:

- Component cooling water heat exchangers – primarily affected by increased reactor decay heat at the EPU power level transferred by the residual heat removal heat exchangers to the component cooling water system during normal cooldown and accidents
- Spent fuel pool heat exchangers – removes the higher fuel decay heat at the EPU power level from the spent fuel stored continuously in the spent fuel pool
- Generator bus duct coolers – generator operation at higher MWe causes added heat release to the bus duct coolers

- Containment recirculation fan coolers – during accident events, removes the additional energy released to containment due to the higher EPU power level

The higher heat loads of these cooled components with the existing service water flow rates causes their service water outlet temperatures to be higher. The worst case outlet temperature occurs when the lake water and, correspondingly, the service water supply temperature, is at its highest temperature of 85°F. The service water piping and valves at the outlets of the above components and the common discharge headers to the circulating water outlet pipes experience higher operating temperatures at EPU due to the effects of the higher outlet temperatures. These higher temperatures have been compared to the design temperatures of these portions of the service water system, and the EPU temperatures are bounded.

Since the existing service water flow rates are not affected by EPU conditions, the service water pumps capacities are acceptable for EPU operation. The existing service water operating pressures at EPU conditions are also not affected since no physical changes are being made to the service water system and the pumps continue to operate at their current discharge pressure.

The service water system also provides the required water supply to the standby auxiliary feedwater pumps at EPU conditions for long term heat removal from the primary system following abnormal scenarios when the preferred auxiliary feedwater system is unavailable. Although the standby auxiliary feedwater system flow rate is increasing by 50 gpm from 200 gpm to 250 gpm, it is insignificant when compared to the total capacity of the service water pumps. The service water system also provides a long term alternate source of water to the preferred auxiliary feedwater pumps. No changes are required to the preferred auxiliary feedwater flow requirements; therefore, the existing service water capability remains acceptable at EPU. See LR section 2.5.4.5, Auxiliary Feedwater Systems for details.

NRC Generic Letter 96-06

The implementation of EPU at the Ginna Station does not affect the previous corrective actions and responses to NRC GL 96-06 and the subsequent NRC Request for Additional Information dated April 14, 1998.

In regard to the GL issue of overpressurizing isolated portions of piping penetrating containment due to heatup from containment accident environments, relief valves were added to the current design on lines that were potentially susceptible. The EPU does not change the current design pressures and temperatures of the containment penetration piping or isolation valves. The small increase in containment post-accident temperature at EPU conditions is less than the original value of 286°F used in the analysis. Therefore, no additional piping is considered a potential concern; no new relief valves are required and the existing relief valves remain acceptable.

The GL also questioned whether the higher heat loads at accident conditions could potentially cause steam bubbles, water hammer and flow blockage due to the higher outlet temperatures from cooled components, particularly the containment recirculation fan coolers.

A detailed design analysis of this concern was originally done by the Ginna Station and accepted by NRC. The analysis considered a conservatively high containment environment of 286°F and determined that the temperatures in the piping downstream of the containment recirculation fan coolers did not adversely affect the system flow or structural integrity of the piping. This analysis was reviewed against the EPU containment environment, service water flow rates and heat removal from the containment recirculation fan coolers. The analysis assumed no changes in the cooler physical design or the service water flow rates (which are being increased due to operation of a second service water pump). The EPU containment environment is slightly lower in temperature than the original Ginna analysis of this event. Therefore, the acceptable conclusions of the original design analysis remain valid for the EPU conditions.

NRC Generic Letter 89-13

The Ginna station has a Service Water System Reliability Optimization Program, which complies with the NRC GL 89-13. The EPU does not change the flow rate through the service water systems. Accordingly, the surveillance and control techniques used to reduce the bio-fouling induced flow blockage will not require change as a result of EPU. The EPU does not change the test program to verify heat transfer capability of the safety-related heat exchangers cooled by the service water system.

Inspection and maintenance program for service water system piping and components will continue after the uprate. The uprate does not change the maintenance practices and training procedures.

The EPU does not affect the programs, procedures, and activities in place at Ginna Station in support of implementation of the requirements of GL 89-13. The program will continue to ensure that the service water system remain reliable and operable after the uprate.

2.5.4.2.4 Conclusion

The Ginna staff review has assessed the effects of the proposed EPU on the station service water system and concludes that the assessment has adequately accounted for the increased heat loads on system performance that result from the proposed EPU. The Ginna staff concludes that with the redefining of the number of operable service water pumps required for a service water train to be operable from one pump to two pumps, the station service water system will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, Ginna has determined that the service water system will continue to meet the current licensing basis with respect to the requirements of GDC-4, GDC-5, and GDC-44. Based on the above, Ginna finds the proposed EPU acceptable with respect to the station service water system.

2.5.4.3 Reactor Auxiliary Cooling Water Systems (Component Cooling Water System)

2.5.4.3.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) review covered the reactor auxiliary cooling water systems (component cooling water) that is required for safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions, or preventing the occurrence of an accident. The system includes a closed-loop cooling water system for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system. The Ginna review covered the capability of the reactor auxiliary cooling water systems (component cooling water) to provide adequate cooling water to safety-related emergency core cooling system components and reactor auxiliary equipment for all planned operating conditions. Emphasis was placed on the reactor auxiliary cooling water systems (component cooling water) for safety-related components (e.g., emergency core cooling system equipment, ventilation equipment, and reactor shutdown equipment). The review focused on the additional heat load that would result from the proposed EPU.

The NRC's acceptance criteria for the reactor auxiliary cooling water systems (component cooling water) are based on:

- GDC-4, insofar as it requires that structures, system, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents

- GDC-5, insofar as it requires that structures, system, and components important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related structures, system, and components to a heat sink under both normal operating and accident conditions be provided

Specific review criteria are contained in SRP, section 9.2.2, as supplemented by NRC Generic Letters 89-13 and 96-06.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were

published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station component cooling water system design relative to conformance to:

- GDC 4 is described in the Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases, and:.
    - Environmental Design Of Mechanical And Electrical Equipment (UFSAR section 3.11)
    - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (UFSAR section 3.6)
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)
- GDC 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, system, and Components, which states that Ginna Station is a single unit installation so there are no shared structures, system or components.
- GDC 44 is described in Ginna UFSAR section 3.1.2.4.15, "General Design Criterion 44 – Cooling water". GDC 44 addresses the provision of a system to transfer heat from structures, system, and components important to safety to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these structures, system and components under normal operating conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling and service water design features to transfer heat to the ultimate heat sink. The system provided to transfer heat from safety-related components to the ultimate heat sink (Lake Ontario) consist of the service water and the component cooling water system described in Ginna UFSAR sections 9.2.1 and 9.2.2, respectively. Component cooling water is supplied by two redundant pumps (one operating, one standby) which are supplied with power from separate emergency buses. The system is operable from offsite power or from emergency onsite power (from the diesel generators).

No single active failure results in system loss of function for those functions important to safety.

Other Ginna UFSAR sections that address the design features and functions of the component cooling water system include:

- Ginna UFSAR sections 3.2.2.1.3, Component Cooling Water Pumps and 3.2.2.5, Storage Tank Design, which describe the materials evaluation of the component cooling pumps and surge tank susceptibility to brittle fracture.

- Ginna UFSAR section 3.5.1.3.1.6, Component Cooling Water System, which discusses the potential for missiles and the missile protection of the system.

- Ginna UFSAR section 3.7.3.7.3.11, Component Cooling Water, which describes the portions of the component cooling water system included in the seismic piping upgrade program.

- Ginna UFSAR section 5.2.5.2, Leakage Limitations, which describes the design features for the detection of leakage through reactor coolant pressure boundary into the component cooling water system.

- Ginna UFSAR section 6.2.4, Containment Isolation system, which describes the design features of the component cooling water system provided for containment isolation.

- Ginna UFSAR section 6.5.1.3, NRC Generic Letter 96-06 Requirements, which addresses overpressurization of isolated piping inside containment and boiling / flow blockage / water hammer effects in service water piping to the containment recirculation fan coolers

- Ginna UFSAR section 9.2.1.2.6, Service Water Fouling, which addresses service water fouling in heat exchangers (NRC Generic Letter 89-13)

In addition to the evaluations described in the UFSAR, the component cooling water system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

Portions of the component cooling water system are within the scope of License Renewal as described in License Renewal SER, Section 2.3.3.2.

2.5.4.3.2 Technical Evaluation

2.5.4.3.2.1 Introduction

The component cooling water system is described in Ginna UFSAR section 9.2.2. The component cooling water system is designed to remove heat from plant components during plant operation, plant cooldown, and post accident conditions. Component cooling water circulates through parallel flow paths into various components, where it picks up heat from other system and transfers the heat to the service water system via component cooling water heat exchangers. The maximum design temperature of the lake is 85°F, which is the value used for the evaluation of safety related design features. A lake temperature of 80°F is used for evaluating the turbine cycle and non-safety related component performance.

There are two component cooling water pumps and two component cooling water heat exchangers. During normal full-power operation, one component water pump supplies flow to both component cooling water heat exchangers, however, one component cooling water heat exchanger can accommodate the heat removal loads. Therefore, especially at lower service water (lake) temperatures, service water may be limited or isolated to one component cooling water heat exchanger. The standby pump provides a 100% backup during MODES 1 and 2. Both pumps and both heat exchangers are utilized to remove the residual and sensible heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operative, safe operation of the plant during cooldown is not affected; however, the time to achieve shutdown is extended.

The component cooling system serves as an intermediate boundary between the radioactive fluids in the cooled components and the service water system. This arrangement reduces the possibility of radioactive fluid leakage to the environment via the service water system. Radiation monitoring is provided to detect radioactivity entering the system from any of the cooled components and the system design includes the ability to isolate any component when necessary.

2.5.4.3.2.2 Description of Analyses and Evaluation

The component cooling water system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations compared the existing design parameters of the system/components with the EPU conditions for the following design aspects:

- Component cooling water heat exchanger performance (flow rates, duty and temperatures) at the increased EPU heat loads during normal power operation, normal cooldown, and abnormal transient and accident conditions
- Component cooling water system temperature limits
- Design pressure / temperature of piping and components versus the EPU operating pressures and temperatures

- Component cooling water relief valve capacities

- Protection of isolated piping sections from heatup effects (NRC Generic Letter 96-06)

The Ginna Station commitments in regard to NRC Generic Letters 89-13 and 96-06 and the potential impact of the EPU are addressed in LR section 2.5.4.2, Service Water, in particular, service water fouling and the service water supply of cooling water to the containment fan recirculation coolers.

Other related evaluations of component cooling water system and components are addressed in the following LR section:

- Piping / component supports – LR section 2.2.2.2, Balance Of Plant Piping and Supports (Non-Class 1)

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip, discharging fluids and flooding - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures

- Component cooling water instrumentation - LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems

- Environmental qualification – LR section 2.3.1, Environmental Qualification

- Safety related valve and pump testing and valve closure, including containment isolation requirements – LR section 2.2.4, Safety Related Valves and Pumps

- Protection against turbine missiles and internal missiles - LR section 2.5.1.2, Missile Protection

- Service water fouling in heat exchangers, overpressurization of isolated piping inside containment and boiling / water hammer in service water cooling to the containment atmosphere recirculation coolers (NRC Generic Letters 89-13 and 96-06) - LR 2.5.4.2, Service Water

- Evaluation of heat exchangers cooled by component cooling water - LR section 2.1.11, Chemical and Volume Control System; LR section 2.2.2.6, Reactor Coolant Pumps and Supports; LR section 2.5.4.1, Spent Fuel Pool Cooling System; LR section 2.5.6, Waste Management Systems; LR section 2.6.5, Containment Heat Removal; LR section 2.8.4.4, Residual Heat Removal System.

- Post-accident heat removal requirements – LR section 2.6.1, Primary Containment Functional Design

- Control of radioactive material and the monitoring of releases - LR section 2.10.1, Occupational and Public Radiation Dose

2.5.4.3.3 Results

The following subsections evaluate the specific component cooling water system and component licensing, design and performance capabilities while at EPU conditions.

General Design Criteria

The evaluation of the component cooling water system capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-4. The system is protected from the dynamic effects of pipe break as described in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures. Safety related equipment is environmentally qualified for the worst case environments as discussed in LR section 2.3.1, Environmental Qualification. As described in LR section 2.5.1.3, Pipe Failures, the flooding analysis has considered the effects of moderate energy failures and evaluated the worst case failure in each plant building / area. The component cooling water system is a moderate energy system and previously analyzed failure effects are not affected by EPU conditions since the component cooling water system flow rate and pressure does not change at EPU and no physical changes are being made.

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

The evaluation of the component cooling water system capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-44. The component cooling water system provides heat removal from the reactor and transfers the heat ultimately to the environment. The component cooling water system provides this capability under both normal operating and accident conditions and is capable of achieving this function considering a single failure. The implementation of EPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described below and by the analysis results discussed in LR section 2.6.1, Primary Containment Functional Design and LR section 2.8.4.4, Residual Heat Removal System using the service water system during the postulated cooldown and accident scenarios.

Component Cooling Water Heat Removal Capability

Component cooling water is provided to the following plant equipment:

- Residual heat removal heat exchangers
- Reactor coolant pumps – bearing oil and thermal barrier
- Non-regenerative heat exchangers
- Excess letdown heat exchanger
- Seal water heat exchanger
- Boric acid evaporator distillate cooler and air ejector condenser
- Boric acid evaporator

- Sample heat exchangers
- Component cooling water system process monitor skid
- Waste gas compressor seal water heat exchanger
- Reactor support coolers
- Residual heat removal pump seal coolers & bearing water jackets
- Safety injection pump bearing seal heat exchanger
- Containment spray pump mechanical seal heat exchanger
- Post accident sampling system coolers

These cooled components are capable of removing the required EPU heat loads with the existing component cooling water supply flow rates. The EPU evaluation of these cooled components is performed in the system-related LR section referenced above.

Since none of the cooled components require more cooling flow, the existing component cooling water and service water flow rates through the component cooling water heat exchangers are not changed by the EPU.

During normal plant full power operation and normal cooldown, the component cooling water heat exchangers are capable of maintaining the cooling water supply temperature to individual cooled components below the following limits:

100°F	Normal Operation
120°F	Normal Cooldown
105°F	RCP Inlet (125°F Max for up to 2 hours)
None	Accident

At normal plant EPU full power operation, the heat loads from the cooled components are not significantly different than before EPU and; therefore, the above temperature limits are not affected.

During normal plant cooldown, the EPU heat loads are higher, primarily caused by the residual heat removal heat exchanger that has a higher duty due to the higher reactor decay heat at the EPU power level. The maximum component cooling water heat load during normal cooldown occurs when the residual heat removal system is first placed in service, four hours after reactor shutdown. During cooldown, the reactor coolant flow through the residual heat removal heat exchangers is throttled to limit cooldown of the reactor coolant system to 50°F per hr and to limit the component cooling water heat exchanger outlet temperature to 120°F. As a result of maintaining these limits with the higher EPU heat loads, the normal cooldown is lengthened as described in LR section 2.8.4.4, Residual Heat Removal System.

During accident conditions, the component cooling water heat exchangers remove heat from the containment sump and reactor coolant system via the residual heat removal heat exchangers. Similar to normal cooldown, the accident heat loads at EPU conditions are

higher due to the higher reactor decay heat at the EPU power level. The EPU analyses described in LR section 2.6.1, Primary Containment Functional Design, and LR section 2.6.5, Containment Heat Removal, confirm that the component cooling water heat exchangers provide sufficient heat removal for mitigation of postulated accidents.

The evaluation also confirmed that, during the recirculation mode following postulated loss of coolant accidents, the operation of two service water pumps in conjunction with two CCW heat exchangers and one CCW pump, supplies sufficient flow to remove the required heat via the containment recirculation fan coolers, component cooling water heat exchanger and emergency diesel generator coolers, including the safety related pump coolers and the containment recirculation fan motor coolers which are in operation at that time.

The EPU heat loads, service water inlet temperatures and the component cooling water outlet temperatures for normal cooldown and accident conditions are shown below. The associated CCW system volume increase associated with these increased temperatures is equivalent to less than 300 gallons which is well within the current available 1000 gallon surge volume in the CCW Surge Tank.

Table 2.5.4.3-1

Component Cooling Water System EPU Conditions

Service Water Temperature (°F)	Heat Load Per CCW HX (MBTU/hr)	CCW HX CCW Inlet Temperature (°F)	CCW HX Service Water Outlet Temperature (°F)	RHR HX CCW Outlet Temperature (°F)
Operation During Normal Cooldown (Note 1)				
85	44.9	156.8	103.0	169.6
Operation During Worst Case Accident Condition				
85	20.25	150.7	101.3	150.7

Note 1: The cooldown heat load to the CCW system from the RHR heat exchangers will be administratively controlled to limit the CCW outlet temperature to a maximum of 170°F.

EPU Operating Conditions versus Design Conditions of Piping and Components

The component cooling water system flow rate does not change at the EPU conditions and no physical changes are being made to the system. Therefore, the component cooling water system operating pressures are not affected by EPU conditions and the existing component design pressures are acceptable.

The existing component cooling water piping to / from the reactor coolant pump thermal barrier is designed for the reactor coolant system pressure and temperature in event of a failure of the thermal barrier. The reactor coolant system design conditions do not change due to EPU; therefore, the design pressure and temperature of this portion of the component cooling water system is acceptable for EPU operation.

The higher heat loads at normal cooldown and accident conditions cause the component cooling water outlet temperatures from cooled components to be higher which, in turn, causes the inlet temperature to the component cooling water heat exchanger to be higher.

The design temperature of the component cooling water heat exchangers, pumps and surge tank is 200°F, the piping design temperature is 500°F and the design temperature of the piping to / from the reactor coolant pump thermal barrier is 650°F. All of these design temperatures bound the maximum component cooling water operating temperatures at EPU conditions.

Component Cooling Water Relief Valve Capacities

The component cooling water system relief valves have either no change or small changes in temperatures that are bounded by the relief valve design. Because the EPU condition is below the system design temperature/pressure, no additional analysis is required to demonstrate their acceptability.

The postulated flow, pressure and temperature from a failure of the RCP thermal barrier does not change for EPU operation since there are no changes to the existing reactor coolant system design conditions and no changes are being made to the reactor coolant pump thermal barrier. Therefore, the relief valves on the component cooling water piping at the reactor coolant pump thermal barrier and on the component cooling water surge tank are unaffected by EPU conditions.

NRC Generic Letters 89-13 and 96-06

The issues in these Generic Letters are service water fouling in heat exchangers, heatup and overpressurization of isolated portions of piping inside containment, and boiling / water hammer in service water cooling lines to the containment atmosphere recirculation coolers. The potential impact of the EPU on the Ginna Station responses to NRC Generic Letters 89-13 and 96-06 and subsequent NRC Requests for Additional Information is addressed in LR section 2.5.4.2, Service Water.

The issue in NRC Generic Letter 96-06 related to the heatup / overpressurization of isolated component cooling water piping inside containment was evaluated by Ginna Station in their previous responses and there were no concerns identified in the component cooling water piping inside containment. This conclusion is not affected by EPU conditions since there are no physical changes or operational changes required by EPU that would affect the containment penetration piping or isolation valves. The small increase in post-accident temperature at EPU conditions raises the containment temperature to less than the value of 286°F used in the original analysis. Therefore, no additional lines from the CCW system that penetrate the containment are considered a potential concern; no new relief valves are required and the existing relief valves remain acceptable.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the component cooling water system are within the scope of License Renewal as described in License Renewal Safety Evaluation Report, NUREG-1786, Section 2.3.3.2. Aging effects, and the programs used to manage the aging effects associated with component cooling water, are discussed in SER Section 3.3.2.4.2, component cooling water. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Because no modifications are necessary for the component cooling water system, the EPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.4.3.4 Conclusion

The Ginna staff has assessed the effects of the proposed EPU on the component cooling water system and concludes the assessment has adequately accounted for the effect of the increased heat loads from the proposed EPU on system performance. The Ginna staff concludes that the component cooling water system will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for structures, systems and components important to safety following implementation of the proposed EPU. Therefore, the Ginna staff concludes that the component cooling water system will continue to meet the current licensing basis with respect to the requirements of GDC-4, GDC-5, and GDC-44. Based on the above, the Ginna staff finds the proposed EPU acceptable with respect to the component cooling water..

2.5.4.4 Ultimate Heat Sink

2.5.4.4.1 Regulatory Evaluation

The ultimate heat sink is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The Ginna Nuclear Power Plant, LLC (Ginna) review focused on the impact that the proposed EPU has on the decay heat removal capability of the ultimate heat sink. Additionally, the Ginna review included evaluation of the design-basis ultimate heat sink temperature limit to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed.

The NRC's acceptance criteria for the ultimate heat sink are based on:

- GDC-5, insofar as it requires that structures, systems, and components important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from important-to-safety structures, systems, and components to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

Specific review criteria are contained in SRP section 9.2.5.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically the adequacy of Ginna Station ultimate heat sink design relative to conformance to:

- GDC 5 is addressed in Ginna UFSAR section 3.1.2.1.5. This GDC addresses shared structures, systems or components and is not applicable since Ginna Station is a single unit installation.

- GDC 44 is addressed in Ginna UFSAR section 3.1.2.4.15, "General Design Criterion 44 – Cooling Water". GDC – 44 addresses provision of a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions. Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling water and (CCW) service water design features to transfer heat to the ultimate heat sink.

The systems provided to transfer heat from the safety-related components to the ultimate heat sink (Lake Ontario) are the service water and the component cooling water systems.

Ginna UFSAR sections that address the design features and functions of the ultimate heat sink include:

- Ginna UFSAR section 1.8.1.27, Safety Guide 27 – Ultimate Heat Sink, which describes the design features of the intake structure and screenhouse in providing a reliable source of water, regardless of weather or lake conditions, from Lake Ontario to the suction of the condenser circulating water pumps, service water pumps and fire water pumps.

- Ginna UFSAR section 2.4.1, Hydrologic Description, which describes the hydrology of the site, including the size and water level of Lake Ontario

- Ginna UFSAR section 2.4.6, Cooling Water Canals and Reservoirs, which describes the general arrangement of the intake structure, screenhouse, circulating water pumps and and discharge canals.

- Ginna UFSAR section 2.4.8, Low Water Considerations, which describes the historical data regarding Lake Ontario water levels

- Ginna UFSAR section 8.1.4.4 Potential Risk of Station Blackout, which describes Lake Ontario as the inexhaustible source of secondary cooling water to the steam generators.

- Ginna UFSAR section 9.2.1, Service Water, which describes the service water system and its use of Lake Ontario as a water source.

- Ginna UFSAR section 9.2.2, Component Cooling Water, which describes the component cooling water system design and its rejection of heat to the service water system.

- Ginna UFSAR section 9.5.1, Fire Protection Systems, which describes the use of Lake Ontario as a water source.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's ultimate heat sink was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, the ultimate heat sink is described in SER sections 2.3.3.5, Service Water and 2.4.2.7, Screenhouse

2.5.4.4.2 Technical Evaluation

2.5.4.4.2.1 Introduction

The ultimate heat sink is described in the Ginna UFSAR sections listed above. The ultimate heat sink is Lake Ontario, which provides water to the service water system, via the intake structure or the discharge canal alternate source. The service water system provides cooling water for heat removal from safety-related heat exchangers, including the CCW heat exchangers and Residual Heat Removal (RHR) heat exchangers, and supplies water from the ultimate heat sink to the standby auxiliary feedwater system and preferred auxiliary feedwater system for emergency heat removal from the reactor coolant system. See LR section 2.5.4.2, Service Water for a description of the cooled components and the water users supplied by the service water system. See LR section 2.5.4.5, Auxiliary Feedwater for a description of the use of service water from the ultimate heat sink as a source of water for auxiliary feedwater requirements.

A maximum lake water temperature of 85°F is used for the safety related analyses which rely on the ultimate heat sink for heat removal. See LR section 2.8.4.4, Residual Heat Removal System and LR section 2.6.1, Primary Containment Design which describe the cooldown and postulated accident scenarios using the ultimate heat sink for heat rejection.

Lake Ontario is also used by the non-safety related circulating water system to provide cooling water for heat removal from the turbine cycle during normal plant power operations. See LR section 2.5.8.1, Circulating Water.

2.5.4.4.2.2 Description of Analyses and Evaluations

The ultimate heat sink was evaluated to ensure it is capable of performing its intended function of supplying a reliable water source and heat removal capacity for normal and accident conditions following EPU.

The ultimate heat sink was evaluated for the circulating water discharge temperature during EPU normal power operation and initial normal cooldown, including the effect of the service water discharge temperatures during normal power operation and cooldown. These effects were evaluated against the New York State SPDES permit limits for the Ginna Station in LR section 2.5.8.1, Circulating Water System.

Evaluations related to the ultimate heat sink are addressed in the following LR section:

- Circulating water discharge temperatures and flow rates to Lake Ontario and the SPDES permit limits - LR section 2.5.8.1, Circulating Water System

- Service water discharge temperatures, heat loads and flow rates of the circulating water discharge to Lake Ontario - LR section 2.5.4.2, Service Water System

- Component cooling water temperatures, heat loads and flow rates to the service water system and to the ultimate heat sink - LR section 2.5.4.3, Component Cooling Water System

- Flow requirements for service water supplied to the auxiliary feedwater system from the ultimate heat sink - LR section 2.5.4.5, Auxiliary Feedwater Systems

- Post-accident heat removal requirements - LR section 2.6.1, Primary Containment Functional Design

- Reactor cooldown requirements - LR section 2.8.4.4, Residual Heat Removal System

- Evaluation of the environmental impact (e.g., humidity, wind speed, etc.) - LR section 2.14, Impact of EPU on License Renewal

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the ultimate heat sink are within the scope of License Renewal are described in the License Renewal Safety Evaluation Report, NUREG-1786, Section 2.3.3.5, Service Water, and Section 2.4.2.7, Screenhouse. Aging effects, and the programs credited with managing the aging effects associated with ultimate heat sink, are discussed in SER Table 3.3-1, Staff Evaluation Table for Ginna Auxiliary System Components Evaluated in the GALL Report, and Section 3.3.2.1, Aging Management Evaluations in the GALL Report That Are Relied on for License Renewal, Which Do Not Require Further Evaluation. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with the operation of the ultimate heat sink at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified. The aging management program is consistent with the Generic Aging Lessons Learned Report.

2.5.4.4.3 Results

The ultimate heat sink continues to meet its licensing, design and performance capabilities at EPU conditions as evidenced by the evaluation results described below.

General Design Criteria

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

The evaluation of the ultimate heat sink capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in Ginna UFSAR section 3.1.2.4.15. The ultimate heat sink provides for heat removal from the reactor and transfers the heat ultimately to the environment. The service water system provides cooling water from the ultimate heat sink (Lake Ontario), under both normal operating and accident conditions, and is capable of achieving this function considering a single failure. The implementation of EPU does not affect the capability of the ultimate heat sink to perform this function as demonstrated by the system and component evaluation results described below and in LR section 2.5.4.2, Service Water and LR section 2.6.1, Primary Containment Functional Design.

Water Supply and Heat Removal Requirements

The ultimate heat sink will continue to provide the required water supply and heat sink capacity at EPU conditions. The service water flow requirements for cooling of safety related heat exchangers are not changed by EPU. The analysis results discussed in LR section 2.8.4.4, Residual Heat Removal System and LR section 2.6.1, Primary Containment Functional Design confirm that the ultimate heat sink will continue to provide sufficient water supply and heat removal for cooldown and to mitigate the postulated accident scenarios.

The service water system provides the flow from the ultimate heat sink to the standby auxiliary feedwater system and is an alternate source for the preferred auxiliary feedwater system following a seismic event and the depletion of the condensate storage tank. See LR section 2.5.4.5, Auxiliary Feedwater for details. The standby auxiliary feedwater flow requirement has increased at EPU. The current preferred auxiliary feedwater flow requirements are not affected by EPU. The minor increase in the service water flow required by the standby auxiliary feedwater system is insignificant in regard to the design of the ultimate heat sink, service water pumps and the intake structure (screenhouse) used by the service water pumps.

The service water returned to the ultimate heat sink from cooled components following EPU is at a slightly higher temperature due to the higher heat loads from the EPU NSSS thermal power level at normal operating conditions and, during cooldown, from the higher reactor decay heat. The temperature of the circulating water discharge to Lake Ontario is higher during normal plant operation due to the higher turbine cycle heat rejection and during the initial stages of plant cooldown due to the EPU reactor power level. Service water flows into the circulating water outlet piping prior to its discharge to Lake Ontario; however, the effect of the higher EPU service water temperatures during cooldown and accident conditions is minimal due to the significantly larger circulating water flow rate.

The heat sink capacity of Lake Ontario is easily able to absorb the added heat from circulating water and service water with negligible effect on the lake temperature. See LR section 2.5.4.2, Service Water and LR section 2.5.8.1, Circulating Water System for discharge flows, heat loads and temperatures.

The discharge of circulating water is governed by the New York SPDES permit which limits its discharge temperature. As discussed in LR section 2.5.8.1, Circulating Water System, the higher circulating water temperature during normal plant power generation is above the current SPDES limit when operating near the design circulating water inlet temperature. The Ginna Station has requested an increase to the SPDES permit discharge temperature to accommodate the calculated EPU discharge temperature, with margin. Refer to the Supplemental Environmental Impact (Attachment 8 of the Power Uprate LAR) for additional discussion.

2.5.4.4.4 Conclusions

The Ginna staff assessment has adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the ultimate heat sink to supply adequate water to the steam generators to ensure adequate cooling of the core. The Ginna staff concludes that the ultimate heat sink will continue to provide its design functions following implementation of the proposed EPU. The Ginna staff further concludes that the ultimate heat sink will continue to meet the current licensing basis with respect to the requirements of GDC-44. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to the ultimate heat sink.

2.5.4.5 Auxiliary Feedwater Systems

2.5.4.5.1 Regulatory Evaluation

In conjunction with a seismic Category I water source, the auxiliary feedwater system functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. The auxiliary feedwater (AFW) system is also used to provide decay heat removal capability necessary for withstanding or coping with a station blackout. The Ginna Nuclear Power Plant, LLC (Ginna) review of the proposed EPU focused on the system's continued ability to provide sufficient emergency feedwater flow at the expected conditions (e.g., steam generator pressure) to ensure adequate cooling with the increased decay heat. The Ginna review also considered the effects of the proposed EPU on the likelihood of creating fluid flow instabilities (e.g., water hammer) during normal plant operation, as well as during upset or accident conditions.

The NRC's acceptance criteria for the auxiliary feedwater system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures.

- GDC-5, insofar as it requires that structures, systems, and components important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

- GDC-19, insofar as it requires that equipment at appropriate locations outside the control room be provided with (a) the capability for prompt hot shutdown of the reactor, and (b) a potential capability for subsequent cold shutdown of the reactor.

- GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from important-to-safety structures, systems, and components to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

Specific review criteria are contained in SRP Section 10.4.9.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna auxiliary feedwater system design relative to conformance to:

- GDC – 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this Ginna UFSAR section, Ginna Station received post-construction review as part of the SEP. The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC-4 is described in the following:
  - Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
  - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
    - Pipe Breaks Inside Containment (SEP Topic III-5.A)
    - Pipe Breaks Outside Containment (SEP Topic III-5.B)
  - Missile Protection (Ginna UFSAR section 3.5)
- GDC – 5 is described in Ginna UFSAR section 3.1.2.1.5. This GDC addresses shared structures, systems or components and is not applicable since Ginna Station is a single unit installation.
- GDC – 19 is described in Ginna UFSAR section 3.1.2.2.10. This GDC addresses control room habitability and requires that the equipment outside the control room be provided with a design capability for prompt shutdown of the reactor through the use of suitable procedures. As described in UFSAR section 3.1.2.2.10, the station is equipped with a control room that contains controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions. The control room is capable of continuous occupancy under all operating and accident conditions, within

specified dose limits. Although the likelihood of the main control room being inaccessible is extremely small, plant operators can shut down and maintain the plant in a safe shutdown condition by means of controls located outside the control room.

- GDC – 34 is described in Ginna UFSAR section 3.1.2.4.5 that addresses the ability of the residual heat removal (RHR) system, in conjunction with the steam power conversion system, to transfer fission product decay heat and other residual heat from the reactor core. Section 5.4.5 describes the RHR system and its heat removal capabilities. System isolation provisions are described in UFSAR section 5.4.5.3. The AFW system is part of the steam power conversion system.

- GDC – 44 is described in Ginna UFSAR section 3.1.2.4.15, "General Design Criterion 44 – Cooling Water". GDC – 44 addresses provision of a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating conditions. Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling and service water design features to transfer heat to the ultimate heat sink. Note that the AFW systems are not specifically addressed; however, the AFW and standby AFW systems do provide the heat removal design functions considered under this GDC.

Other Ginna UFSAR sections that address the design features and functions of the AFW system include:

- Ginna UFSAR section 3.7.3.7, Seismic Piping Upgrade Program, which upgraded certain Seismic Category I Piping systems to more current requirements and provided a seismic data base for use with modifications, the inservice inspection program, and NRC requests for information. The upgrade program included portions of piping to / from the AFW and standby AFW pumps.

- Ginna UFSAR section 6.2.1.2.3, Secondary System Pipe Break Analysis, which describes the operation and failure modes of the AFW system during postulated steam line breaks.

- Ginna UFSAR section 6.2.4, Containment Isolation System, which describes the containment isolation features to isolate the containment boundaries in the AFW and standby AFW systems post-accident.

- Ginna UFSAR sections 8.1.4.4, Potential Risk of Station Blackout, and 8.1.4.5 Station Blackout Program, which describe the operation of the AFW system during a station blackout.

- Ginna UFSAR section 9.5.1.4, Safe Shutdown Capability, which describes the operation of the AFW system during Appendix R fire scenarios.

- Ginna UFSAR section 10.5, Auxiliary Feedwater Systems, which describes the AFW design basis, system operation and instrumentation.

The AFW and standby AFW systems are also credited in the mitigation of the transient and accident analyses described in the UFSAR Chapter 15 analyses.

In addition to the evaluations described in the Ginna UFSAR, the Ginna's AFW system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

The above SER discusses the AFW system in section 2.3.4.3, Auxiliary Feedwater.

2.5.4.5.2 Technical Evaluation

2.5.4.5.2.1 Introduction

The AFW system is described in Ginna UFSAR section 10.5. The system consists of a preferred AFW system (2 motor driven and a turbine driven pump) and a standby AFW system (2 motor driven pumps). The preferred AFW system and standby AFW system are required to operate to provide water to maintain the steam generators water level and remove reactor heat when the normal feedwater system is not available. During accident conditions, either the preferred AFW system or standby AFW system supply feedwater to the steam generators to provide emergency heat removal from the RCS using secondary heat removal capability (atmosphere or main condenser).

The AFW systems are engineered safety features because they provide a secondary heat sink for residual heat removal and; therefore, provide core protection and prevention of reactor coolant release through the pressurizer safety valves. The plant conditions that impose performance requirements on the design of the AFW systems are as follows:

a. Loss of main feedwater transient
    1. Loss of main feedwater with offsite power available
    2. Loss of main feedwater without offsite power available
    3. Rupture of feedwater line
b. Rupture of a main steam line
c. Loss of all ac power (offsite and onsite)
d. Loss-of-coolant accident
e. Cooldown under normal and abnormal conditions, including Appendix R fire, high energy line break and station blackout.
f. Anticipated transients without scram (ATWS)

The preferred AFW system consists of two motor driven pumps and one turbine driven pump. Normally each motor driven pump supplies one steam generator, but alignment can be altered to allow either motor driven pump to supply either or both steam generators. The turbine driven pump normally supplies feedwater to both steam generators. Each pump supplies the steam generators through a normally open, motor operated discharge valve. The AFW system operates during startup, normal cooldown and emergency cooldown following abnormal and accident conditions. The AFW system is located entirely outside containment. That is, the discharge headers tie in to the main feedwater headers in the intermediate building with the feedwater headers continuing on to each steam generator. The check valves and manual test connection valve outside containment on each AFW line designated in Ginna UFSAR Table 6.2-15a and Figure 6.2-78, are part of the containment isolation boundary.

The standby AFW system consists of two motor driven pumps located in a plant area separate from the AFW system. The standby AFW system is manually actuated and aligned so that each pump supplies one steam generator, though a cross-connect line is also available. The standby AFW system is required to operate when the AFW system has been rendered inoperable by a high-energy line break, fire or other abnormal event. The standby AFW system penetrates the containment since the standby AFW discharge headers connect the main feedwater headers to each steam generator inside containment. The motor operated valve and manual test connection valve outside containment on each standby AFW line, designated in Ginna UFSAR Table 6.2-15a and Figure 6.2-26, are part of the containment isolation boundary.

The primary impact of the EPU on the AFW system and standby AFW system is the increased core thermal power and the resulting higher heat removal requirements during abnormal, transient and accident conditions.

2.5.4.5.2.2 Description of Analyses and Evaluations

The AFW and standby AFW systems and associated components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations compared the existing design parameters of the systems / components with the EPU conditions in conjunction with the following design aspects:

- Required flow rates / pump capabilities
- Design versus operating pressure / temperature of piping and components
- Flow velocities / erosion – corrosion concerns
- Water supplies / sources
- Pump design and performance

Other evaluations of the AFW system, piping and components are addressed in the following LR section:

- Piping / component supports and water hammer effects – LR section 2.2.2.2, Balance Of Plant Piping and Supports (Non-Class 1)
- Steam supply to the turbine driven AFW pump - LR section 2.5.5.1, Main Steam

- Operation of the AFW and standby AFW systems during postulated abnormal and accident scenarios is discussed in LR section 2.8.5, Accident and Transient Analyses
- Operation of the AFW and standby AFW systems during station blackout and Appendix R fire scenarios is discussed in LR section 2.3.5, Station Blackout and LR section 2.5.1.4, Fire Protection, respectively
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Auxiliary and standby auxiliary feedwater instrumentation - LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems
- Environmental qualification of pumps and valves – LR section 2.3.1, Environmental Qualification
- Safety related valve and pump testing and valve closure, including containment isolation requirements – LR section 2.2.4, Safety Related Valves and Pumps
- Protection against turbine missiles and internal missiles is discussed in LR section 2.5.1.2, Missile Protection
- Operation to mitigate anticipated transients - LR section 2.8.5.7, Anticipated Transients Without Scram (ATWS)

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The AFW and standby AFW systems are within the scope of License Renewal as described in the License Renewal Safety Evaluation Report, NUREG-1786, Section 2.3.4.3. Aging effects, and the programs used to manage the aging effects associated with AFW, are discussed in section 3.4.2.4.3, Auxiliary Feedwater System. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the AFW system at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.4.5.2.3 Results

The following subsections evaluate the specific AFW and standby AFW system and component licensing, design and performance capabilities while at EPU conditions.

General Design Criteria

The evaluation of the AFW and standby AFW systems capabilities at EPU conditions demonstrates that Ginna will continue to meet the current licensing basis with respect to GDC-19, described in Ginna UFSAR section 3.1.2.2.10, which states that, although the likelihood of conditions which could render the main control room inaccessible even for a short time is extremely small, provisions have been made so that plant operators can shut down and

maintain the plant in a safe condition following implementation of the EPU by means of controls located outside the control room. The AFW and standby AFW systems will be operated during such a period of control room inaccessibility to maintain the plant in a safe shutdown condition. The implementation of EPU does not affect the capability of these systems to perform this function.

The evaluation of the AFW and standby AFW systems capabilities at EPU conditions demonstrates that Ginna will continue to meet the current licensing basis with respect to the requirements of GDC-34, described in Ginna UFSAR section 3.1.2.4.5. The AFW system is considered one of the steam power conversion systems, which, together with the residual heat removal system, transfer the heat from the reactor core at a rate such that design limits of the fuel and the primary system coolant boundary are not exceeded. Suitable redundancy is provided in the AFW and standby AFW pumps, piping paths and valves. The AFW and standby AFW systems are able to operate with either onsite or offsite power systems. The AFW and standby AFW systems will continue to provide these same capabilities after implementation of EPU as demonstrated by the system and component evaluation results described below and by the analysis results discussed in LR section 2.8.5, Accident and Transient Analyses, using the auxiliary feedwater systems to mitigate the postulated abnormal and accident scenarios.

The evaluation of the AFW and standby AFW systems capabilities at EPU conditions demonstrates that Ginna will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in Ginna UFSAR section 3.1.2.4.15. Although the AFW systems are not specifically addressed by this GDC, the AFW system does provide for heat removal from the reactor and transfers the heat ultimately to the environment. The AFW and standby AFW systems provide this capability under both normal operating and accident conditions and are capable of achieving this function considering a single failure. The implementation of EPU does not affect the capability of these systems to perform this function as demonstrated by the system and component evaluation results described below and by the analysis results discussed in LR section 2.8.5, Accident and Transient Analyses, using the AFW systems to mitigate the postulated abnormal and accident scenarios.

The evaluation of the AFW and standby AFW systems capabilities at EPU conditions demonstrates that Ginna will continue to meet the current licensing basis with respect to the requirements of GDC-4, as described in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures.

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

AFW Flow Rates / AFW and Standby AFW Pumps Design Capabilities

The changes to preferred AFW and standby AFW system flow rate requirements resulting from EPU are described in the table below.

Table 2.5.4.5-1
Auxiliary Feedwater Flow Rate Requirements

Parameter	Current Flow Rate GPM	EPU Flow Rate GPM
Preferred Auxiliary Feedwater		
Feedwater Line Break (to intact S/G)	200	≥195 within 1 minute(1)
Steam Line Break (to intact S/G)	≤230	≤235
Standby Auxiliary Feedwater		
Feedwater Line Break	200 within 10 minutes	≥235 within 14.5 minutes(2)
Appendix R to each SG (water solid operation)	225	250

1. Reduction in required flow due to more detailed analyses.
2. Flow increased to account for increased decay heat. The time was increased to allow additional margin for placing the system in service.

With the exception of the above, the current preferred and standby AFW system flows for other events such as loss of normal feedwater (LONF), station blackout (SBO), ATWS, and normal cooldown remain bounding.

The increased flow rates of the preferred AFW pumps (motor driven or turbine driven AFW pumps), are within the existing pump design capabilities; therefore, no changes are required to the preferred AFW system pumps or valves.

The maximum flow required from a standby AFW pump has increased due to EPU conditions from the current value of 225 gpm to 250 gpm during the Appendix R fire (water solid operation) and from 200 gpm to 235 gpm during a feed line break affecting the other AFW pumps. The existing standby AFW pumps are capable of providing this additional flow due to the flow margin provided in the original pump design. At the higher flow rates, the pump developed head is slightly lower than the pump design point at 200 gpm. The system pressure drop at the increased flow rate and the steam generator pressure during each scenario has been evaluated against the standby AFW pump developed head. The results indicate that it is necessary to modify the trim internals of the standby AFW pump discharge flow control valves prior to EPU.

Design vs. Operating Pressures and Temperatures – Preferred AFW

Table 2.5.4.5-2 below provides a comparison of AFW operating pressures and temperatures versus the design pressures and temperatures of the piping system.

Table 2.5.4.5–2
Auxiliary Feedwater System Pressure / Temperature Comparison

	Design Conditions		Maximum EPU Conditions	
	Pressure, psig	Temperature, °F	Pressure, psig	Temperature, °F
Suction Piping	150	150	130	104
Discharge Piping	2095	150	1755	104

The AFW system maximum pump suction pressure under EPU operation occurs when the service water pumps are aligned to the supply water to the AFW pumps (service water is an alternate water source for operation of the AFW system). Since there are no changes to the existing service water pumps or piping due to EPU, the maximum pressure (i.e., pump shut off head) at the AFW system pump suction pressure is conservatively determined to be 130 psig at cold water conditions.

The maximum AFW pump discharge pressure under EPU conditions is 1755 psig based on the shutoff head of the turbine-driven AFW pumps plus the maximum pump suction pressure of 130 psig. Worst case temperature is based on maximum temperature of the water sources (i.e., condensate storage tank or lake water). The condensate storage tank maximum water temperature of 104°F is the worst case as it is higher than the maximum lake water temperature (85°F). Note that the condensate storage tank temperature of 104°F has been increased from the 100°F used in the original plant design basis. This change was made to be consistent with the maximum building environmental temperature of 104°F. The piping design temperature envelopes this slight change.

As shown in Table 2.5.4.5-2, the design conditions, pressure and temperature, of the AFW suction and discharge piping envelope the EPU operating conditions.

Design vs. Operating Pressures and Temperatures - Standby Auxiliary Feedwater System

Table 2.5.4.5-3 below provides a comparison of standby AFW operating pressures and temperatures versus the design pressures and temperatures of the system.

As shown on the table, the design conditions, pressure and temperature, of the standby AFW pump suction and discharge piping envelope the EPU operating conditions.

Table 2.5.4.5–3
Standby Auxiliary Feedwater System Pressure / Temperature Comparison

	Design Conditions		Maximum EPU Conditions	
	Pressure, psig	Temperature, °F	Pressure, psig	Temperature, °F
Suction Piping	225	150	130	85
Discharge Piping	1685	200	1505	85

The maximum standby AFW pump suction pressure under EPU operation occurs when the service water pumps are aligned to supply water to the standby AFW pumps. As stated above, the pressure due to the service water pump shut off head is 130 psig.

The maximum pressure in the standby AFW pump discharge piping under EPU operation is based on the shutoff head of the standby AFW pumps (1375 psig) plus the discharge pressure of the service water pumps (130 psig) when aligned to the standby AFW pump suction. The total discharge pressure would therefore be 1505 psig. The temperature in the standby AFW pump suction and discharge piping during EPU conditions is based on the temperature of its source of water (i.e., lake water) that ranges from 32°F to 85°F. This temperature range is not affected by EPU.

Flow Velocities and Erosion / Corrosion Concerns -Preferred AFW System

The required preferred AFW system flow rate changes are within the original design parameters. Therefore, the maximum velocity in the piping system remains the same and, there is no greater potential for erosion / corrosion due to EPU conditions.

Flow Velocities and Erosion / Corrosion Concerns - Standby AFW System

The maximum standby AFW pump flow requirement during EPU operation has increased to 250 gpm from the current pump design value of 200 gpm. The velocity at this operating condition increases correspondingly; however, it is not expected to increase the potential for erosion / corrosion. The standby AFW system operates rarely (i.e., only on loss of AFW pumps and during testing). Additionally, 250 gpm is a maximum flow rate, which decreases in a short period of time as the RCS heat removal requirements decrease. Since the frequency and duration of the increased flow rate / velocity in the standby AFW system is extremely short over the life of the plant, the potential for increased erosion / corrosion is negligible.

AFW and Standby AFW Pumps - Design Pressures / Temperatures

The design pressures / temperatures of the existing AFW pumps and standby AFW pumps are acceptable for operation under EPU conditions. There are no physical changes to the existing AFW and standby AFW pumps or to the service water pumps that supply water at pressure to the pumps' suctions. Therefore, there is no change to worst case operating pressure under EPU conditions and the existing manufacturer's design pressure ratings for the AFW and standby AFW pumps remain bounding. Similarly, the slight 4°F increase in the temperature of the condensate storage tank water source does not affect the manufacturer's pump design ratings.

AFW and Standby AFW Valves and In-Line Components - Design Pressures / Temperatures

Valves and in-line piping components were generally purchased in accordance with the pressure / temperature ratings of the applicable design codes (e.g., ANSI B16.5). These design code pressure / temperature ratings were selected consistent with the piping design pressure / temperature in which the components are installed (e.g., 900 lb class for discharge piping and 150 lb class for suction piping). Uniquely designed components (i.e., those not governed by

design code ratings) have been evaluated by reviewing the manufacturer's design documentation.

Based on the above evaluations, the valves and in line components in the AFW and standby AFW systems have design pressures / temperatures which bound the maximum EPU operating pressure / temperature conditions.

AFW System Water Supply - Condensate Storage Tank

Two condensate storage tanks are used as a source of water for AFW operation to provide the Technical Specification minimum required usable volume. This minimum useable volume for EPU operation is an inventory of 24,350 gallons to meet the plant licensing basis of decay heat removal for 2 hours after a reactor trip from full power. The current operating Technical Specification volume is 22,500 gallons.

Two tanks, normally operated cross-connected, provide the preferred source of AFW. Technical Specifications stipulate the minimum required volume as stated above. For operational flexibility, Ginna maintains the required Technical Specification volume in one tank whenever possible.

The evaluation of the condensate storage tank was done with the intent of maintaining this operational flexibility. Therefore, one tank was considered, including the unusable height of water over the tank outlet needed to prevent vortexing and the level instrument inaccuracies. To increase the available margin, the EPU requirement of 24,350 gallons with one condensate storage tank, the condensate storage overflow piping will be modified to accommodate a higher level. Plant technical specifications are being changed to incorporate the new volume requirement. The license amendment request for EPU includes this change for NRC review.

Alternate Water Supplies – Service Water & Fire Protection Water

The service water system and yard fire protection system provide alternate sources of water for operation of the AFW and standby AFW pumps. The source of water for the service water pumps is Lake Ontario. The yard fire protection system receives a supply of water from the Town of Ontario water system.

The standby AFW system pump flow requirements increase during a feed line break or Appendix R fire (water solid operation). This increase is within the existing standby pump design capacity and is small with respect to the service water system flow. Therefore, the flow required from the service water system is essentially unchanged, and the SW pumps are capable of providing sufficient flow during EPU operation.

The yard fire protection system also provides an alternative supply of water to the standby AFW pumps. Although the standby AFW pumps maximum flow requirement has increased to 250 gpm during an Appendix R fire (water solid operation), this maximum flow can be provided by the yard fire protection system. See LR section 2.5.1.4, Fire Protection, for details.

AFW Pump - Net Positive Suction Head (NPSH)

Since there are no physical changes to the condensate storage tank AFW suction connections, AFW suction piping, AFW pumps and the worst case flow rate has not changed, the NPSH required by the AFW pumps remains the same pre- and post-EPU. The minimum available NPSH occurs when the condensate storage tank is at its lowest level without vortexing. The increase in the condensate storage tank level to meet the new EPU Technical Specification volume has no impact on available NPSH since only the uppermost level setpoints are raised by the modifications at the top of the tank. The maximum flow in the AFW suction piping has not increased; therefore, the dynamic losses are unchanged. The slight increase in condensate storage tank maximum temperature, 100°F to 104°F, described above, is insignificant in regards to NPSH available. Therefore, the motor driven AFW pumps and turbine driven AFW pumps will continue to have sufficient NPSH for proper operation.

Net Positive Suction Head (NPSH) - Standby AFW Pumps

The flow required from the standby AFW pumps increases at EPU conditions to 250 gpm during an Appendix R fire (water solid operation). The manufacturer's performance curve shows that the NPSH required for the existing standby AFW pumps is slightly higher at 250 gpm (21 feet) versus the NPSH required of 20 feet for the existing pump design of 200 gpm. Based on an evaluation of the suction piping conditions when supplied from the service water pumps, the available NPSH is sufficient for pump operation at 250 gpm. Similarly, the suction pipe conditions with the fire water system as the source of water provide sufficient available NPSH for the standby AFW pump to operate at the reduced flow rates at that time.

Pump Brake HP Requirements - Auxiliary Feedwater Pumps

The brake HP required by the motor-driven AFW pumps is unaffected by EPU conditions since there are no changes to the existing pump design and the maximum operating flow rate and discharge head are the same or less than the current operating maximums. Therefore, there is no additional HP demand on the existing motors of the AFW pumps due to EPU operation.

Note that the original AFW pump motors were 250 HP. However, the motors have been replaced with 300 HP motors as part of normal plant maintenance. Although this change is not related to EPU, it is considered an enhancement of the pump / motor capability.

The maximum EPU flow requirement for the turbine-driven AFW pump has not increased from the current operating requirement and thus remains within the current design capacity of the turbine-driven AFW pump. Since worst case flow requirements do not increase and the existing pump design is unchanged, the maximum BHP requirement of the turbine-driven AFW pump remains the same as current operating conditions. Therefore, there is no additional HP demand on the existing turbine-driver for these AFW pumps due to EPU operation.

The steam to the turbine driver of the turbine-driven AFW pump is supplied by the main steam system. The evaluation of the steam supply capability concluded that sufficient steam can be supplied at EPU conditions to meet the BHP requirements of the turbine-driven AFW pump. See LR section 2.5.5.1, Main Steam, for details.

Pump Brake HP Requirements - Standby AFW Pumps

The EPU flow requirements for the standby AFW system increase to 235 gpm following a main feedwater line break that affects the AFW pumps and to 250 gpm following an Appendix R fire (water solid operation). Since the flow requirements increase, the BHP of the pump increases with the maximum BHP of 320 HP occurring at a flow rate of 250 gpm. This BHP requirement is above the motor rating of the existing 300 HP motors. The motors have a service factor of 1.15. The capability of the standby AFW pump motors is evaluated in LR section 2.3.3, AC Onsite Power System.

AFW and Standby AFW Safety Related Valves

The safety related valves in the AFW and standby AFW systems were reviewed to confirm that the EPU conditions of flow and differential pressure do not affect valve operation. Since, under EPU operation, there are no increases in the worst case AFW system flow rates, pressures and no hardware changes to the AFW pumps, the safety related valves are not impacted and are acceptable for operation under EPU conditions.

The flow rate in the standby AFW system has increased at EPU conditions from 200 gpm to 250 gpm. There are no increases in the maximum EPU pressures. The evaluation of safety related standby AFW valves confirms that these valves will close against the higher flow rates. See LR section 2.2.4, Safety-Related Valves and Pumps.

Summary

The Ginna review has adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the preferred AFW and standby AFW systems to supply adequate water to the steam generators to ensure adequate cooling of the core.

The preferred AFW system pumps and valves have been shown capable of providing the required flows and pressures necessary to perform the required design functions at EPU conditions and are therefore acceptable.

Following modification of the trim internals of the standby AFW pumps discharge control valves to accommodate the required EPU flow rates with the existing standby AFW pump flow / head output, the standby AFW system been shown capable of providing the required flows and pressures necessary to perform the required design functions at EPU conditions and is therefore acceptable.

The condensate storage tanks will maintain the current plant operational flexibility to store the required EPU water volume (based on the EPU decay heat removal requirements) in one tank per the existing plant technical specifications.

2.5.4.5.3 Conclusions

The Ginna staff has reviewed the assessment related to the AFW and standby AFW systems. The Ginna staff concludes the assessment adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the AFW and standby AFW systems to supply adequate water to the steam generators to ensure adequate cooling of the core. The Ginna staff finds that the AFW and standby AFW systems will continue to meet its design functions following implementation of the proposed EPU. The Ginna staff concludes that the AFW and standby AFW systems will continue to meet the current licensing basis with respect to the requirements of GDC-4, GDC-19, GDC-34, and GDC-44. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to the preferred AFW and standby AFW systems.

2.5.5 Balance-of-Plant Systems

2.5.5.1 Main Steam

2.5.5.1.1 Regulatory Evaluation

The main steam supply system transports steam from the NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. Ginna Nuclear Power Plant, LLC's (Ginna) review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the main steam supply system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures
- GDC-5, insofar as it requires that structures, systems, and components important-to-safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions
- GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core

Specific review criteria are contained in SRP, Section 10.3.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station main steam system design relative to conformance to:

- GDC – 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant

Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance relating to the requirements of GDC-4 is also described in the following:

- Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
- Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
  - Pipe Breaks Inside Containment (SEP Topic III-5.A)
  - Pipe Breaks Outside Containment (SEP Topic III-5.B)
- Missile Protection (Ginna UFSAR Section 3.5)

- GDC – 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna Station is a single unit installation so there are no shared structures, systems or components.
- GDC – 34 is described in Ginna UFSAR section 3.1.2.4.5, General Design Criterion 34 – Residual Heat Removal. As described in Ginna UFSAR section 5.4.5, Residual Heat Removal (RHR) system provides details as to how the system is used to transfer fission product decay heat and other residual heat from the reactor core.
- The Main Steam Header Dynamic Load Factor Analysis is discussed in Ginna UFSAR section 3.9.2.1.5 and the Seismic Piping Upgrade Program in section 3.9.2.1.8.

In addition to the evaluations described in the UFSAR, the Ginna Station's main steam system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The above SER describes main steam in section 2.3.4, Steam and Power Conversion Systems. The programs used to manage the aging effects associated with main steam are discussed in section 3.4 of the SER.

2.5.5.1.2 Technical Evaluations

2.5.5.1.2.1 Introduction

The main steam system is described in Ginna UFSAR section 10.3. The system provides heat removal from the reactor coolant system during normal, accident and post accident conditions. During off-normal conditions, the system provides emergency heat removal from the reactor coolant system using secondary heat removal capability. System components are also credited for safe shutdown following station blackout events and some fire events.

The main steam system is designed to produce dry saturated steam in the steam generators and direct it to the high pressure turbine, as well as other steam driven components and auxiliary systems. The main steam system includes the steam piping, main steam safety valves, atmospheric relief valves, main steam isolation valves, main steam non-return check valves, main steam flow venturis and other miscellaneous valves and piping. The main steam system also provides a flow path for steam from the steam generators to the steam dump valves (turbine bypass system) discussed in LR section 2.5.5.3 Turbine Bypass.

The reheat steam system is considered part of the main steam system for the Ginna Station. The reheat system delivers steam from the high pressure turbine exhaust through the moisture separator reheaters and then to the low pressure turbine inlets. The system is designed to remove up to 70% of the moisture in the preseparators prior to entering the moisture separator reheaters where the remaining moisture is removed, steam is dried and then super-heated (using main steam) and sent to the low pressure turbines. The reheat system includes the preseparators, the moisture separator reheaters, reheater stop and intercept valves, and the piping associated with this equipment.

The main steam system design functions are:

- Supply steam from the steam generators to the main turbine, turbine-driven auxiliary feedwater pump, moisture separator reheaters, main steam safety valve support heating, turbine gland sealing system, air ejectors, high pressure turbine flange heating, atmospheric relief valves and steam dump (turbine bypass) system
- Control steam generator pressure during startup and shutdown and when the condenser is not available
- Provide over-pressure protection for the steam generators
- Provide a primary containment isolation boundary
- Provide for main steam line and turbine warm-up
- Provide a means to dissipate the heat generated in the Nuclear Steam Supply System during all modes of normal operation, transient and accident conditions

The reheat steam system design functions are:

- Moisture removal from the high pressure turbine exhaust steam via moisture preseparators and moisture separators
- Supply superheated steam from the moisture separator reheaters to the low pressure turbines

The main steam system is described in the Ginna Station UFSAR Section 10.3, Main Steam System. Additional main steam system details are provided in Section 5.4.6, Main Steam and Feedwater Piping, Section 6.2.4, Containment Isolation System and Table 6.2-15a, Containment Piping Penetrations and Isolation Boundaries, Section 7.4.1.3, Main Steam System Instrumentation, and Section 10.1.1.1, Functional Description. Sections 10.2.1.2.1, 10.3.2.9 and 10.3.2.11 describe how the turbine speed is controlled.

2.5.5.1.2.2 Description of Analyses and Evaluations

The main steam and reheat steam systems and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations were conservatively performed for an analyzed NSSS thermal power of 1781 MWt. The evaluations compared the existing design parameters of the systems / components with the EPU conditions for the following design aspects:

- Design pressure / temperature of piping and components
- Flow velocities
- Vibration due to increased flow
- Capacities, closure times and set pressures for the main steam isolation and non-return check valves, main steam safety valves and atmospheric relief valves
- Moisture removal capability, thermal performance, vibration and erosion / corrosion of the moisture separator reheaters
- Main steam supply capacity to the turbine driven auxiliary feedwater pump and to other auxiliary loads
- Main steam drain system capacity

Other evaluations of main steam and reheat steam systems and components are addressed in the following LR section:

- Erosion / corrosion issues – LR section 2.1.8, Flow Accelerated Corrosion
- Piping / component supports – LR section 2.2.2.2, Balance of Plant Piping and Supports (Non-Class 1)
- Protection against dynamic effects, including missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Environmental qualification of the main steam isolation valves actuators – LR section 2.3.1, Environmental Qualification
- Main steam isolation valve testing and closure requirements – LR section 2.2.4, Safety Related Valves and Pumps
- Protection against turbine missiles and internal missiles in accordance with GDC-4 is discussed in LR section 2.5.1.2, Missile Protection
- Safety related instrumentation – LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems
- Turbine control / overspeed protection - LR section 2.5.1.2.2, Turbine Generator

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the main steam system are within the scope of License Renewal. EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the Main Steam System at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU, although parameter changes have been made to the Erosion-Corrosion/Flow Accelerated Corrosion Program. Thus, no new aging effects requiring management are identified.

2.5.5.1.2.3 Results

System Operating Conditions

Heat balances were developed to determine the steam cycle parameters while operating at the increased NSSS power level. Heat balances were developed for the current power level based on actual plant operating data and at the EPU power level of 1781 MWt. The EPU heat balances specify the required main steam flow rate at the high pressure turbine inlet throttle pressure of 730 psia. The process parameters are used by the manufacturer as the basis for redesign of the high pressure turbine to achieve the EPU electrical power generation.

Based on the heat balances, the table below lists the main steam conditions at the steam generator outlet and the reheat steam conditions to / from the moisture separator reheaters.

Table 2.5.5-1

	Current Operating Condition	EPU Operating Condition
Main Steam - Steam Generator Outlet		
Flow Rate, lbm/hr	6,541,741	7,751,835
Pressure, psia	782.4	802.0
Temperature, °F	515.7	518.5
Main Steam – Main Steam to HP Turbine		
Flow Rate, lbm/hr	6,025,398 (1)	7,193,996 (1)
Pressure, psia	730.0	730.0
Temperature, °F	515.7	518.5
Main Steam – Heating Steam to Moisture Separator Reheaters		
Flow Rate, lbm/hr	516,343	557,839
Pressure, psia	737.8	743.4
Temperature, °F	509	509.8
Reheat Steam – Moisture Separator Reheaters Inlet (Crossunder)		
Flow Rate, lbm/hr	5,473,140	6,567,208
Pressure, psia	148.3	176.9
Temperature, °F	357.6	371.7
Reheat Steam - Moisture Separator Reheaters Outlet (Crossover)		
Flow Rate, lbm/hr	4,467,166	5,354,408
Pressure, psia	137	163.1
Temperature, °F	497.5	492.4

Note 1: Includes minor steam flows diverted to gland steam system and condenser air ejectors (6,529 lb/hr at current operating conditions and 5,671 lb/hr at EPU conditions)

Piping Evaluations

Design Pressure / Temperature

The main steam system design pressure / temperature of 1085 psig (1100 psia) and 557°F bound the maximum EPU operating conditions of approximately 800 psia and 519°F. The system design pressure also bounds the highest normal operating pressure, which occurs at no load conditions of 1020 psia (547°F no load Tavg). The no load conditions are not affected by the EPU. Therefore, the existing design conditions are unchanged by the EPU.

The reheat crossover / crossunder piping was originally specified with the design conditions of 150 psig / 358°F for the crossunder piping (high pressure turbine exhaust to the moisture separator reheaters) and 150 psig / 480°F for the crossover piping (moisture separator reheaters to the low pressure turbine inlet). For EPU operating conditions the system is being rerated for a design pressure of 175 psig. Evaluation of the crossover and crossunder piping by the OEM has determined that the maximum working pressures of these pipe sections is greater than 175 psig.. Similarly, an analysis of the expansion joints in these reheat lines was performed and has concluded that they are acceptable for EPU conditions.

Flow Velocities

Flow velocities through the main steam piping from the steam generators to the turbine control and stop valves were calculated at current and EPU conditions. The flow velocities increased approximately 18% primarily due to the increased flow required by the EPU power level. The highest EPU velocity is 240 fps which is still significantly below the industry design guideline. EPU flow velocities in the main steam piping used for heating steam to the moisture separator reheaters increased approximately 8% with the highest velocity at 134 fps, again significantly below the industry design guideline.

Flow velocities through the reheat steam piping to / from the moisture separator reheaters were calculated at current and EPU conditions. Flow velocities in the reheat piping essentially stayed the same since, although the EPU flow rates and temperatures are higher, the increase in EPU pressures is enough to offset these effects on the steam specific volume. The highest EPU velocity (255 fps) is in the crossover piping downstream of the moisture separator reheaters. This velocity is essentially unchanged from the current condition and is well below the industry design guidelines.

The reheat steam crossunder piping from the high pressure turbine exhaust to the moisture separator reheaters inlets has experienced erosion to the extent that repairs have been necessary, either metal overlays or replacement of the affected area.

The Ginna erosion / corrosion program already monitors these lines, particularly at past erosion locations. The EPU velocity in this piping (215 fps) is only slightly higher than at the current operating conditions (211 fps) and, at EPU conditions, the moisture content is relatively unchanged. Therefore, the potential for erosion / corrosion is essentially unchanged by the EPU. Present monitoring activities will be continued after EPU.

Vibration

The increased steam flow velocity through piping and components has the potential to increase vibrations. Accordingly, during power ascension, piping will be monitored to identify line vibration anomalies. These vibration monitoring activities are discussed in LR section 2.12, Power Ascension and Testing Plan.

Component Design Evaluations

Design Pressure / Temperature

As described above under Piping Evaluations – Design Pressure / Temperature, the main steam design pressure and temperature are not affected by EPU operation. The design conditions of the main steam components: isolation valves, relief valves, venturis, etc., were reviewed and, in all cases, were above the EPU operating conditions and the main steam design conditions of 1085 psig and 557°F.

The moisture separator reheaters shells have a design pressure and temperature of 175 psig / 500°F which envelopes the maximum operating pressures. The moisture separator reheaters tubes contain main steam for heating and have a design pressure and temperature of 1100 psig / 575°F which envelopes the main steam design conditions of 1085 psig / 557°F.

Main Steam Safety Valves Capacities and Setpoints

The setpoints of the MSSVs are based on the design pressure of the steam generators (1085 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generator has not changed for EPU, there is no need to revise the setpoints of the safety valves.

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code (Section III, NB-7300 and NC-7300)) for the worst-case loss-of-heat-sink event. Based on this requirement, the original plant design applied a conservative guideline that the valves should be sized to relieve 100 percent of the design steam flow to ensure that maximum system pressure did not exceed 110 percent of MSS design pressure.

The Ginna Station has 8 safety valves with a total rated capacity of 6.58×10^6 lb/hr, which provides about 85 percent of the maximum EPU full-load steam flow.

The plant safety analysis for EPU (LR section 2.8.5, Accident and Transient Analysis) confirms that the installed safety valve capacity of 6.58×10^6 lb/hr is adequate for overpressure protection. Accordingly, the analysis demonstrates that the existing MSSVs are capable of maintaining the secondary side steam pressure below 110 percent of the steam generator shell design pressure.

The original design requirements for the MSSVs (as well as the Atmospheric Relief Valves (ARVs) and steam dump valves) included a maximum flow limit per valve of 890,000 lb/hr at 1085 psig. Since the capacity of any single MSSV, ARV, or steam dump valve has not been changed as a result of EPU, therefore, the maximum capacity criteria remains satisfied at EPU conditions.

The performance of MSSVs are acceptable at EPU conditions with no plant changes.

Atmospheric Relief Valves Capacities and Setpoints

The ARVs, which are located upstream of the MSIVs and adjacent to the MSSVs, are automatically controlled by steam line pressure during plant operations. The ARVs automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam line pressure decreases, the ARVs modulate closed and reseat at a pressure below the opening pressure. The ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest set MSSVs. Since neither of these pressures changes for the proposed range of NSSS design parameters, there is no need to change the ARV setpoint.

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs, in conjunction with the AFWS, permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point at which the Residual Heat Removal System (RHRS) can be placed in service. During cooldown, the ARVs are either automatically or manually controlled. In automatic, each ARV proportional and integral (P&I) controller compares steamline pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss-of-offsite power (LOOP), the ARVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere (LR section 2.8.5.6.2, Steam Generator Tube Rupture).

In the event of a loss-of-offsite power, the capacity of the ARVs permits a plant cooldown to RHRS operating conditions (350°F) in about 10 hours (at a rate of about 25°F/hr), assuming cooldown starts 4 hours after reactor shutdown. This capacity requirement is limiting with respect to sizing the ARVs, and bounds the capacity required for tube rupture.

An evaluation of the installed capacity of both ARVs (682,204 lb/hr at 1020 psia) indicates that the required plant cooldown can still be achieved for the range of EPU NSSS design parameters (LR section 2.8.7.2, Natural Circulation Cooldown).

The performance of the ARVs are acceptable at EPU conditions with no plant changes required to satisfy the decay heat removal requirements in accordance with Ginna Station current licensing basis requirements with respect to GDC-34.

Main Steam Isolation and Non-Return Check Valves

The main steam isolation valves and operators and non-return check valves are designed to close within 5 seconds after receipt of a closure signal. These valves must close for the purpose of main steam pipe break isolation, either inside or outside containment, and for containment isolation post accident.

The valves are designed for a differential pressure of 1200 psi, which is above the maximum system design pressure of 1085 psig. As discussed above, although the EPU operating pressures are higher than current operating pressures at full power, the design pressure of 1085 psig is not affected by EPU. Therefore, the valve design is unaffected.

The impact of the higher main steam flow rates through these valves during EPU operation was evaluated to confirm that the valves are not adversely affected in the open position during normal full power EPU operation and that the valves will close within the required time period during accident conditions.

The added pressure drop through the main steam isolation valves and non-return check valves at the normal EPU flow rates has been included in establishing the main steam supply pressure at the HP turbine inlet; thus ensuring adequate steam pressure for EPU full power generation.

The closure time of the main steam isolation valves and non-return check valves is not affected since the valve and operator design is based on the flow rate due to the worst case break flow that the valve experiences. The EPU does not affect the pipe break flows since the factors that affect maximum possible break flow, such as break size, location, steam generator pressure, and exit nozzle characteristics, etc are not affected by EPU. The higher flow rates during normal EPU operation will actually cause the valve to close faster once the disc enters the flow stream during scenarios requiring containment isolation without a pipe break. Refer to LR section 2.2.4, Safety-Related Valves and Pumps.

The higher flow rate during normal EPU operation through the main steam isolation valves and non-return check valves has the potential to cause vibration of the valve discs. The main steam isolation valves are installed in the reverse direction to flow and are provided with an operator which keeps its disc out of the flow stream. This design should minimize any added vibration caused by the higher EPU flow rate. Maintaining the disc out of the flow stream also minimizes any additional loads on the operator.

The main steam non-return check valve on each main steam header outside containment is installed in the direction of flow and closes on reverse flow due to a main steam pipe break inside containment. Plant operating data indicates that these valves have their discs operating in the flow stream and may experience turbulence. Since the increased steam flow due to EPU may aggravate this situation, the valves will be inspected during first outage after the EPU for potential wear / damage.

Moisture Separator Reheaters Relief Valves

The MSR relief valves are provided as overpressure protection for the moisture separator reheater vessels and for the piping from the high pressure turbine exhaust through the moisture separator reheaters and to the low pressure turbine inlet valves. The valves' relieving capacity and setpoints must be sufficient to pass the maximum operating EPU steam flow exiting the moisture separator reheaters and maintain the ASME pressure limits on the moisture separator reheaters and connected piping.

The higher steam flow and pressure at EPU operating conditions requires that the relief capacity for the MSRs must increase in order to provide sufficient overpressure protection. Modifications will be made to the relief system to meet the EPU conditions.

Moisture Separator Reheaters

The moisture separator reheaters were evaluated by Thermal Engineering International, Inc, (TEI) to determine the impact of the increased steam flow and pressure during EPU operation on tube vibration, thermal performance, moisture removal capability and erosion / corrosion effects.

The TEI evaluation concluded:

- The reheater bundles will experience EPU flow induced vibration for at least 18 more years without failure.
- The reheater bundles will perform sufficiently to meet the thermal requirements of EPU operation at 100% power generation.
- The moisture removal capability of the moisture separator and the internal drain system is sufficient for the EPU moisture loading however, minor modifications are required to the second and fourth pass drain paths to ensure adequate venting of the these drain lines..
- Based on previous MSR inspections at Ginna and similarly designed moisture separator reheaters at other plants and on a technical analysis of the impact of temperature, velocity and the moisture separator reheaters physical characteristics, there are no erosion / corrosion concerns during EPU operation. The Ginna station will continue to inspect the moisture separator reheaters shell and nozzles as part of the erosion / corrosion program after EPU to confirm the evaluation conclusions. See LR section 2.1.8, Flow Accelerated Corrosion.

Following modification to the MSR second and fourth pass drain lines, the moisture separator reheaters are acceptable for operation at the EPU.

Turbine Stop, Control, and Intercept Valves

The HP turbine stop valves have been evaluated by the OEM as being adequate for EPU conditions. The HP turbine control valves are being modified to improve overall plant efficiency at the EPU conditions. The low pressure turbine reheat stop and intercept valves have been evaluated by the OEM as being adequate for EPU conditions. The ability of the Reheat stop and intercept valves to provide overspeed protection is discussed in LR section 2.5.1.2, Missile Protection.

Auxiliary Main Steam Supply Flow Rates

The auxiliary feedwater pump turbine supply and exhaust piping were determined to be acceptable for EPU conditions. The pressure ratings remain bounding. The required steam supply flow rate to the pump turbine is not affected by the EPU since the design brake horsepower (bhp) of the auxiliary feedwater pump turbine bounds the bhp required to supply the

maximum EPU auxiliary feedwater flow rate. See LR section 2.5.4.5, Auxiliary Feedwater System.

The main steam system's ability to supply steam to auxiliary components, including the turbine gland steam supply and the condenser air ejectors, will not be affected by the EPU. None of these steam flow requirements change appreciably due to EPU conditions. The EPU heat balances include these required auxiliary flows and confirm that sufficient main steam flow exists to ensure the high pressure turbine and moisture separator reheaters performance meets the desired EPU power generation requirements.

Main Steam Piping Drain Capacity

The main steam piping is provided with drains to collect water condensing in the piping. The drains were originally designed with sufficient capacity based on start-up conditions where hot steam is introduced into cold piping. Since startup steam conditions, e.g., flow, quality, temperature and pressure, are not affected by the EPU and since there are no changes to the main steam piping arrangement, the existing drain system is acceptable.

Summary

The operating pressure and temperature associated with the EPU conditions are bounded by the design of the main steam system piping and components. The main steam flow velocities at EPU conditions are below industry guidelines and remain acceptable. Pressure drops in main steam and reheat steam lines are acceptable as shown by the EPU heat balances, which confirm that the predicted thermal performance of the plant results in the expected EPU MWe output.

The main steam safety valves and atmospheric relief valves design conditions, capacity, and setpoints are adequate for EPU operation. In the event of a loss-of-offsite power, the capacity of the ARVs permits a plant cooldown to RHRS operating conditions.

The main steam isolation valves and operators are acceptable for EPU flow conditions and will satisfactorily close in the required time period because the existing valve design parameters of flow rate and differential pressure bound the EPU conditions. The main steam non-return check valves will be inspected at the first outage after EPU to monitor the wear of the disc and internal parts.

The moisture separator reheaters of the reheat steam system will continue to provide sufficient moisture removal capability and adequate thermal performance during EPU operation and will operate without excessive vibration or erosion / corrosion at the EPU flow rates and pressure / temperature conditions. Minor modifications are required to the MSR reheater second and fourth drain path drain piping to ensure adequate venting at EPU condition.

The reheat steam piping is acceptable for EPU conditions since the design pressure / temperature envelope the EPU conditions. However, the MSR relief valve capacity is inadequate to provide the required MSR shell and reheat piping system overpressure

protection. Therefore, the MSR pressure lift capacity will be modified to provide the required system overpressure protection.

The increased steam flow velocity through piping has the potential to increase vibrations. Accordingly, during power ascension, piping will be monitored to identify line vibration anomalies.

As described in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, 2.2.2.2, Balance of Plant Piping and Supports (Non-Class 1), and 2.5.1.3, Pipe Failures, the main steam system is capable of withstanding steam water hammer and the safety related portions of the system continue to be protected against the dynamic effects of EPU, including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures.

2.5.5.1.3 Conclusions

The Ginna staff reviewed the assessment of the effects of the proposed EPU on the main steam supply system and concludes that the assessment adequately accounted for the effects of changes in plant conditions on the design of the main steam supply system, with the modifications as described. The Ginna staff concludes that the main steam supply system will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. The Ginna staff further concludes that the main steam supply system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4, GDC-5 and GDC-34. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to the main steam supply system.

2.5.5.2 Main Condenser

2.5.5.2.1 Regulatory Evaluation

The main condenser is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. The Ginna Nuclear Power Plant, LLC (Ginna) review focused on the effects of the proposed EPU on the condenser's ability to accommodate the higher heat removal requirements of the turbine exhaust steam flow. The Ginna review also focused on the steam bypass following a load rejection assumption, and on the ability of the main condenser system to withstand the blowdown effects of steam from the turbine bypass system.

The NRC's acceptance criteria are based on:

- GDC-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

Specific review criteria related to these GDC are contained in NRC Standard Review Plan (SRP), section 10.4.1.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station Main Condenser design relative to conformance to:

- GDC-60 is described in UFSAR section 3.1.2.6.1, General Design Criterion 60 – Control of Release of Radioactive Materials to the Environment. As described in this UFSAR section, the handling, control, and release of radioactive materials during MODES 1 and 2 are in compliance with 10CFR50, Appendix I, and are described in the Offsite Dose Calculation Manual. Control of releases of noble gases from the main condenser is described in UFSAR 11.5.2.2.8.

In addition to the evaluations described in the UFSAR, Ginna Station systems were evaluated for plant License Renewal. System and system component materials of construction, operating

history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, the main condenser is not within the scope of License Renewal. However, the programs used to manage the aging effects associated with steam and power conversion systems are discussed in section 3.4 of the Ginna Station License Renewal Application.

2.5.5.2.2 Technical Evaluation

2.5.5.2.2.1 Introduction

The main condenser is discussed in the Ginna UFSAR section 10.4.3. The main condenser is a two-shell, single-pressure, deaerating type surface condenser of the radial flow type with semi-cylindrical water boxes bolted at both ends. The condenser extracts the latent heat of vaporization from the low pressure turbine exhaust steam, the steam dump system (when in operation) and miscellaneous flows, drains and vents during normal plant operation. This heat is transferred to the circulating water system. The resulting condensate is collected in the condenser hotwell before entering the condensate and feedwater system. The condensate hotwell level control system maintains sufficient level to provide the suction head for the condensate pumps. The condenser deaerates the condensate before it leaves the condenser hotwell.

The condenser utilizes circulating water for heat removal which then transfers the rejected heat to Lake Ontario. The circulating water system is described in UFSAR section 10.6. The evaluation of the EPU effect on the circulating water system is described in LR section 2.5.8.1, Circulating Water.

The steam dump system is discussed in the Ginna UFSAR section 10.7.1. The purpose of the steam dump system is to minimize the stresses on the nuclear steam supply system induced by changes in the secondary plant steam loads. At EPU conditions, the steam dump valves will by-pass to the main condenser a maximum steam flow of approximately 30 percent of the EPU total steam flow from the main steam headers. Refer to LR section 2.5.5.3, Turbine Bypass, for additional discussion of the steam dump system.

2.5.5.2.2.2 Description of Analyses and Evaluations

The main condenser will experience higher steam flows due to the increase in LP turbine exhaust flow at the EPU power level during normal power operation. The main condenser will also experience higher steam demands following a load rejection event at EPU operating condition due to operation of the turbine by-pass system which would cause part of the total main steam flow to be directly rejected to the condenser. The evaluation determined the impact of the EPU conditions on condenser performance and integrity as follows:

- Determine the increased condenser duty and confirm the condenser's ability to reject heat to the circulating water system and maintain a low enough condenser backpressure for the turbine to meet its EPU MW output and performance requirements.

- Evaluate the condenser hotwell storage capacity to provide sufficient storage volume with the maximum flow rate at EPU conditions.

- Evaluate the capability of the main condenser to remove dissolved gases and air in-leakage from the condensate.

- Evaluate the steam blowdown effects of increased steam flow at normal EPU power operation and during steam dump to the condenser following load rejection on condenser tube vibration.

- Evaluate the impact of the increased steam dump flow on condenser backpressure during steam dump conditions and confirm that none of the automatic plant protection setpoints, such as turbine trip on loss of condenser vacuum, are initiated.

- Evaluate the impact of the increase steam flow on the condenser spargers, baffles, and impingement plates, provided to protect the condenser tube and internal components from damage due to incoming steam and water flows.

- Evaluate the impact of the increased steam flow on the plant design to control the release of radioactive effluents in accordance with GDC-60.

2.5.5.2.2.3 Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal

Ginna Station systems were evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004. With respect to the above SER, the main condenser is not within the scope of License Renewal. However, the programs used to manage the aging effects associated with steam and power conversion systems are discussed in section 3.4 of the Ginna Station License Renewal Application. The changes associated with operating the main condenser at EPU conditions do not add any new or previously unevaluated materials to the system. With the possible exception of increased wear of condenser internals due to increased LP turbine exhaust and bypass steam flows, component internal and external environments remain within the parameters previously evaluated. Main condenser internals will be subject to periodic inspection to identify any increased wear.

2.5.5.2.3 Results

The evaluation determined that the condenser satisfactorily removes the increased EPU heat loads, condenses the required steam flows and maintains an acceptable vacuum using circulating water at the current normal operating flow rate. Table 2.5.5.2-1 describes the key design parameters of the main condenser for each condenser shell and compares its performance at current operating and EPU conditions.

Table 2.5.5.2-1

Main Condenser Performance Characteristics

	Condenser Design Specification	Current Operating Value 100 % Power	EPU Operating Value 100% Power
Condenser Duty / Shell	1.640 x 10^9 Btu/hr	1.71 x 10^9 Btu/hr	2.01 x 10^9 Btu/hr
CW Inlet temperature	50 °F	47.5°F	40°F
CW Temperature Rise	19.6 °F	20.5 °F	24.0°F
Condenser Backpressure	1.35 inch HgA	1.31 inch HgA	1.43 inch HgA

As stated in UFSAR Section 10.4.3, the condenser hotwell for the existing design conditions provides approximately 3 minutes of condensate storage capacity at maximum throttle flow with an equal free volume for surge protection.. At EPU conditions, for the existing hotwell level, the hotwell will provide approximately 2.5 minutes of storage and surge capacity which is more than the volume recommended by the HEI Standard for Steam Surface Condensers (i.e., one minute of condensate reserve at full power operation). Therefore, the existing hotwell capacity is acceptable for EPU conditions. UFSAR section 10.4.3 will be revised to reflect EPU conditions as part of the UFSAR update program.

The ability of the condenser to maintain the required deaeration of condensate flow remains acceptable at EPU conditions. The condensate pump discharge oxygen level (CDPO) at EPU conditions remains below the original design value for the condenser of 7.0 ppb. Only the noncondensibles, not the air-inleakage, collected in the condenser will increase at EPU conditions. Although the total accumulation from both sources increases from approximately 2.5 to 3.0 scfm, it remains well within the capability of the existing condenser evacuation system as discussed in LR section 2.5.3.2, Main Condenser Evacuation System. The HEI Standard for Steam Surface Condensers indicates that, in order to maintain CPDO at or below 7 ppb with the existing condenser evacuation system capacity of 30 SCFM, the total rate to be removed should be limited to 4.5 scfm. Since the EPU condition of 3 scfm air in-leakage rate is well below the HEI recommendation of 4.5 scfm, the condenser deaeration ability is acceptable for the EPU conditions.

The evaluation also confirmed that the condenser adequately withstands the steam blowdown effects of a steam dump following a load rejection. A main condenser tube vibration evaluation determined that the existing tube span lengths of 25 inches for the spacing associated with

intermediate support plates and 20 inches for the end plate spans is adequate for the EPU conditions. The tube spans lengths used credit the tube stakes that were installed into the tube bundle during the 1995 re-tubing of the condenser with stainless steel tubes. Therefore, no modifications are required for the existing condenser tube supports for EPU operation.

The increased steam flow rates at EPU conditions of normal operation and steam dump may increase the wear of condenser internal spargers, baffles, and impingement plates. Therefore, these components were inspected during the 2005 refueling outage to baseline their condition. Based on the inspection results, modifications were implemented to ensure the spargers, baffles, and impingement plates are adequate for EPU operation. Following EPU, monitoring of these components will continue.

The current turbine trip and alarm set points for condenser backpressure are not affected by the increased steam flow rates at EPU conditions for normal operation and steam dump flow following a load rejection. The highest normal backpressure at EPU full power operation is expected to be ~3.5 inches HgA based on a lake temperature of 80°F. The maximum possible condenser back-pressure following a load rejection at EPU was calculated to be 5.2 inches HgA based on conservatively assuming all condenser dump valves open with 35 % total steam dump flow and continued full power operation. Since the maximum steam dump capacity at EPU is ~30 % and control rod runback due to the load rejection will decrease reactor power, the actual condenser back-pressure for a load rejection would be less than the 5.2 in HgA calculated value. The 5.2 inch HgA calculated back-pressure is below the 5.5 inch HgA alarm setting for the turbine "Do Not Operate" region and is well below the turbine trip set point of 9.92 inches HgA. The steam dump permissive setpoint of 20 inches HgA condenser backpressure is similiarly not affected by either of these EPU operating conditions.

The design of the main condenser does not change following the implementation of the EPU. Therefore, the EPU does not impact the ability of the Ginna Station regarding the control of radioactive material in accordance with GDC-60. Monitoring of the air and non-condensibles leaving the condenser is accomplished by a radiation monitor in the condenser evacuation system, described in LR section 2.5.3.2, Main Condenser Evacuation System. The impact of EPU on radiological effluent releases from the Ginna Station, radiation monitoring setpoints and compliance with 10CFR50, Appendix I, are discussed in LR section 2.10.1, Occupational and Public Radiation Doses.

2.5.5.2.4 Conclusion

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the main condenser system and concludes the assessment adequately accounted for the effects of changes in plant conditions on the design of the main condenser. The Ginna staff concludes that the main condenser system will continue to maintain its ability to withstand the blowdown effects of the steam from the turbine bypass system and; thereby, continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-60 for preventing the consequences of failures in the system. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to the main condenser.

2.5.5.3 Turbine Bypass

2.5.5.3.1 Regulatory Evaluation

The turbine bypass system, which at the Ginna Nuclear Power Plant LLC (Ginna) is referred to as the steam dump system, is designed to discharge a portion of main steam flow directly to the main condenser system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the turbine bypass system capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control steam generator pressure. The Ginna review focused on the effects that EPU has on load rejection capability, analysis of postulate system piping failures, and on the consequences of inadvertent turbine bypass system operation. The NRC's acceptance criteria for the turbine bypass system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures

- GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded

Specific review criteria are contained in SRP section 10.4.4.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station Turbine Bypass System design relative to conformance to:

- GDC – 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic

Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC-4 is described in the following:

- Environmental Design Of Mechanical And Electrical Equipment (UFSAR section 3.11)
- Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (UFSAR section 3.6)
  - Pipe Breaks Inside Containment (SEP Topic III-5.A)
  - Pipe Breaks Outside Containment (SEP Topic III-5.B)
- Missile Protection (UFSAR Section 3.5)

- GDC – 34 is described in the Ginna UFSAR section 3.1.2.4.5, General Design Criterion 34 – Residual Heat Removal. Ginna UFSAR section 5.4.5 also provides details as to how the residual heat removal system is used to transfer fission product decay heat and other residual heat from the reactor core.
- Main steam header dynamic load factor analysis is discussed in the Ginna UFSAR section 3.9.2.1.5 and the seismic piping upgrade program in section 3.9.2.1.8.

In addition to the evaluations described in the Ginna UFSAR, the main steam system, which includes the steam dump system, was evaluated as part of the Ginna Plant License Renewal. System and component materials of construction, operating history and the plant programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R. E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Turbine Bypass (steam dump) piping and components are not within the scope of License Renewal. With respect to the above SER, the main steam system is described in sections 2.3.4.1, Main and Auxiliary Steam System. The programs used to manage the aging effects in main steam are discussed in section 3.4.2.4.1, Main and Auxiliary Steam System.

2.5.5.3.2 Technical Evaluation

2.5.5.3.2.1 Introduction

The steam dump system (turbine bypass system) is described in the Ginna UFSAR section 10.7.1. The steam dump system consists of eight condenser dump valves and piping from the main steam headers to the condenser. The purpose of the steam dump system is to minimize the stresses on the nuclear steam supply system induced by changes in the secondary plant steam demand. The condenser steam dump valves are designed to pass 40% of main steam flow at current 100% power operation and at full-load steam pressure. In conjunction with the rod control system, which accommodates 10% of the load reduction, the steam dump system permits the NSSS to withstand an external load reduction of up to 50% of plant rated electrical

load without a reactor / turbine trip. In addition to limiting the reactor coolant system temperature and pressure transients following reductions in steam loads, the steam dump system also serves to minimize the undesirable possibility of lifting the pressurizer and main steam safety valves and aids in conducting reactor coolant system cooldowns and heatups. There is a control system interlock which prevent initiation of turbine bypass when the condenser is at or above a pressure of 20" HgA.

2.5.5.3.2.2 Description of Analyses and Evaluations

The components in the steam dump system were evaluated to ensure they are capable of performing their intended functions at EPU conditions thereby ensuring the functionality of the system at EPU conditions. The evaluations addressed the following:

- Component design parameters versus the EPU operating conditions.
- Piping velocities versus industry standard guidelines.
- Dynamic effects, related to GDC-4, including the effects of missiles, pipe whipping, and discharging are addressed in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and 2.5.1.3, Pipe Failures.
- Response to design basis loading and unloading transients are discussed in LR section 2.4.2, Plant Operability

The steam dump system was also evaluated to ensure that the system was capable of performing its intended function for the range of NSSS design parameters approved for EPU (LR section 1.1, Design Parameters, Table 1-1). The evaluation was conservatively performed for the analyzed NSSS thermal power of 1817 MWt.

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse original sizing criterion conservatively recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant-rated electrical load without a reactor / turbine trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the rod control system, which accommodates 10 percent of the load reduction. The steam dump system prevents MSSV lifting following a reactor trip from full power.

The Ginna Station is equipped with 8 condenser steam dump valves and each valve is specified to have a flow capacity of 302,500 lbm/hr at a valve inlet pressure of 710 psia.

The capacity of the steam dump system (as a percentage of full-load steam flow) decreases as full-load steam pressure decreases and full-load steam flow increases. Accordingly, NSSS operation within the proposed range of design parameters for EPU will result in a reduced steam dump capability relative to the original Westinghouse sizing criteria. An evaluation indicates steam dump capacity could be as low as 24.3 percent of rated steam flow (7.87×10^6

lb/hr), or 1.91 x 10^6 lb/hr at a full-load steam pressure equal to 700 psia. At full-load steam pressures higher than 700 psia (T_{avg} = 564.6°F), steam dump capacity would increase. For example, at a full-load steam pressures of 804 psia (T_{avg} = 576°F), steam dump capacity would be 30.3 percent of rated flow (7.44 x 10^6 lb/hr), or 2.26 x 10^6 lb/hr (Ref. A). Typically a steam dump capacity of 30 percent or less is adequate to prevent challenging the main steam ARVs and MSSVs following a reactor trip from full load. Therefore the steam dump capacity at EPU will is sufficient to satisfy this requirement following EPU.

As described in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems, minor adjustments are being made to the steam dump control system to offset the less than 40 percent of EPU rated flow to ensure the plant will continue to satisfactorily respond to design basis loading and unloading transients described in Licensing Report 2.4.2, Plant Operability.

The NSSS stability and operability analysis described in LR section 2.4.2, Plant Operability, provides an evaluation of the adequacy of the steam dump system in conjunction with the changes to the control system setpoints at EPU conditions. This section states that a 50-percent load rejection with steam dumps available to the condenser can be accommodated without resulting in either a reactor / turbine trip or steam generator safety valve actuation. The analysis results indicate that for the range of NSSS design parameters approved for EPU, a rapid ramp load decrease equivalent to 50-percent of the EPU rated thermal power at a maximum turbine unloading rate of 200% per minute can be accommodated with no plant hardware changes. At less than 50 percent of the EPU rate thermal power, a turbine trip can be accommodated without a reactor trip occurring.

Based on these analyses, the condenser steam dumps meet requirements at EPU conditions as discussed above.

The condenser steam dump valves have NSSS requirements on time for opening and for modulating steam flow. To provide effective control of flow on large step-load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full-load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. For modulating steam dump flow, the positioning response may be slower with an allowed maximum full-stroke time of 20 seconds. These time response requirements are not affected by the EPU and must still be met.

The original design requirements for the steam dump valves, as well as the atmospheric relief valves (ARVs) and main steam safety valves (MSSVs) included a maximum flow limit per valve of 890,000 lb/hr at 1085 psig. The capacity of any single MSSV, ARV, or steam dump valve is not being changed for EPU. LR section 2.8.5.1.1.2.4, Inadvertent Opening of a Steam Generator Relief or Safety Valve, demonstrates that the results of a stuck-open steam generator relief valve or a safety valve are bounded by the large (hypothetical) steam line break results. Therefore it can be logically concluded that a stuck-open steam dump valve is also bounded by the large steam line break.

The performance of the steam dump system is acceptable at EPU conditions with no plant changes in accordance with Ginna Station current licensing basis requirements with respect to GDC-34.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Turbine Bypass (steam dump) piping and components are not within the scope of License Renewal. With respect to the License Renewal SER (NUREG-1786), the steam dump system is part of the main steam system which is described in sections 2.3.4.1, Main and Auxiliary Steam System. The programs used to manage the aging effects in main steam are discussed in section 3.4.2.4.1, Main and Auxiliary Steam System.

2.5.5.3.2.3 Results

The steam dump system piping design pressure of 1085 psig (1100 psia) bounds the EPU operating conditions of approximately 800 psia and also bounds the highest operating pressure which occurs at no load conditions of 1020 psia (no load T_{avg} 547°F). The no load conditions are not affected by the EPU.

The steam dump system performance is acceptable at EPU conditions with no plant changes. The results of the EPU steam dump system performance evaluation coupled with other EPU analysis results can be summarized as follows:

- NSSS operation within the proposed range of design parameters for EPU will result in a reduced steam dump capability relative to the original Westinghouse sizing criteria. However, the NSSS stability and operability analysis (LR section 2.4.2, Plant Operability) concludes that the reduced capacity is adequate to achieve 50-percent load rejection (that is, no reactor / turbine trip) for the range of NSSS design parameters approved for EPU.

- The steam dump capacity at EPU conditions is adequate to prevent MSSV lifting following reactor trip from full power.

- The actual capacity of any single steam dump valve at EPU conditions is not impacted by EPU. Accordingly, as documented in LR section 2.8.5.1.1.2.4, inadvertent opening of a steam dump valve is acceptable at EPU conditions with no plant changes.

- The performance of the steam dump system is acceptable at EPU conditions with no plant changes in accordance with Ginna Station current licensing basis requirements with respect to GDC-34.

Although the steam dump system is not within the scope of License Renewal, portions of the main steam system are within the scope. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the steam

dump system at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.5.3.3 Conclusion

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the steam dump (turbine bypass) system. The Ginna staff concludes that the assessment has adequately accounted for the effects of changes in plant conditions on the design of the system. The Ginna staff concludes that the steam dump system will continue to provide a means for shutting down the plant during normal operations. The Ginna staff further concludes that steam dump system failures will not adversely affect essential systems or components. Based on this, the Ginna staff concludes that the steam dump system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4 and GDC-34. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the steam dump system.

2.5.5.4 Condensate and Feedwater

2.5.5.4.1 Regulatory Evaluation

The condensate and feedwater system provides feedwater at the appropriate temperature, pressure, and flow rate to the steam generators. The only part of the condensate and feedwater system classified as safety-related is the feedwater piping from the steam generators up to and including the outermost containment isolation valve. The Ginna Nuclear Power Plant, LLC (Ginna) staff review focused on the effects of the proposed EPU on previous analyses and considerations with respect to the capability of the condensate and feedwater system to supply adequate feedwater during plant operation and shutdown, and to isolate components, subsystems, and piping in order to preserve the system's safety function. The Ginna staff's review also considered the effects of the proposed EPU on the feedwater system, including the auxiliary feedwater system piping entering the steam generator, with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation, as well as during upset or accident conditions.

The NRC's acceptance criteria for the condensate and feedwater system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such structures, systems, and components be protected against dynamic effects

- GDC-5, insofar as it requires that structures, systems, and components important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to ensure that the system safety function can be accomplished, assuming a single failure

Specific review criteria are contained in SRP Section 10.4.7.

Note that Ginna Plant Technical Specification 3.7.3, Main Feedwater Isolation Valves, Associated Bypass Valves, and Main Feedwater Pump Discharge Valves, require additional controls on components that provide automatic feedwater isolation. Because of these controls, the Main Feedwater Isolation Valves and their bypass valves are maintained safety-related. Additionally, to prepare for EPU, Ginna submitted a License Amendment Request dated April 29, 2005 to change this technical specification to substitute faster acting Main Feedwater Isolation Valves located outside containment in place of the Main Feedwater Pump Discharge Valves to terminate feedwater flow to the affected steam generator in case of a steam line break

or feedwater line break inside containment in order to limit the mass and energy release from such breaks.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station condensate and feedwater system design relative to conformance to:

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC-4 is described in the following:
    - Environmental Design Of Mechanical And Electrical Equipment (UFSAR section 3.11)
    - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (UFSAR section 3.6)
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)
    - Missile Protection (UFSAR Section 3.5)
        - Externally generated missile effects (SEP Topic III-4.A)
        - Internally generated missile effects (SEP Topic III-4.C)
        - Turbine missiles (SEP Topic III-4.B)
- GDC-5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna Station is a single unit installation so there are no shared structures, systems or components.

- GDC-44 is described in Ginna UFSAR section 3.1.2.4.15, General Design Criterion 44 Cooling Water. GDC-44 addresses provision of a system to transfer heat from safety-related structures, systems, and components to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions. Ginna UFSAR section 3.1.2.4.15 states that the Ginna Station includes redundant component cooling and service water design features to transfer heat to the ultimate heat sink. Note that the condensate and feedwater systems are not specifically addressed; however, the feedwater system does have safety functions, which are considered as part of this GDC, by providing redundant flow paths for the auxiliary feedwater system flow to the steam generators for heat removal from the reactor coolant system and by providing the required safety related, redundant isolation functions of main feedwater during postulated steamline breaks.

Other Ginna UFSAR sections that address the design features and functions of the condensate and feedwater systems include:

- Ginna UFSAR section 3.7.3.7, Seismic Piping Upgrade Program (SEP Topic III-6), which upgraded certain Seismic Category I Piping systems to more current requirements and provided a seismic data base for use with modifications, the inservice inspection program, and NRC requests for information. The upgrade program included the main headers of the feedwater piping inside containment from the steam generators to the containment penetrations and the main feedwater headers outside containment up to just upstream of the main feedwater regulating valves.

- Ginna UFSAR section 6.2.1.2.3, Secondary System Pipe Break Analysis, which describes the isolation requirements for the feedwater system during postulated pipe breaks .

- Ginna UFSAR section 6.2.4, Containment Isolation System, which describes the containment isolation features to isolate the feedwater system containment boundaries (SEP Topic VI-4).

- Ginna UFSAR Chapter 15 analyses, which describe the isolation of feedwater lines, containment isolation and auxiliary feedwater supply via the main feedwater headers to the steam generators during the transient and accident analyses described in the UFSAR.

- Ginna UFSAR section 3.9.2.1.6, Secondary System Water Hammer, and UFSAR section 3.9.2.1.8, Seismic Piping Upgrade Program (SEP Topic III-6), which describe the analysis of postulated water hammer in feedwater piping and seismic analysis / documentation of piping and support designs.

In addition to the evaluations described in the UFSAR, the Ginna Station's condensate and feedwater system was evaluated for plant License Renewal. System and system component

materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

With respect to the above SER, feedwater and condensate is described in section 2.3.4.2, Feedwater and Condensate Systems. Aging effects, and the programs used to manage the aging effects associated with feedwater and condensate, are discussed in section 3.4.2.4.2, Feedwater and Condensate Systems.

2.5.5.4.2 Technical Evaluation

2.5.5.4.2.1 Introduction

The condensate and feedwater systems are described in Ginna UFSAR sections 10.4.4 and 10.4.5, respectively. The condensate and feedwater systems function to collect steam condensed from the low pressure turbines'exhaust in the main condenser, heat this condensate, and then send it back to the steam generators at the temperature and pressure required for heat removal from the reactor coolant system as well as power generation at EPU conditions. Safety-related components and piping within the condensate and feedwater system are used for auxiliary feedwater addition and containment isolation and feedwater isolation during accidents and transients as well as being the main feedwater flow paths to each steam generator during normal operation.

Specific condensate and feedwater system design functions include:

Condensate System:

- Provides sufficient flow (utilizing the condensate booster pumps and heater drain tank pumps) at the required pressure to meet the feedwater pump requirements
- Improves plant cycle thermal efficiency by heating the condensate water through four stages of condensate heaters prior to entering the suction of the feedwater pumps
- Provides for makeup and transfer of water between the condenser hotwell and condensate storage tank during plant load changes and to offset fluid losses during normal operation
- Provides a condensate-grade preferred source of water from the condensate storage tanks for auxiliary feedwater use
- Supplies flow for cold water injection to the heater drain tank
- Provides sufficient cold water flow through the condensate cooler to the main generator hydrogen coolers for removing heat from the main generator

- Provides cold water flow to the feedwater pumps, condensate pumps, condensate booster pumps and heater drain pumps for seal cooling and lubrication
- Performs condensate and feedwater clean-up operations during plant start-up
- Performs purification of the condensate and feedwater system through the use of the all-volatile water treatment system (when needed)
- Provides condensate water to vacuum packed valves
- Provides blocking water to the preferred auxiliary feedwater and standby auxiliary feedwater pump suction
- Acts as a back-up water supply to the demineralizer water system for primary demineralized make-up water
- Supplies condensate water to the standby auxiliary feedwater test tank
- Provides an alternate source of condensate-grade water to the condensate storage tanks from the outside condensate storage tank
- Provides cooling water to the air ejector condenser and gland steam condenser
- Provides cooling water to the condenser expansion joints
- Maintains the outside condensate storage tank above freezing
- Recycles water to the primary demineralizer (DI) system for treatment

Feedwater System:

- Provides feedwater flow and pressure to the steam generators
- Maintains the steam generator water level
- Supplies feedwater at the required temperature to the steam generators
- Improves the plant cycle thermal efficiency by heating the feedwater through one stage of feedwater heaters
- Provides sufficient flow within the maximum / minimum limitations established by the NSSS transient analyses
- Meets containment isolation requirements
- Provides isolation of feedwater in event of high energy pipe breaks

The condensate system includes three nominal 50% capacity condensate pumps and three nominal 50% capacity condensate booster pumps. The feedwater system includes two nominal

50% capacity feedwater pumps. The heater drain system includes two nominal 50% capacity heater drain tank pumps.

Condensate drains from the condenser hotwell to the condensate pumps; which, in series with the condensate booster pumps, supply water to the suction of the feedwater pumps, which provides feedwater to the steam generators. The heater drain tank collects the drainage from the feedwater heaters 5A/B & 4A/B, preseparator tanks A/B, and moisture separator reheaters 1A/B & 2A/B. The heater drain tank pumps take suction from the heater drain tank and deliver the water to the condensate system at the suction header of the main feedwater pumps.

During normal plant power generation, the condensate polishing demineralizers are bypassed. Four stages of low pressure feedwater heating and one stage of high pressure feedwater heating are provided; arranged in two separate, parallel trains. The condensate system also provides cooling water for the air ejector condensers, gland steam condenser and, using the condensate coolers cooled by circulating water, to the generator hydrogen coolers.

2.5.5.4.2.2 Description of Analyses and Evaluations

The condensate and feedwater systems and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluation considered the effects of the EPU on the following system / component design aspects:

- Design pressures / temperatures of piping, valves and components versus EPU operating pressures / temperatures
- Flow velocities
- Feedwater isolation valves closure within the required time period at EPU hydraulic conditions of flow and pressure drop
- Capacity and control capability of the feedwater regulating valves
- Feedwater heaters design parameters and operating characteristics listed below.
    - Thermal performance
    - Shell side and tube side velocities, including steam dome velocity
    - Steam and water nozzle velocities
    - Tube support plates spacing / thickness
    - Tube vibration
    - Shell and tube side pressure drops
    - Shell and tube side relief valve capacities & setpoints
    - Shell side venting capacity
    - Steam impingement and tube vibration
    - Shell side and tube side design pressure / temperature
- Pump and pump supporting subsystems design capabilities, including NPSH, flow, head, brake horsepower, minimum flow protection and seal water supplies

- Auxiliary heat exchangers heat removal and flow requirements
- Process setpoints for protective functions, such as pump NPSH

The condensate and feedwater systems were evaluated by utilizing a hydraulic model of the system components and piping and the EPU heat balances. Physical plant data for the installed components and piping were utilized in the hydraulic model. Any physical changes to condensate and feedwater components, valves and piping which resulted from the EPU evaluations were incorporated into the hydraulic model and verified as acceptable.

Current plant operating data were gathered and included in the current operating heat balances to reflect the present day performance of the existing components. The current operating heat balances were then scaled to the EPU operating conditions and issued as EPU heat balances. The EPU heat balances were used to establish the flow, temperatures and heat transfer requirements at the EPU power level.

Other evaluations of condensate and feedwater systems and components are addressed in the following LR section:

- Effects of increased flow and velocity on erosion / corrosion concerns – LR section 2.1.8, Flow Accelerated Corrosion
- Piping / component supports and water hammer effects – LR section 2.2.2.2, Balance Of Plant Piping (Non-Class 1)
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Environmental qualification of the solenoids for the feedwater regulating valves and bypass valves – LR section 2.3.1, Environmental Qualification
- Feedwater regulating and isolation valve testing and valve closure requirements – LR section 2.2.4, Safety Related Valves and Pumps
- Operation of the condensate and feedwater systems, including isolation features during postulated abnormal and accident scenarios is discussed in LR section 2.8.5, Accident and Transient Analyses
- Condensate and feedwater instrumentation – LR section 2.4.1, Reactor Protection, Safety Features Actuations and Control Systems
- Feedwater isolation valve testing and valve closure, including containment isolation requirements – LR section 2.2.4, Safety Related Valves and Pumps

- Protection against turbine missiles and internal missiles is discussed in LR section 2.5.1.2, Missile Protection

- Condensate storage tank capacity for supplying the auxiliary feedwater volume requirements for cooldown, fire, station blackout and accident conditions – LR section 2.5.4.5, Auxiliary Feedwater System

- Post-accident heat removal requirements – LR section 2.6.1, Primary Containment Functional Design

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the condensate and feedwater systems are within the scope of License Renewal. EPU activities include component replacement / modifications to meet the EPU conditions. These changes do not introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operating the condensate and feedwater systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.5.4.2.3 Results

The following subsections evaluate the specific condensate and feedwater system capabilities while operating at EPU conditions.

General Design Criteria (GDC)

The evaluation of the condensate and feedwater systems capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-4, as described in LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures.

GDC-5 is not applicable to the Ginna Station as it is a single unit installation.

The evaluation of the condensate and feedwater systems capabilities at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in Ginna UFSAR section 3.1.2.4.15.Although the condensate and feedwater systems are not specifically addressed in GDC-44 or in UFSAR section 3.1.2.4.15, the feedwater system does provide an essential isolation function of feedwater flow to the steam generators and for containment isolation. The condensate and feedwater systems provide this capability during accident conditions and are capable of achieving this function considering a single failure, with the addition of enhanced feedwater isolation capability discussed below. The implementation of EPU does not affect the capability of these systems to perform these functions as demonstrated by the system and component evaluation results described below and by the results of the analyses of postulated abnormal

and accident scenarios discussed in LR section 2.8.5, Accident and Transient Analyses and LR section 2.6.1, Primary Containment Functional Design.

System Operating Conditions – Current versus EPU Conditions

The condensate and feedwater system operating conditions; flow, temperature and pressure, were determined from hydraulic modeling of the piping systems and from the current operating (benchmark) and EPU heat balances. The following table compares the current condensate and feedwater system conditions to the EPU conditions:

Table 2.5.5.4-1
Condensate and Feedwater System Operating Conditions

	Current Operating Condition	EPU Operating Condition
Condensate System		
Flow Rate, lbm/hr	4,513,000	5,384,000
Condenser Pressure, inches HgA @ Circ. Water Temperature, °F	1.32 @ 47.5°F CW	1.44 @ 40°F CW
Condensate Pump Discharge Pressure, psia	320	305
Condensate Booster Pump Discharge Pressure, psia	395	383
Condensate Supply Temperature, °F (FW Pump Suction)	349	360
Heater Drain System		
Heater Drain Pump Flow, lbm/hr	2,063,000	2,401.000
Feedwater System		
Flow Rate, lbm/hr	6,576,000	7,785,000
Feedwater Pump Discharge Pressure, psia	1157	1040
Steam Generator Supply Pressure, psia	797	823
Steam Generator Supply Temperature, °F	426	432

Design Pressures / Temperatures – Components and Piping

The design pressures and temperatures of condensate and feedwater components and piping bound the EPU operating conditions.

Feedwater Flow Venturi

Although the design documentation for the feedwater flow venturis specify a design temperature that is 18°F less than the EPU maximum temperature of 432°F, evaluation of the venturi materials and construction show that there is adequate margin in the existing design. The allowable stresses for the venturi material are constant up to a temperature of 650°F which envelopes the EPU operating temperature. Therefore, the venturis are acceptable for EPU operation.

Feedwater Heaters

Feedwater heaters 1A/B through 5A/B were evaluated as acceptable for EPU operation based on their current design, materials, construction, and performance. Current plant operating and inspection data and the predicted EPU heat balance conditions have been reviewed to reach these conclusions. The industry criteria established by Heat Exchange Institute (HEI) standards have been generally used as the guidelines for acceptance, along with specific industry design guidelines.

The feedwater heaters meet the thermal performance requirements of the EPU conditions. Current plant operating data quantifying their performance have been gathered and used to develop the current operating heat balance. This current operating heat balance was then adjusted to predict the plant performance at EPU conditions. These EPU heat balances show that the expected EPU power generation will be achieved thus confirming that the capability of the existing feedwater heaters is adequate for EPU operation.

The design and construction of the feedwater heaters is acceptable for continued operation at EPU conditions with specific monitoring programs in place to evaluate the potential for long term degradation. During the initial EPU design review phase, several potential areas of concern were identified. These were resolved by performing more refined analyses or by incorporating the identified degradation mechanism into a monitoring program so that affected components could be effectively managed. The specifics of the plant monitoring program for erosion / corrosion effects are described in LR section 2.1.8, Flow Accelerated Corrosion. Other monitoring requirements are being implemented as noted below.

- Feedwater heater tube velocities are acceptable at EPU conditions.

- Feedwater heater tube side pressure drops are acceptable at EPU conditions.

- Feedwater heaters shell design pressures are acceptable for EPU conditions. All feedwater heaters at EPU have more than 10 % margin between the operating pressure and the design pressure. For feedwater heaters 3A/B and 4A/B the operating pressure to design pressure margin at EPU is less than the industry guideline of 30 psi. For feedwater heaters 4A/B the operating margin is ~29 psi and is therefore considered acceptable. For feedwater heaters 3A/B the margin is only ~15 psi due to its low operating pressure as compared to design pressure. However, based on the actual feedwater heater shell wall thickness, the maximum calculated working pressure for feedwater heater 3A/B is significantly above the EPU conditions. Thus, these feedwater heater shells are acceptable for EPU operation.

- The steam / water flow through the subcooling zones of the feedwater heaters is acceptable at EPU conditions. The subcooling zone tubes of feedwater heaters 5A/B may experience some vibration and long term wear since the mass flow in the drain cooler section of the feedwater heater exceeds the industry guidelines. The pressure drop through the subcooling zone of heaters 5A/B is also above HEI guidelines which can cause flashing. After uprate the tubes in the subcooling zone of feedwater heater 5A/B will be monitored

periodically using eddy current testing to detect tube wall thinning at the tube mid-spans and at the support plates

- The velocities of some feedwater heater nozzles and internal sections are above the HEI guidelines, manufacturer's guidelines or both, at EPU conditions. No physical changes are considered necessary. However, the long-term effects of the higher velocity in the nozzles and shells will be monitored as described in LR section 2.1.8, Flow Accelerated Corrosion.

- The feedwater heaters shell and tube side relief valves were evaluated. All of the existing relief valve setpoints are acceptable for EPU operation.

- The feedwater heater shell side vents are acceptable for EPU operation. The only modification required for the feedwater heater steam vent due to EPU is to increase the size of the existing vent orifices for feedwater heaters 1A/B to provide sufficient steam vent flow.

Flow Velocities – Piping

Flow velocities through the condensate and feedwater system were calculated at current and EPU conditions. Generally, the flow velocities increased approximately 17% primarily due to the increased flow required by the EPU power level. Velocities generally remain below the industry standard guidelines for these services although there are some pipes whose velocities exceed the guidelines. These individual pipes are evaluated as part of the erosion / corrosion program as described in LR section 2.1.8, Flow Accelerated Corrosion.

Feedwater Regulating Valves

The existing feedwater regulating valves are being modified to provide the required flow at the required pressure drop at EPU conditions. The valve modifications allow the valve to remain less than 85 percent open at EPU normal plant operation so as to provide sufficient control over a range of operating conditions.

The sizing and control capability of the feedwater regulating valves, together with the hydraulic operation of the modified condensate booster pumps and feedwater pumps, provides sufficient flexibility to accommodate plant load rejection transients by providing 96% of rated flow with a 100 psi increase in steam generator pressure.

Condensate and Feedwater Pumps and Supporting Subsystems

The condensate pumps, condensate booster pumps, feedwater pumps and their supporting subsystems will continue to operate successfully during EPU conditions based on the evaluation results, modifications and post-EPU inspections described below:

- The existing condensate pumps operate adequately at EPU with sufficient NPSH, flow and head. The motors continue to provide sufficient motive force for pump operation at EPU conditions.

- The condensate pump discharge pressure alarm and autostart setpoints are being changed to provide sufficient operating margin based on the EPU hydraulic conditions. Refer to LR section 2.4.1, Reactor protection, Safety Features Actuation and Control Systems.

- The existing condensate pump recirculation system provides sufficient flow for condensate pump protection and supplies the minimum flow required by the gland steam condenser and steam jet air ejectors.

- The condensate booster pumps are being modified to provide the necessary EPU flows at the required total dynamic head. The modifications include a new impeller, modifications to the existing pump casing to fit the new impellers and new pump motors.

- The feedwater pump impellers are being replaced to provide the required discharge pressure and flow at EPU. New feedwater pump motors are also being provided.

- The existing feedwater pumps speed increasers will be used at the EPU conditions. Although they will experience increased duty and higher horsepower and operate in their service factor rating due to the modified pump and motor, the speed increaser design is capable of handling the increased load. The speed increasers will be monitored after uprate to verify that the increased load is not causing unacceptable wear.

- The existing low pressure feedwater heater bypass line will operate adequately at EPU conditions to provide sufficient flow and pressure at the feedwater pump suctions. The pressure actuation and reset setpoints for bypass line operation are being changed to match the EPU hydraulic conditions. Refer to LR section 2.41, Reactor protection, Safety Features Actuation and Control Systems.

- The actuation signal for the low pressure feedwater heater bypass valve is being modified to include a time delay of no longer than 10 seconds to lessen the severity of the abnormal events, such as a loss of a condensate pump, condensate booster pump or heater drain pump. Refer to LR section 2.41, Reactor protection, Safety Features Actuation and Control Systems.

- The NPSH margin setpoint, which is calculated from the measured pressure and temperature at the feedwater pump suction, is also being changed to match the EPU operating conditions and the NPSH requirements of the new pump impeller.

- The existing feedwater pump recirculation subsystem provides sufficient flow to meet the modified pump minimum flow requirements at EPU conditions.

- The existing seal water subsystem for the feedwater pumps continues to provide sufficient seal water flow and pressure from the condensate booster pumps. No changes are necessary to the seal water subsystem or the seal

pressure interlock which prevents operation of the feedwater pumps should the seal pressure be insufficient.

Feedwater Isolation Valves

Feedwater isolation is required for a variety of postulated transients and accident events. The current plant design provides for feedwater isolation using the main feedwater regulating valves, associated bypass valves, and the feedwater pump discharge isolation valves. In order mitigate a design basis steam line break in containment at EPU conditions following a failure of the main feedwater regulating valves to go close, faster isolation of feedwater addition to the faulted steam generators needs to occur so as to minimize the mass and energy released to containment. To perform this redundant isolation function at EPU a safety related, automatically actuated operator is being added to the existing manual isolation valve on each feedwater header outside containment in the intermediate building. These new operators are designed to go full close in less than 30 seconds or less to meet the containment steam line break safety analysis requirements discussed in LR section 2.6.1, Primary Containment Functional Design. Since the redundant isolation valves are only required to mitigate steam line breaks inside containment, these actuators do not need to be environmentally qualified for any harsh environments due to postulated high energy line breaks in the intermediate building.

Note that Ginna Plant Technical Specification 3.7.3, Main Feedwater Regulating Valves, Associated Bypass Valves, and Main Feedwater Pump Discharge Valves, requires additional controls on components that provide automatic feedwater isolation. Because of these controls, the main feedwater regulating valves and their bypass valves are maintained safety related. Additionally, to prepare for EPU, Ginna submitted to the NRC a license amendment request dated April 29, 2005 to change this technical specification to substitute a faster acting isolation capability for the Main Feedwater Isolation Valves located outside containment in place of the Main Feedwater Pump Discharge Isolation Valves isolation capability.

Containment isolation is accomplished by the provision of check valves on the feedwater headers outside containment and by normally closed manual valves on branch lines from the headers penetrating containment. The containment isolation requirements are unaffected by EPU and the current plant design features remain acceptable.

The feedwater regulating valves, associated bypass valves and containment isolation check valves will experience a lower differential pressure at EPU conditions versus the current conditions. Thus, these valves will continue to meet the required closure times.

Condensate Coolers

The condensate coolers are being replaced with larger capacity units to meet the increased heat load of the main generator hydrogen coolers while operating at the increased MWe at EPU conditions. The pressure drop and flow requirements of the larger coolers have been included in the hydraulic analysis of the condensate and feedwater systems and the modifications to the system pumps and control valves.

2.5.5.4.3 Conclusions

The Ginna staff has reviewed the effects of the proposed EPU on the condensate and feedwater system and concludes that its assessment has adequately accounted for the effects of changes in plant conditions on the design of the condensate and feedwater system. The Ginna staff concludes that the condensate and feedwater system, with the implementation of the modifications, monitoring and inspections described above, will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related structures, systems, and components. The Ginna staff further concludes that the condensate and feedwater system will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-4, GDC-5, and GDC-44. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the condensate and feedwater system.

2.5.6 Waste Management Systems

2.5.6.1 Gaseous Waste Management Systems

2.5.6.1.1 Regulatory Evaluation

Ginna Nuclear Power Plant, LLC's (Ginna) review of the gaseous waste management system focused on the effects that the proposed EPU may have on previous analyses and considerations related to the gaseous waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in gaseous effluents.

The NRC's acceptance criteria for the gaseous waste management systems are based on:

- 10CFR20.1302 insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values
- GDC-3 insofar as it requires that:
  - Structures, systems, and components important to safety be designed and located to minimize the probability and effect of fires
  - Noncombustible and heat-resistant materials be used
  - Fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety
- GDC-60 insofar as it requires that the plant design include means to control the release of radioactive effluents
- GDC-61 insofar as it requires that systems that contain radioactivity be designed with appropriate confinement
- 10CFR50, Appendix I Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably achievable" (ALARA) criterion

Specific review criteria are contained in SRP Section 11.3.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. UFSAR sections 11.1.1.1 AIF General Design Criterion 70 (1967) and 11.1.1.2 Appendix A General Design Criteria (1972), provide additional description of the design criterion utilized for the station radioactive waste management system. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, the Final Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

As discussed in Ginna UFSAR section 11.3, the gaseous waste management system includes design features to appropriately monitor discharge streams and safety features to preclude releases in excess of 10CFR20 limits and to maintain radioactive discharges to levels as low as is reasonably achievable (ALARA) according to the requirements of 10CFR50, Appendix I. Ginna Station will continue to meet the current licensing basis requirements with respect to GDC 3 as addressed in Ginna UFSAR section 3.1.2.1.3. GDC 3 requires that structures, systems and components be designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions. Ginna UFSAR section 3.1.2.1.3 states that the fire detection and fighting systems of appropriate capacity and capability are provided in accordance with the GDC 3 requirements. The fire protection system and its compliance with 10CFR50 Appendix R is discussed in Ginna UFSAR section 9.5.1. The evaluation of the fire protection system and program for EPU conditions is described in LR section 2.5.1.4.

Ginna Station's current licensing basis with respect to the requirements of GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. GDC 60 requires that the plant design include means to control suitably the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual. On May 5, 1975, the NRC published Appendix I to 10CFR50 which finalized the numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as practical." Rochester Gas and Electric Corporation responded to these requirements in submittals to the NRC in June and October 1976. Implementation of the overall requirements of 10CFR50, Appendix I, as to the utilization of radwaste treatment equipment to ensure that radioactive discharges are as low as is reasonably achievable (ALARA), has been formalized in the Technical Specifications requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual. The Gaseous Waste System is described in UFSAR 11.3. As discussed in UFSAR Section 11.3.3, conformity

by the Ginna Station with the numerical guides on design objectives of 10CFR50, Appendix I (which includes Sections II.B, II.C, and II.D) has previously been demonstrated for gaseous effluents.

Ginna Station's current licensing basis with respect to the requirements of GDC 61 is addressed in Ginna UFSAR section 3.1.2.6.2. The requirements of GDC 61 relevant to the gaseous waste management system require that the system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, and (3) with appropriate containment, confinement, and filtering systems. Ginna UFSAR section 3.1.2.6.1 states that the gaseous waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

- Components are designed and located such that appropriate periodic inspection and testing may be performed
- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Ginna UFSAR chapter 12
- Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems

Additional information concerning the gaseous waste management system is provided in Ginna UFSAR sections 11.3.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's gaseous waste management system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

The above SER discusses gaseous waste management systems in section 2.3.3.4, Waste Disposal Systems. Aging effects, and the programs credited with managing those effects, are described in section 3.3.2.4.4.

2.5.6.1.2 Technical Evaluation

2.5.6.1.2.1 Introduction

The gaseous waste management system is described in Ginna UFSAR section 11.3. Potentially radioactive gases are collected and processed according to physical and chemical properties and radioactive concentrations in accordance with station operating procedures.

The gaseous waste management system design functions are:

- Collect gas from the gas stripper operation, volume control tank purges, sampling system discharges, chemical and volume control system holdup tank cover gas, spent resin storage tank venting, gas analyzer discharge, gas decay tank rupture disk discharge, pressurizer relief tank venting, reactor coolant drain tank venting and charging pump leakoff collection tank venting
- Compress the gas and store it in gas decay tanks
- Sample and analyze the gas prior to release
- Supply nitrogen to various components as a cover gas
- Supply hydrogen to the volume control tank to maintain hydrogen partial pressure as hydrogen dissolves in the reactor coolant

2.5.6.1.2.2 Description of Analyses and Evaluations

The gaseous waste management system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluation compared the existing design parameters of the systems / components with the EPU conditions.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The Ginna Station's radioactive waste management systems were evaluated for their impact on License Renewal evaluations. The gaseous waste management system flow rates, gaseous inventory and process conditions are not changed by the EPU and are within the original design parameters of the system. The increased concentration of radionuclides within the system has no significant effect on the aging of systems / components and there are no system / component modifications necessary. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004. Components of the radioactive waste management systems that are within the scope of License Renewal are described in Section 2.3.3.4 of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, Section 3.3.2.4.4. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the radioactive waste management systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.6.1.2.3 Results

The implementation of power uprate does not significantly increase the inventory of gas normally processed by the gaseous waste management system since the plant system functions are not changing and the assumptions related to volume inputs remain the same. The concentration of activity will remain limited to 100,000 curies. The EPU does not add or change any of the sources of potentially explosive mixtures. The activity and explosive mixture are controlled by Ginna Station procedure.

Potentially radioactive gas is collected from selected systems and components and is directed to the gaseous waste management system. Gases resulting from process operations, gases used for tank cover gas, gases collected during venting, and gases generated in the radio-chemistry laboratory enter the gas decay tanks during all plant operating modes. The implementation of EPU does not add any new sources of potentially contaminated gases, nor does it create any new flow paths or routes that would allow the contamination of uncontaminated gases.

The EPU results in an increase in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the waste disposal systems. The radiological impact of the increased activity in the waste disposal systems is detailed in LR section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the gaseous waste management system at EPU conditions shows concurrence with 10CFR20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This will be demonstrated by the continued compliance post-EPU with the annual dose objective of 10CFR50 Appendix I as discussed in LR section 2.10.1, Occupational and Public Radiation Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the Offsite Dose Calculation Manual.

The evaluation of the gaseous waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-3, insofar as it requires that the plant design includes fire detection and fighting systems of appropriate capacity and capability for the protection of structures, systems and components important to safety. There is no impact to the fire detection and fighting systems due to EPU. There are no new gaseous waste components added as a result of the EPU and the gaseous waste flow rates, gaseous inventory and process conditions are not changed by the EPU. Thus the existing systems retain their compliance to GDC-3.

The evaluation of the gaseous waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC -60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, and are described in the offsite dose calculation manual. Since the design

objectives of Appendix I have previously been demonstrated, the increased radioactive source term due to EPU operation is minimal, and no changes to system design or operation result from EPU, the gaseous waste management system will continue to meet the current licensing basis with respect to 10CFR50, Appendix I. As presented in LR Table 2.10.1-2, the maximum dose due to gaseous effluents following EPU is significantly below the 10CFR50, Appendix I limits.

The evaluation of the gaseous waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC -61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the EPU.

Since conformance to the design objectives of 10CFR50, Appendix I has previously been demonstrated and the increased radioactive source due to EPU operation is minimal, the evaluation of the gaseous waste management system at EPU conditions continues to demonstrate conformance with the requirements of 10CFR50, Appendix I, sections II.B, II.C, and II.D. These criteria set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion as defined in the technical specifications requirements for the radioactive effluent controls program and the Offsite Dose Calculation Manual. Refer to LR section 2.10.1, Occupational and Public Radiation Doses for details.

2.5.6.1.3 Conclusions

The Ginna staff has reviewed the assessment related to the gaseous waste management system. The Ginna staff concludes that the assessment has adequately accounted for the effects of the increase in fission product and the amount of gaseous waste on the ability of the system to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The Ginna staff finds that the gaseous waste management system will continue to meet its design functions following implementation of the proposed EPU. The Ginna staff further concludes that the gaseous waste system will continue to meet the Ginna Station current licensing basis with respect to the requirements of 10CFR20.1302 and 10CFR50, Appendix I, sections II.B, II.C, and II.D and GDC-3, GDC-60, and GDC-61. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the gaseous waste management system.

2.5.6.2 Liquid Waste Management System

2.5.6.2.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) review of the liquid waste management system focused on the effects that the proposed EPU may have on previous analyses and considerations related to the liquid waste management system's design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents.

The NRC's acceptance criteria for the liquid waste management system are based on:

- 10CFR20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement
- 10CFR50, Appendix I, sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion

Specific review criteria are contained in NRC SRP section 11.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. UFSAR section 11.1.1.1, AIF General Design Criterion 70 (1967) and 11.1.1.2 Appendix A General Design Criteria (1972), provide additional description of the design criteria utilized for the station radioactive waste management system. In the late 1970s, the Systematic Evaluation Program (SEP) was initiated by NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

As discussed in Ginna UFSAR section 11.2, the liquid waste management system includes design features to appropriately monitor discharge streams and safety features to preclude

releases in excess of 10CFR20 and to maintain radioactive discharges to levels as low as reasonably achievable (ALARA) according to the requirements of 10CFR50, Appendix I.

Ginna Station's current licensing basis with respect to the requirements of GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. GDC 60 requires that the plant design include means to control suitably the release of radioactive materials in liquid effluents and that sufficient holdup capacity be provided for retention of liquid effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual. On May 5, 1975, the NRC published Appendix I to 10CFR50 which finalized the numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as practical". Rochester Gas and Electric Corporation responded to these requirements in submittals to the NRC in June and October 1976. Implementation of the overall requirements of 10CFR50, Appendix I, as to the utilization of radwaste treatment equipment to ensure that radioactive discharges are as low as is reasonably achievable (ALARA), has been formalized in the Technical Specifications requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual. As discussed in UFSAR Section 11.2.3, conformity by the Ginna Station to the numerical guides on design objectives of 10CFR50, Appendix A (which includes Sections II.A and II.D) has previously been demonstrated for radioactive liquid effluents.

Ginna Station's current licensing basis with respect to the requirements of GDC 61 is addressed in Ginna UFSAR section 3.1.2.6.2. The provisions of GDC 61 relevant to the liquid waste management system require that the system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems. Ginna UFSAR section 3.1.2.6.2 states that the liquid waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

- Components are designed and located such that appropriate periodic inspection and testing may be performed
- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Ginna UFSAR chapter 12
- Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems

Additional information concerning the liquid waste management system is provided in Ginna UFSAR section 11.2.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's liquid waste management system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

With respect to the above SER, liquid waste management systems are included in section 2.3.3.4, Waste Disposal Systems. Aging effects, and the programs credited with managing those effects are described in section 3.3.2.4.4.

2.5.6.2.2 Technical Evaluation

2.5.6.2.2.1 Introduction

The liquid waste management system is described in Ginna UFSAR section 11.2. Potentially radioactive liquids are collected and processed according to physical and chemical properties and radioactive concentrations in accordance with station operating procedures.

The liquid waste management system design functions are:

- Collect liquids from the auxiliary building floor drains, equipment and drains in the residual heat removal sump and auxiliary building sump, and water from resin transfer
- Collect liquids from the intermediate building floor and equipment drains sump, hot shower drains, decontamination area drains, radioactive chem lab drains, hot shop sink drains and the steam generator blowdown tank
- Collect steam generator blowdown from the blowdown flash tank
- Collect liquids from reactor coolant system and various NSSS auxiliary systems sources in the reactor coolant drain tank
- Collect containment floor drains
- Process radioactive liquid waste through a demineralization system
- Collect high conductivity waste from the condensate polisher regenerant wastes
- Collect neutralized regenerant waste from the primary makeup water demineralization system.

The waste disposal system requirements are also discussed in Ginna UFSAR section 9.3.3 Equipment and Floor Drains System.

2.5.6.2.2.2 Description of Analyses and Evaluations

The liquid waste management system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluation determined whether the EPU operating conditions are enveloped by the design parameters of the existing system / components.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The Ginna Station's radioactive waste management systems were evaluated for their impact on License Renewal evaluations. The liquid waste management system flow rates, water inventory and process conditions are not changed by the EPU and are within the original design parameters of the system. The increased concentration of radionuclides within the system has no significant effect on the aging of systems / components and there are no system / component modifications necessary. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004. Components of the radioactive waste management systems that are within the scope of License Renewal are described in Section 2.3.3.4 of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, Section 3.3.2.4.4. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the radioactive waste management systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.6.2.2.3 Results

The implementation of power uprate does not significantly increase the inventory of liquid normally processed by the liquid waste management system since the system functions are not changing and the assumptions related to volume inputs remain the same.

The design flow rates processed by the steam generator blowdown system, LR section 2.1.10 and chemical & volume control system, LR section 2.1.11, are not changed by EPU operating conditions.

Potentially radioactive drainage is collected in tanks and drain sumps from selected systems and components and is directed to the appropriate radwaste processing system. Liquids leaking from process systems, liquids used during cleaning activities, liquid spills from maintenance activities, and liquids generated in the radio-chemistry laboratory enter the equipment and floor drain system during all plant operating modes. The implementation of EPU does not add any new sources of potentially contaminated leakage, nor does it create any new flow paths or routes that would allow the contamination of drainage systems designed for uncontaminated fluids.

The EPU results in an increase in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the waste disposal systems. The radiological impact of the increased activity in the waste disposal systems is detailed in LR section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the liquid waste management system at EPU conditions shows conformance with 10CFR20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This will be demonstrated by the continued compliance post-EPU with the annual dose objective of 10CFR50, Appendix I as discussed in LR section 2.10.1, Occupational and Public Radiation Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the Offsite Dose Calculation Manual. Since the design objectives of Appendix I have previously been demonstrated, the increased source term due to EPU operation is minimal, and no changes to system design or operation result from EPU, the liquid waste management system will continue to meet the current licensing basis with respect to 10CFR50, Appendix I. As presented in LR Table 2.10.1-2, the maximum dose due to liquid effluents following EPU is significantly below the 10CFR50, Appendix I limits.

The evaluation of the liquid waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

The evaluation of the liquid waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the EPU.

The evaluation of the liquid waste management system at EPU conditions demonstrates conformance with the requirements of 10CFR50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion has been formalized in the technical specifications for the radioactive effluent controls program and the Offsite Dose Calculation Manual. Refer to LR section 2.10.1, Occupational and Public Radiation Doses for details.

2.5.6.2.3 Conclusions

The Ginna staff has reviewed the assessment related to the liquid waste management system. The Ginna staff concludes that the assessment has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management system to control releases of radioactive materials. The Ginna staff finds that the liquid waste management system will continue to meet its design functions following implementation of the proposed EPU. The Ginna staff further concludes that the assessment has demonstrated that the liquid waste management system will continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR20.1302; 10CFR50, Appendix I, Sections II.A and II.D; GDC-60, and GDC-61. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the liquid waste management system.

2.5.6.3 Solid Waste Management System

2.5.6.3.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) review of the solid waste management system focused on the effects that the proposed EPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the solid waste management system.

The NRC's acceptance criteria for the solid waste management system are based on:

- 10CFR20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values
- GDC-60, insofar as it requires that the plant design include a means to control the release of radioactive effluents
- GDC-63, insofar as it requires that systems be provided in waste-handling areas to detect conditions that may result in excessive radiation levels
- GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents
- 10CFR71, which states requirements for radioactive material packaging

Specific review criteria are contained in SRP section 11.4.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. UFSAR sections 11.1.1.1, AIF General Design Criterion 70 (1967) and 11.1.1.2, Appendix A General Design Criteria (1972), provide additional description of the design criteria utilized for the station radioactive waste management system. In the late 1970s, the Systematic Evaluation Program (SEP) was initiated by NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna plant were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Ginna Station's current licensing basis with respect to the requirements of GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. The requirements of GDC 60 related to the solid waste management system require that the plant design include means to control suitably the release of radioactive materials and that sufficient holdup capacity be provided. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual. On May 5, 1975, the NRC published Appendix I to 10CFR50 which finalized the numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as practical". Rochester Gas and Electric Corporation responded to these requirements in submittals to the NRC in June and October 1976. Implementation of the overall requirements of 10CFR50, Appendix I, as to the utilization of radwaste treatment equipment to ensure that radioactive discharges are as low as reasonably achievable (ALARA), has been formalized in the Technical Specifications' requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual.

Ginna Station's current licensing basis with respect to the requirements of GDC 63 is addressed in Ginna UFSAR section 3.1.2.6.4. The requirements of GDC 63 related to the solid waste management system require that appropriate systems shall be provided in radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. Ginna UFSAR section 3.1.2.6.4 states that radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity. Appropriate safety actions will be initiated by operator action.

Ginna Station's current licensing basis with respect to the requirements of GDC 64 is addressed in Ginna UFSAR section 3.1.2.6.5. The requirements of GDC 64 related to the solid waste management system require that means shall be provided for monitoring the effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents. Ginna UFSAR section 3.1.2.6.5 states that radioactivity levels contained in the effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for Ginna Station as described in Ginna UFSAR sections 11.5 and 12.5.

The solid waste management system is described in Ginna UFSAR Section 11.4. Solid radioactive waste sources and processing are described in UFSAR Section 11.4.1. As described in UFSAR Section 11.4.3, the Offsite Dose Calculation Manual / Process Control Program outline the method for processing wet solid waste and solidification of liquid waste. It includes applicable process parameters and evaluation methods used to ensure compliance with the requirements of 10CFR20 and 10CFR71 prior to shipment of containers of radioactive waste from the site. All radioactive waste is shipped to a licensed burial site in accordance with applicable NRC, U.S. Department of Transportation, and state regulations.

In addition to the evaluations described above, the Ginna Station's solid waste management system was evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

With respect to the above SER, solid waste disposal management systems are included in section 2.3.3.4, Waste Disposal Systems. Aging effects, and the programs credited with managing those effects associated with waste disposal systems are described in section 3.3.2.4.4.

2.5.6.3.2 Technical Evaluation

2.5.6.3.2.1 Introduction

The solid waste management system is described in Ginna UFSAR section 11.4. The types of solid waste that are produced at Ginna Station in addition to dry active waste are sludge, oily waste, bead resin and spent filters. Potentially radioactive sludge, oily wastes and solids are collected and processed according to physical and chemical properties and radioactive concentrations, in accordance with station operating procedures.

The solid waste management system design functions are:

- Package all solid waste in standard liners and other approved packages for removal to burial or processing facilities
- Provide onsite solid waste storage

Components associated with the solid waste processing system are the spent resin storage tanks and the two onsite solid waste storage facilities. All phases of the solidification process incorporate "as low as is reasonably achievable" (ALARA) design features and operational procedures to ensure that personnel exposure is minimized.

The spent resin storage tanks retain the spent resin which has been used to remove chemical impurities and radioactive contamination from the reactor coolant, the chemical and volume control system, and the spent fuel pool and liquid waste processing system. Normally, a tank is filled over a long period of time, the contents are allowed to decay, and are then emptied prior to receiving any additional resin. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel.

As described in UFSAR Section 11.4.1.3, there are two onsite solid waste storage facilities with a combined capacity sufficient to accommodate approximately 5 years of operation. The two facilities are located northeast of the plant within the security fence. The upper radwaste storage facility typically provides temporary storage for plant solid waste. The high integrity container storage facility is a concrete walled, open topped structure designed as a shadow shield for the storage of spent resin which is stored in a form ready for shipment (i.e., in shielded casks).

2.5.6.3.2.2 Description of Analyses and Evaluations

The solid waste management system and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluation determined whether the EPU operating conditions are enveloped by the design parameters of the existing system / components.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The Ginna Station's radioactive waste management systems were evaluated for their impact on License Renewal evaluations. The solid waste management volumes, storage and handling conditions are not significantly impacted by the EPU. The increased concentration of radionuclides within the system has no significant effect on the aging of systems / components and there are no system / component modifications necessary. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004. Components of the radioactive waste management systems that are within the scope of License Renewal are described in Section 2.3.3.4 of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, Section 3.3.2.4.4. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the radioactive waste management systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the systems. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.6.3.3 Results

The proposed EPU has no significant effect on the generation of solid waste volume from the primary and secondary systems since the system functions are not changing and the assumptions related to volume inputs remain the same. The EPU results in slight increases in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the waste disposal systems. The impact of the increased activity in the waste disposal systems is detailed in LR section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at EPU conditions demonstrates compliance with 10CFR20.1302 since the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This is demonstrated by the continued compliance post-EPU with the annual dose objective of 10CFR50, Appendix I as discussed in LR section 2.10.1, Occupational and Public Radiation Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the Offsite Dose Calculation Manual. No solid waste volumes are expected to leave the site except as properly

packaged and shipped by an authorized carrier to a licensed burial site in accordance with NRC, U.S. Department of Transportation, and state regulations.

The evaluation of the solid waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

The evaluation of the solid waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. This design capability remains unchanged by the EPU. Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity. Refer to LR section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to the requirements of GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. This design capability remains unchanged by the EPU. Radioactivity levels contained in the effluent discharge paths in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for Ginna Station. Refer to LR section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at EPU conditions demonstrates conformance with the requirements of 10CFR71, insofar as the radioactive material packaging accounts for the maximum dose rate allowed on the surface of the container by shielding of the package in which the container is shipped. Packaging, shielding and handling of radioactive material are not changed by EPU; thus, compliance with 10CFR71 is not affected.

2.5.6.3.4 Conclusions

The Ginna staff has reviewed the assessment related to the solid waste management system. The Ginna staff concludes that the assessment has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the solid waste management system to process the waste. The Ginna staff finds that the solid waste management system will continue to meet its design functions following implementation of the proposed EPU. The Ginna staff further concludes that the assessment has demonstrated that the solid waste management system will continue to meet the Ginna Station current licensing basis requirements with respect to 10CFR20.1302, 10CFR71, GDC-60, GDC-63, and GDC-64. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the solid waste management system.

2.5.7 Additional Considerations

2.5.7.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

2.5.7.1.1 Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies (e.g., diesel engine-driven generator sets) of sufficient capacity to perform their safety functions, assuming a single failure. Ginna Nuclear Power Plant LLC's (Ginna) review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks

- GDC-5, insofar as it requires that structures, systems, and components important-to-safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure

Specific review criteria are contained in SRP, Section 9.5.4.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station's emergency diesel engine fuel oil storage and transfer system design relative to conformance to:

- GDC – 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC-4 is also described in the following:
  - Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
  - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
    - Pipe Breaks Inside Containment (SEP Topic III-5.A)
    - Pipe Breaks Outside Containment (SEP Topic III-5.B)
  - Missile Protection (Ginna UFSAR Section 3.5)
- GDC – 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna Station is a single unit installation so there are no shared structures, systems or components.
- GDC - 17 is described in the Ginna UFSAR section 3.1.2.2.8, General Design Criterion 17 - Electrical Power Systems. Additional details that define the licensing basis are described in Ginna UFSAR sections 9.5.4, 8.1.1, and 8.3.1.1.6.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's emergency diesel generator fuel oil and transfer system was evaluated for plant License Renewal. System and system component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The above SER discusses the emergency diesel generator and fuel oil and transfer system in sections 2.3.3.8, Emergency Power. The programs used to manage the aging effects associated with this system are discussed in section 3.4 of the SER.

2.5.7.1.2 Technical Evaluation

2.5.7.1.2.1 Introduction

The emergency diesel generator fuel oil and transfer system is discussed in the Ginna UFSAR section 9.5.4. The emergency diesel engine fuel oil storage and transfer system function is to maintain and provide a minimum inventory of diesel fuel oil to ensure that both emergency diesel generators can operate at their design ratings for 24 hours. This ensures that both emergency diesel generators can supply the design loads of the required engineered safeguards equipment for any loss-of-coolant accident for at least 40 hours, or for one engineered safety feature train for 80 hours. Commercial oil supplies and trucking facilities are available to ensure deliveries of additional fuel oil within 8 hours. Diesel fuel oil storage for each engine is separate and independent of the other, however storage tanks can be manually crosstied.

2.5.7.1.2.2 Description of Analyses and Evaluations

The emergency diesel generator fuel oil and transfer system and its components were evaluated to ensure they are capable of performing their intended function at EPU conditions. The evaluation is based on the system's required design functions and a comparison between the existing equipment ratings and the anticipated operating requirements at EPU conditions.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The emergency diesel engine fuel oil storage and transfer systems are within the scope of License Renewal. EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the diesel fuel oil systems at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.7.1.2.3 Results

The fuel oil inventory required to support the emergency diesel generators has been evaluated. Review of the electrical loads for operation at the EPU conditions indicates that there are no load additions or modifications required to the existing emergency diesel generators. Therefore, there is no impact to the existing emergency diesel generator loading analysis or fuel oil quantity and consumption rate analysis, which bounds the EPU conditions. No additional analysis is required to demonstrate their acceptability and no modifications are required to support EPU operation. The

emergency diesel generator electrical loading is discussed in LR section 2.3.3, AC Onsite Power System.

Since there are no changes, the independence and redundancy features of the system are not impacted by EPU and it continues to meet the Ginna Station current licensing basis with respect to the requirements of GDC 4, 5 and 17. The design for missile protection and protection against dynamic effects associated with the postulated rupture of piping will be maintained. Refer to LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures.

EPU does not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Operating the fuel oil storage and transfer system at EPU conditions does not add any new or previously unevaluated materials to the system. No new aging effects requiring management are identified.

2.5.7.1.3 Conclusion

The evaluation concluded that the emergency diesel generator fuel oil and transfer system will continue to function as designed and continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4, GDC-5 and GDC-17 following implementation of the proposed EPU. The fuel oil and transfer system will continue to provide an adequate amount of fuel oil to allow the emergency diesel generators to meet the onsite power requirements at EPU conditions.

2.5.7.2 Light Load Handling System (Refueling)

2.5.7.2.1 Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The Ginna staff=s review focused on the effects of the new fuel on system performance and related analysis. The NRC's acceptance criteria for the LLHS are based on:

- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection, and

- GDC-62 insofar as it requires that criticality be prevented.

Specific review criteria are contained in SRP Section 9.1.4 and guidance provided in Matrix 5 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR) section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR section 3.1.1 and 3.1.2. In the late 1970's the Systematic Evaluation Program (SEP) was initiated by the NRC to review the design of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of the Ginna Station control of new and spent fuel movement relative to conformance to:

- GDC-61 is described in Ginna UFSAR section 3.1.2.6.2, "Fuel Storage and Handling and Radioactivity Control." As described in this UFSAR section, the fuel handling system is designed to ensure adequate safety under normal operation and postulated accident conditions. The fuel handling system is discussed in UFSAR section 9.1.2.1.7, "Fuel Handling System," and section 9.1.4, "Fuel Handling Systems." The ability of the spent fuel storage pool to accommodate heavy bearing loads is discussed in UFSAR section 9.1.2.1.13, "Bearing Loads on Pool Liner."

- GDC-62 is described in Ginna UFSAR section 3.1.2.6.3, "Prevention of Criticality in Fuel Storage and Handling." As described in this UFSAR section,

criticality in new and spent fuel storage areas is prevented both by physical separation of fuel assemblies and by the presence of borated water in the spent fuel storage pool. The fuel handling system is discussed in UFSAR section 9.1.2.1.7, "Fuel Handling System," and section 9.1.4, "Fuel Handling Systems."

In addition, as a result of the NRC review of load-handling operations at nuclear power plants, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was issued by the NRC on July 1, 1980. NUREG-0612 provides guidance and methods acceptable to the NRC for the handling of spent fuel, fuel in the core, and heavy loads which would prevent load handling accidents that could result in the potential release of radioactive material in excess of regulatory limits. Following the issuance of NUREG-0612, Generic Letter (GL) 80-113, "Control of Heavy Loads," dated December 22, 1980, was sent to all plants requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses to the NRC GL and subsequent NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," are discussed in UFSAR section 9.1.5, "Control of Heavy Loads." As indicated in UFSAR section 9.1.5, the Ginna submittals concerning control of heavy loads have been reviewed by the NRC and are considered acceptable.

NUREG-0612 defines a heavy load as a load which is greater than the weight of a single spent fuel assembly and its handling tool. Although the fuel handling tools do not handle heavy loads according to the NUREG-0612 definition, the Ginna response to the NRC dated 3/2/83 committed Ginna to meeting NUREG-0612 requirements for the manipulator crane, SFP, and Auxiliary Building cranes. The gripper, new fuel handling tool, and spent fuel handling tools are in the load path for these cranes, therefore these components are maintained in an augmented quality program.

Ginna UFSAR sections that address design features of the fuel handling systems include:

- 1.8.1.13, Safety Guide 13 -Fuel Storage Facility Design Basis
- 9.1.1, New Fuel Storage
- 9.1.2, Spent Fuel Storage
- 9.1.4, Fuel Handling Systems
- 9.1.5, Control of Heavy Loads

In addition to the evaluations described in the UFSAR, the Ginna fuel handling systems were evaluated for the Ginna Station License Renewal. The fuel handling system was evaluated separate of the cranes, new and spent fuel storage racks, and the SFP and cavity liners. Portions of the fuel handling system are considered within the scope of license renewal. The evaluation of the fuel handling system and the subsequent review and conclusions are discussed in section 2.3.3.16, "Fuel Handling," of the License Renewal SER NUREG-1786. The cranes, new and spent fuel storage racks, SFP, and cavity liners are discussed in SER section 2.3.3.3, Spent Fuel Pool Cooling and Fuel Storage.

2.5.7.2.2 Technical Evaluation

Introduction

The Fuel Handling system is described in UFSAR section 9.1.4, "Fuel Handling Systems." The fuel handling systems provide a safe and effective means for transporting and handling reactor fuel from the time the fuel reaches the plant in an unirradiated condition until it is placed in the spent fuel pool racks to await final long term storage. The fuel handling system is divided into the two categories of fuel storage and fuel handling. The fuel handling category covers the facilities other than storage, the equipment and tools used to refuel the reactor and includes the following components:

- Reactor Cavity
- Refueling Canal
- Auxiliary Building Crane
- New Fuel Elevator
- Spent Fuel Pool Bridge
- Fuel Transfer System
- Manipulator Crane
- Reactor Vessel Head Lifting Device
- Reactor Internals lifting Device
- Upper Internals Storage Stand

The fuel handling tools are used to prevent close operator exposure to the fuel. Several other specialized tools are also available to aid the operators in performing specific MODE 6 (Refueling) functions and are comprised of the following:

- New Fuel Assembly Handling Tool
- Spent Fuel Handling Tool
- Burnable Poison Rod Assembly Handling Tool
- Control Rod Drive Shaft Tool
- Thimble Plug Handling Tool
- Irradiation Sample Handling Tool
- Stud Tensioners

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool (SFP) and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the Spent Fuel Pool fuel storage area, administrative controls and geometric constraints ensure that the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. Also, crane interlocks and administrative controls prevent carrying heavy objects, such as a spent fuel transfer cask, over the fuel assemblies in the storage racks. In addition, administratively, only one fuel assembly can be handled at a given time over storage racks containing spent fuel. The motions of the cranes which move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or spent fuel pool. The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

With respect to EPU, the changes to the fuel handling systems are associated with the transition from Westinghouse 14x14 OFA fuel to Westinghouse 14x14 422V+ fuel. With respect to the fuel handling systems, the significant differences between the OFA and 422V+ fuel is the design of the assembly top nozzle and fuel assembly weight.

Evaluation of the effect of the transition to the 422V+ fuel on the new and spent fuel storage areas is described in LR section 2.6, "Fuel Storage." The effects of EPU with respect to the radiological consequences of a fuel handling accident are described in LR section 2.9.8, "Radiological Consequences of Fuel Handling Accidents". The effects of the transition to the 422V+ fuel on the fuel handling tools are described below.

Description of Analyses and Evaluations

This section of the report provides an assessment of the existing analysis for fuel handling equipment and evaluates the impact of the new fuel weight on that equipment. This review focused on the effects of the proposed EPU fuel assembly changes and its impact on the fuel handling equipment in place at Ginna Station as discussed in UFSAR section 9.1.4.

Fuel Weight Comparison

According to Ginna Spent Fuel Pool Licensing Re-rack Report, 51-1258768 Rev 01, 3-31-97, the maximum weight for existing spent fuel assemblies stored in the spent fuel pool, including weight of the rod cluster control assembly (RCCA), is 1450 lbs. Westinghouse provides the weight of a 422V+ fuel assembly including a RCCA as 1402 lbs (1262 lbs. +140 for control components). Since existing analysis utilizes a weight which bounds the weight of the new fuel assembly, the fuel handling equipment is acceptable for use with the 14x14 422V+ fuel to be used for EPU.

Handling Equipment

For the Ginna station power EPU, a Ginna specific version of Westinghouse 422V+ 14 x14 fuel will be used starting with the uprate in 2006. This fuel has a shorter standard top nozzle instead of the unique top nozzle previously used at Ginna. The change to a shorter top nozzle necessitates changes to the following fuel handling devices and tools:

- The refueling machine gripper will be replaced with a functionally equivalent gripper that can interface with both OFA and 422V+ fuel.

- The lower 3-foot section of the RCCA change fixture will be replaced due to a 0.25-inch interference with an RCCA hub in the 422V+ fuel.
- New spent fuel and new fuel handling tools will be procured which are compatible with both types of fuel.
- The RCCA stop on the fuel conveyor car will be adjusted to prevent interference when transferring a 422V+ assembly with RCCA between containment and the spent fuel pool.
- The portable RCCA change tool will be modified to ensure compatibility with the OFA and 422V+ fuel.

The new spent fuel handling tools are materially and structurally similar to the existing tools and are designed to accommodate handling the existing and new fuel design. Following the changes to the fuel handling tools, the length of the tools will remain unchanged, therefore there is no reduction in shielding and radiation protection provided to personnel performing fuel handling operations.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The majority of the fuel handling system is not considered within the scope of license renewal. This includes the new fuel elevator, the underwater conveyor car, the control equipment for the fuel manipulator cranes, fuel and reactor internals handling tools, control equipment for safety interlocks, and essential valves and air tubing. The fuel handling license renewal boundary only includes cranes, hoists, or lifting devices categorized under NUREG-0612. The evaluation of the fuel handling system and the subsequent review and conclusions are discussed in section 2.3.3.16, "Fuel Handling," of the License Renewal SER, NUREG-1786. The cranes, hoists, and lifting devices associated with fuel handling are discussed as an equipment group in SER section 2.3.3.11, "Cranes, Hoists, and Lifting Devices." The crane rails and supports that interface with building structural members are evaluated within the building that contains them.

EPU activities are not adding any new components within the existing license renewal scoping evaluation boundaries nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the fuel handling system components at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus, no new aging effects requiring management are identified.

2.5.7.2.3 Results

The fuel handling systems for both new (unirradiated) fuel and spent (irradiated) fuel have been evaluated for the impact of EPU operation. In order to support EPU operations, Ginna will transition from Westinghouse 14x14 OFA fuel to Westinghouse 14x14 422V+ fuel.

The existing analysis utilizes a limiting weight for fuel handling equipment which bounds the weight of the new fuel assembly, therefore the fuel handling equipment is acceptable for use with the 14x14 422V+ fuel to be used for EPU.

The Westinghouse 422V+ 14 x14 fuel, to be used starting with the uprate in 2006, has a shorter standard top nozzle than previously used at Ginna which necessitates changes to fuel handling devices and tools. The new spent fuel handling tools are materially and structurally similar to the existing tools and are designed to accommodate handling the existing and new fuel design. There is no reduction in shielding and radiation protection provided to personnel performing fuel handling operations.

2.5.7.2.4 Conclusions

The Ginna staff has reviewed its assessment of the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that Ginna has adequately incorporated the effects of the new fuel in the analysis. Based on this review, Ginna further concludes that the LLHS will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-61 and GDC-62 for radioactivity releases and prevention of criticality accidents. Therefore, Ginna finds the proposed EPU acceptable with respect to the LLHS.

2.5.8 Additional Review Areas (Plant Systems)

2.5.8.1 Circulating Water System

2.5.8.1.1 Regulatory Evaluation

The circulating water system provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. Ginna Nuclear Power Plant, LLC's (Ginna) review of the circulating water system focused on changes to the amount of heat absorbed by the system from increased heat rejection from the condenser and other turbine cycle heat exchangers due to the higher EPU power level. The impact of this increased heat on the circulating water components was evaluated to ensure that the system accomplishes its design functions after implementation of EPU. Specific temperature limits for the circulating water discharged to Lake Ontario are contained in the site New York State Pollutant Discharge Elimination System (SPDES) permit.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to conformance to General Design Criterion 2 is addressed in Ginna UFSAR Section 3.1.2.1.2, "General Design Criterion 2 – Design Bases for Protection Against Natural Phenomena." As noted in that section, flood protection measures are discussed in Ginna UFSAR Section 3.4. Protection against the effects of circulating water pipe breaks is also addressed in UFSAR section 3.6.2.4.8.2.

As stated in Ginna UFSAR section 10.6, Circulating Water System, the function of the circulating water system is to provide a reliable supply of water to condense the steam exhausted from the low-pressure turbines. The circulating water system is a non-seismic piping system whose primary function is to remove heat from the steam cycle via the main condensers.

Other Ginna UFSAR sections that address the design features and functions of the auxiliary feedwater system include:

- UFSAR sections 1.2.6, Waste Disposal System and 11.2.1, Design Bases - Liquid Waste Management System, which discuss the release of liquid wastes via the circulating water discharge canal

- UFSAR section 3.4, which discusses the plant design features provided for protection against flooding, including floods due to a failure in the circulating water system.

In addition to the licensing bases described in the UFSAR, the circulating water system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The above SER discusses the circulating water system in section 2.3.3.14, Circulating Water. Aging effects, and the programs credited with managing those effects, are described in section 3.3.2.4.14.

2.5.8.1.2 Technical Evaluation

2.5.8.1.2.1 Introduction

The circulating water system is discussed in Ginna UFSAR section 10.6. The function of the circulating water system is to provide a reliable supply of water to condense the steam exhausted from the low-pressure turbines and cool miscellaneous heat exchangers and to provide dilution of liquid discharges prior to release. The water source and sink for the circulating water system is Lake Ontario. The circulating water system consists of an intake structure, an inlet tunnel, four traveling screens, two circulating water pumps, a discharge tunnel and associated piping, valves and expansion joints. The pumps are located in the screen house. Lake water is supplied to the screen house via an intake structure on the bottom of Lake Ontario and an intake tunnel which passes under Lake Ontario. The traveling screens are located in the screen house upstream of the circulating water pump inlets. The circulating water is pumped from the screen house through the main condensers to remove heat rejected from the steam cycle and through the condensate cooler to absorb heat rejected from the main generator via the hydrogen cooler. The water is then discharged back to the lake via the discharge tunnel.

2.5.8.1.2.2 Description of Analyses and Evaluations

The circulating water system and its components were evaluated to ensure they are capable of performing their intended function at EPU conditions. The circulating water system was evaluated for a NSSS power level of 1781 MWt. The evaluation reviewed the circulating water system to determine whether the existing flow rate is capable of removing the higher steam cycle heat duty at EPU conditions.

The increased heat rejection to the circulating water system from the turbine cycle heat loads at EPU conditions raises the system operating temperature downstream of the condenser and other affected heat exchangers. Heat loads during normal plant full power operation and during plant load changes which cause a steam dump directly to the condenser were utilized in the evaluation.

The existing component design temperatures and pressures were reviewed to confirm that the higher operating temperatures are bounded by the component designs. The higher circulating water outlet temperatures were also reviewed against the site New York State Pollutant Discharge Elimination System permit.

Other evaluations related to the circulating water system, piping and components are included in the following LR section:

- Liquid waste effluent discharge to the discharge canal - LR section 2.5.6.2, Liquid Waste Management System
- Protection against flooding due to a failure in the circulating water system - LR section 2.5.1.1, Flooding
- Piping and component supports - LR section 2.2.2.2, BOP (Non-Class 1)
- Environmental impact / SPDES review - Supplemental Environmental Report
- Heat removal and cooling of the main condenser – LR section 2.5.5.2, Main Condenser

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The circulating water system was evaluated for the Ginna Station License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in License Renewal Safety Evaluation Report, NUREG-1786. The portions of the circulating water system that interface with the service water system are within the scope of License Renewal and are evaluated with the service water system. The service water system is described in SER section 2.3.3, Auxiliary Systems. Aging effects, and

the programs used to manage the aging effects associated with service water, are discussed in section 3.3 of the SER. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the circulating water system at EPU conditions do not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.5.8.1.2.3 Results

During normal operation, the circulating water system provides a nominal flow of approximately 334,600 gpm to the condenser. This flow rate is adequate to remove the increased heat rejected by the steam cycle at EPU as shown by the EPU heat balances that predict the expected plant electric power output at the EPU NSSS power level of 1781 MWt. Since no physical changes are required in the circulating water system due to EPU, there is no required change to the system flow capability. Therefore, the current circulating water system flow rate is acceptable for the EPU conditions.

The outlet temperature of the circulating water system is higher at EPU conditions due to the higher heat load being rejected to the condenser. With the lake temperature at its design maximum operating temperature of 80°F, the discharge temperature of the circulating water increases from 102°F to 104.6°F during normal 100% power operation. The temperature rise across the condenser at EPU is~ 24.6°F. Conservatively assuming operation of all eight steam dump valves in the turbine by-pass system with no runback in reactor power, the maximum possible CW outlet temperature is ~110°F with a corresponding maximum possible condenser temperature rise of ~ 30°F. Since a full actuation of the turbine by-pass system at full power would be accompanied by a power reduction due to control rod runback, this bounding condition is a short duration event and will not exceed the circulating water maximum discharge temperature, computed on a daily average basis.

The maximum circulating water discharge temperature for normal operation at EPU is above the existing Ginna SPDES discharge limit of 102°F. The temperature rise across the circulating water system during normal operation is below the existing SPDES permit limit of 28°F.

Since the existing SPDES maximum discharge temperature limit could be exceeded at EPU conditions, Ginna Station has requested a change to the SPDES permit to increase the discharge temperature limit from 102°F to 106 °F. Additionally, Ginna is requesting the SPDES permit maximum temperature rise to be increased from 28°F to 35 °F, and to increase the size of the mixing zone in Lake Ontario from 320 acres to 360 acres.

The design temperatures and pressures of the circulating water piping and components are acceptable for EPU operating conditions, although the design temperature of 100°F of the circulating water discharge piping downstream of the condenser could be exceeded when exposed to the maximum EPU temperature of 104.6°F during normal 100% power operation, or steam dump temperature of 110.2°F. An evaluation determined that the discharge pipe and fittings are acceptable since they are constructed of carbon steel material, ASTM A283-58, and

the joints are butt welded. The concrete discharge tunnel is also acceptable for the EPU temperature since the maximum discharge temperature is well below the 150°F limit for concrete.

EPU operating pressures within the system do not change since the current flow rates are acceptable and the circulating pumps continue to operate at the same flow / discharge head at EPU conditions. Therefore, the existing component design pressures are unaffected by the EPU conditions.

The current capacity of the circulating water vacuum priming system is acceptable for EPU operation. The circulating water flow rate does not change at EPU conditions. Therefore, the only impact on the circulating water air release rate is the temperature increase downstream of the condenser. The current capacity of the vacuum priming system envelopes the slight increase in the air release rate.

Liquid wastes are released, after appropriate cleaning and filtering, to the circulating water discharge canal with appropriate monitoring. See LR section 2.5.6.2, Liquid Waste Management System for details.

2.5.8.1.3 Conclusions

Ginna has assessed the effects of the proposed EPU on the circulating water system. Ginna concludes that the evaluation adequately accounts for the effects of the proposed EPU on the system's capability to remove heat rejected from the turbine cycle and auxiliary heat exchangers. The current design of the circulating water system provides a reliable supply of water at EPU conditions to condense the steam exhausted from the low-pressure turbines and to cool miscellaneous heat exchangers. The current design of the system and its components accommodates the higher condenser duty and higher temperatures at EPU conditions. The Ginna Station has requested a revision to the SPDES permit to incorporate the higher EPU temperatures. Additional information is provided in Attachment 8 to this EPU application, "Supplemental Environmental Report". Therefore, the proposed EPU is acceptable with respect to the circulating water system.

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

2.6.1.1 Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident.

The Ginna Nuclear Power Plant, LLC (Ginna) review covered the pressure and temperature conditions in the containment due to a spectrum of postulated loss-of-coolant accidents (LOCAs) and secondary system line-breaks.

The NRC's acceptance criteria for primary containment functional design are based on:

- GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment

- GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA

- GDC-38, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels

- GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions

- GDC-64, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents

Specific review criteria are contained in the SRP section 6.2.1.1.A.

Ginna Current Licensing Basis

As noted in *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predates those provided in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to:

- GDC-16 is described in Ginna UFSAR section 3.1.2.2.7, General Design Criterion 16 – Containment Design. As described in this UFSAR section, the containment building is a reinforced concrete structure which is designed for an internal pressure of 60.0 psig. Periodic leak rate measurements are conducted as defined in the Technical Specifications to ensure the containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

- GDC-50 is described in Ginna UFSAR section 3.1.2.5.1, General Design Criterion 50 – Containment Design Basis. As described in this UFSAR section, the containment structure was designed with margin to accommodate the temperature and pressure conditions associated with the loss-of-coolant accident and main steam line break, without loss of function. SEP review of the containment functional design is described in UFSAR section 6.2.1.2.1. The SEP containment response analysis was subsequently revised to account for steam generator replacement as described in UFSAR section 6.2.1.2.2. A more detailed discussion of the evolution of the containment analysis licensing basis is provided below.

- GDC-38 is described in UFSAR section 3.1.2.4.9, General Design Criterion 38 – Containment Heat Removal. As described in this UFSAR section, two systems based on different principles are provided to remove heat from the containment following postulated loss-of-coolant or steamline break accidents in order to maintain the containment internal pressure below its design value. As described in UFSAR section 6.2.2, this post-accident heat removal function is provided by the containment spray system operating in conjunction with the

containment recirculation fan coolers (CRFCs). The CRFCs are sized to remove sufficient heat from the containment following design basis accidents and subsequent pressure transient to keep the containment pressure from exceeding its design value. Working in conjunction with the containment spray system, the CRFCs reduce the post-accident containment pressure to near atmospheric within 24-hours following the accident, taking into account the most limiting single failure.

- GDC-13 is described in UFSAR section 3.1.2.2.4, General Design Criterion 13 – Intrumentation and Control. As described in this UFSAR section, instrumentation and controls essential to avoid any undue risk to the health and safety of the public are provided to monitor variables and systems over their anticipated ranges for both normal operation and postulated accident conditions. Process and containment instrumentation measure temperatures, pressure, flow, and levels in the reactor coolant system, steam systems, containment, and other auxiliary systems. Detailed discussion of instrumentation and control systems is provided in UFSAR chapter 7. A comparison of Ginna's post accident monitoring instrumentation with the provisions of Regulatory Guide 1.97, Revision 3, is provided in UFSAR Table 7.5-1.

- GDC-64 is described in UFSAR section 3.1.2.6.5, General Design Criterion 64 – Monitoring Radioactivity Releases. As described in this UFSAR section, the containment atmosphere is continually monitored during normal and transient station operation using the containment particulate and gas monitors. In addition, radioactivity levels in the facility effluent discharge paths and environs are continually monitored during normal operation and accident conditions by the Ginna Station radiation monitoring system as described in UFSAR sections 11.5 and 12.5. Release limits consistent with regulatory requirements are established in the Offsite Dose Calculation Manual.

In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. As described in UFSAR section 6.2.1.2.1, the SEP review of the containment analysis performed by the NRC compared the Ginna analysis to the criteria used by the NRC for licensing of new facilities in the early 1980s. Two differences in containment design methodology from the then current criteria were identified. First, for the LOCA analysis, the cold-leg pump suction break location, the core reflood phase of M&E release, and the release of secondary system energy were not considered. Second, the main steam line break (MSLB) analysis was not performed in detail.

To assess the significance of these two differences, the NRC performed independent LOCA and MSLB analyses. It was concluded, overall, that in the original LOCA analysis the design-basis pressure enveloped the NRC results, and the design-basis temperature profile exceeded the NRC results, except in the range between 10,000-20,000 seconds after the design-basis event. In this range, the design-basis temperature profile was revised for the purposes of environmental qualification of equipment (see UFSAR section 3.11.3.1.1). Regarding the MSLB analysis, the NRC concluded that the calculated peak containment pressure is less than the containment design pressure and the temperature profile as revised for the LOCA case was acceptable for use in equipment qualifications.

In addition to the SEP review of containment functional design, the NRC reviewed the Ginna response to NRC IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition." The NRC concluded that there was no potential for a containment overpressure event at Ginna resulting from the MSLB issues identified in the IEB 80-04.

Subsequent to the SEP review, the Ginna containment analysis licensing basis was modified with the advent of steam generator replacement as described in UFSAR section 6.2.1.2.2. The containment response portion of the SEP analysis was revised to incorporate the characteristics of the replacement steam generators and correct inconsistencies between the SEP analysis modeling of containment and the containment modeled in UFSAR section 6.2.1.2.3 for secondary system pipe breaks. The M&E release out of the break calculated by the SEP analysis is modeled for the higher primary side volume and higher operating pressure of the replacement steam generators. This was used as input to the containment response analysis. This previous analysis concluded the limiting secondary system pipe break was a steamline break of 1.4 ft^2 occurring at 100% power and assuming the failure of one main feedwater regulating valve to close. Peak containment pressure remained below the containment design value.

2.6.1.2 Technical Evaluation

2.6.1.2.1 Introduction

2.6.1.2.1.1 Loss-of-Coolant Accident

The evaluation of the design basis LOCA event relative containment peak pressure and temperature response was completed to demonstrate the acceptability of the containment heat removal system to mitigate the consequences of a LOCA inside containment and to support the EPU program operation. This evaluation is documented in the subsections below.

The containment response analysis demonstrates the acceptability of the containment heat removal systems to mitigate the consequence of a large LOCA inside containment. The impact of LOCA M&E releases on the containment pressure and temperature are addressed to assure that the containment pressure and temperature remain below their respective design limits. The

systems must also be capable of maintaining the Environmental Qualification (EQ) parameters to within acceptable limits at the EPU program conditions.

The Ginna LOCA containment response analysis considered a spectrum of cases as discussed in LR section 2.6.3.1, M&E Release Analysis for Postulated Loss of Coolant Accidents. The cases address break location, and postulated single failure (minimum and maximum safeguards). Only the limiting cases, which address the containment peak pressure case and limiting long-term EQ case, are presented herein.

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC. GOTHIC version 7.2 was used for the LOCA containment response analysis. The GOTHIC Technical Manual (Reference 2) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (Reference 3) provides a comparison of the solver results with both analytical solutions and experimental data.

The GOTHIC containment modeling for Ginna is consistent with the recent NRC approved Kewaunee evaluation model (Reference 5). The latest code version is used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC (Reference 5) for use in Kewaunee containment analyses with the condition that mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). The GOTHIC containment modeling for GINNA has followed the conditions of acceptance placed on Kewaunee. Kewaunee and Ginna both have large dry containment designs with similar sized containment volumes and active heat removal capabilities. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes (Reference 6). Version 7.2 is used consistent with the restrictions identified in Reference 5; none of the user-controlled enhancements added to version 7.2 were implemented in the Ginna containment model. A description of the Ginna GOTHIC model is provided in this LR section 2.6.1.2.3, Description of Analyses and Evaluations.

The Ginna model thermal-hydraulic response was bench-marked against the double-ended pump suction break case presented in the Ginna UFSAR, Section 6.2. The results show that the GOTHIC 7.2 model agrees quite well with the UFSAR results. The peak pressure and gas temperature (with GOTHIC version 7.2) are slightly higher. This is attributed mainly to the paint layers and gap resistances that were added to the heat sinks.

2.6.1.2.1.2 Main Steamline Break

Steamline ruptures occurring inside the reactor containment may result in significant releases of high-energy fluid to the containment environment that could produce high pressure conditions for extended periods of time. In order to assess the containment response to steamline rupture, the analysis considered a spectrum of cases that vary the initial power condition and the postulated single failure.

The mass and energy release from a spectrum of steamline breaks was reanalyzed for EPU conditions. The containment pressure response to a steamline break was also analyzed for the EPU program. A description of this reanalysis, including results, was submitted to the NRC in a license amendment request dated April 29, 2005 (Reference 1), as supplemented by Reference 9.

2.6.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

2.6.1.2.2.1 Loss-of-Coolant Accident

The major modeling input parameters and assumptions used in the Ginna containment evaluation model for the LOCA event are identified in this section. The assumed initial conditions and input assumptions associated with the fan coolers and containment sprays are listed in Table 2.6.1-1. The containment recirculation fan cooler (CRFC) heat removal capability data used is presented in Table 2.6.1-2. The primary function of the residual heat removal system (RHR) is to remove heat from the core by way of the emergency core cooling system (ECCS). The recirculation system alignment is outlined in Table 2.6.1-3. The containment structural heat sink input is provided in Table 2.6.1-4, and the corresponding material properties are listed in Table 2.6.1-5.

The Ginna current Licensing Basis is presented in the Ginna Updated Final Safety Analysis Report (UFSAR), Section 6.2.1.2, Containment Integrity Evaluation. GINNA UFSAR Tables 6.2-1 through 6.2-4 present the major assumption used to support the UFSAR analysis. The LOCA containment analysis described herein utilized revised input and assumptions in support the Ginna EPU program, while addressing analytical conservatisms. The following summarized assumptions are areas where known differences exist between the current licensing analysis and the EPU Program containment integrity analysis.

1. All exposed concrete and carbon steel surfaces areas are conservatively assumed to have an overcoat and primer coatings (reduces heat transfer through heat sink). The UFSAR licensing basis assumed no surfaces were painted.

2. To simulate the gap between insulation, steel and concrete, a thin air gap was modeled between these layers. The air gap resistance layer was not modeled for the Ginna UFSAR licensing basis,

3. For the GOTHIC LOCA model, all of the sump heat sinks (3, 4, 5, 6 and 7) are considered to be submerged. The UFSAR model assumed the basement floor as submerged with the sump wall split between liquid and vapor. The conductors assumed submerged are essentially insulated from the vapor after the pools develop.

4. The containment initial pressure was assumed to be 15.7 psia for the EPU program versus 16.7 psia in the Ginna UFSAR licensing basis analysis.

5. Non condensable accumulator gas addition is modeled in the EPU Program model; no accumulator gas addition is considered in the UFSAR current licensing analysis.

6. Current Ginna calculations support a containment high-high pressure setpoint for containment spray pump initiation of 32.5 psig. The analysis value modeled was 33.5 psig to provide more margin for instrument uncertainty.

7. A sump recirculation model i.e., modeling coupled residual and component cooling heat exchangers, and service water piping was developed for the EPU Program.

Design Basis Accident

The Ginna LOCA containment response analysis considered a spectrum of cases as discussed in LR section 2.6.3.1, M&E Release Analysis for Postulated Loss of Coolant Accidents. The cases address break location, and postulated single failure (minimum and maximum safeguards). Only the limiting cases, which address the containment peak pressure case and limiting long-term EQ case, are presented herein. The LOCA pressure and temperature response analyses were performed assuming a loss of offsite power and a worst single failure (loss-of-one emergency diesel generator [EDG] that is, loss-of-one containment cooling train).

The limiting minimum safeguards case was based on a diesel train failure (loss of one cooling train) i.e., the active heat removal is:

- One containment spray pump, injection-phase only
- Two CRFCs
- One RHR pump and heat exchanger (HX)
- One component cooling water pump and two CCW HXs
- Two service water pumps

The calculation for the double-ended pump suction (DEPS) case was performed for a 30-day transient in support of EQ. The sequence of events for the containment peak pressure case, a double-ended hot leg break (DEHL) and the DEPS (EQ transient) is shown in Tables 2.6.1-6 and 2.6.1-7, respectively.

The Ginna GOTHIC containment evaluation model consists of a single-lumped parameter node; the DLM heat and mass transfer option is used.

The containment response for design basis LOCA containment integrity is an ANS Condition IV event, an infrequent fault. The relevant requirements to satisfy Nuclear Regulatory Commission acceptance criteria are as follows:

- GDC-16 and -50: In order to satisfy the requirement of GDC-16 and -50, the peak calculated containment pressure should be less than the containment design pressure of 60 psig.

- GDC-38: In order to satisfy the requirement of GDC-38, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for containment leakage assumptions as affecting doses at 24 hours.)

Note that although Ginna is not a SRP plant, for completeness, the SRP long-term cooling criterion is also examined.

The containment design pressure for Ginna is 60 psig. The containment design temperature is 286°F.

2.6.1.2.2.2 Main Steamline Break

Plant input assumptions for the steamline break analysis (identified in section 4.4 of the containment response analysis of Reference 1, as supplemented by Reference 9) are the same as, or slightly more restrictive, than in the current licensing basis analysis. The GOTHIC containment modeling for Ginna followed the conditions of acceptance placed on Kewaunee for which a power uprate was approved on February 27, 2004. Kewaunee and Ginna both have large, dry containments with similar containment volumes and active heat removal capabilities.

The major input value assumptions for the containment that are used in the containment analysis are listed in Tables 1 through 3 of the containment response analysis of Reference 1 which are similar to UFSAR Tables 6.2-9 through 6.2-11, respectively. The major input assumptions for secondary-side systems are presented in section 4.2 of the containment response analysis of Reference 1. Reactor coolant system assumptions are presented in section 4.3 of the containment response analysis of Reference 1.

The acceptance criterion is to demonstrate through analysis that the most restrictive steamline break will not result in a peak containment pressure in excess of the containment design pressure of 60.0 psig.

2.6.1.2.3 Description of Analyses and Evaluations

2.6.1.2.3.1 Loss-of-Coolant Accident

Noding Structure

The Ginna GOTHIC containment evaluation model for the LOCA event consisted of one volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled by a boundary condition. The recirculation system model uses GOTHIC component models for the RHR and CCW HXs and the CCW pumps. Recirculation flow from the sump is modeled using a boundary condition.

Volume Input

Gothic requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 1,000,000 ft^3 was used. The height, diameter, and elevation input values are not important for this single-volume containment model, so standard values of 100 feet, 10 feet, and 0 feet were used respectively.

A conservatively calculated pool surface area is used to model interfacial heat and mass transfer to liquid pools on the various floor surfaces inside containment. The conductors representing the floors are essentially insulated from the vapor after the pools develop; however, there can still be condensation or evaporation from the surface of the liquid pools. The pool area input value represents the sum of the three floor conductor surface area (3, 4, 5, 6, and 7) from Table 2.6.1-4. Using this method to model the interfacial heat and mass transfer between the pools and the atmosphere was previously approved by the NRC for the Kewaunee containment design basis accident (DBA) and equipment qualification analyses (Reference 5).

The LOCA containment response model input values for the RHR and CCW system's volume, height, diameter, and elevation are not important for modeling the sump temperature response after recirculation. Values of 50 $feet^3$, 5.0 feet, 10.0 feet, and 0 feet were used; the model is not sensitive to these representative input values.

Initial Conditions

The containment initial conditions are listed below:

- Pressure: 15.7 psia
- Relative Humidity: 20%
- Temperature: 120°F

The LOCA containment response model contains volumes representing the RHRS and the CCW system. The RHR system volumes were initially filled with water (120°F) at containment pressure (15.7 psia). The CCW system volumes in the LOCA containment response model were initially filled with water (85°F), but at a higher pressure of 60 psia. The CCW surge tank was modeled as a boundary node at a constant pressure.

Flow Paths

Flow paths connect the boundary conditions to the containment volume. The flow rate is specified by the boundary condition, so most of the flow path input is not important. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path. Since this is a single volume model, the elevation of the break flow paths is arbitrarily set to 50 feet and the elevation of the spray flow paths is arbitrarily set to 90 feet.

Flow boundary conditions model the LOCA break flow to the containment. The boundary conditions are linked to functions that define the mass and energy of the break flow. The boundary conditions are connected to the containment control volume via flow paths.

The containment spray is also modeled as a boundary condition which is connected to the containment control volume via a flow path.

Heat Sinks

The heat sinks in the containment are modeled as GOTHIC thermal conductor. The heat sink data is based on conservatively low surface areas and is summarized in Table 2.6.1-4.

A thin air gap is assumed to exist between the steel and concrete for steel-jacketed heat sinks. A gap conductance of 10 Btu/hr/ft^2/°F is assumed between steel and concrete. A gap of 20 Btu/hr/ft^2/°F is assumed for the minimum gap conductance between layers of insulation and steel or concrete. The gap width is determined by dividing the gap thermal conductivity by the gap conductance.

The volumetric heat capacity and thermal conductivity for the heat sink materials is summarized in Table 2.6.1-5. The specific heat value was calculated based on the volumetric heat capacity.

Heat and Mass Transfer Correlations

GOTHIC has a number of heat transfer coefficient options that can be used for containment analyses.

The direct heat transfer coefficient set is used, along with the Diffusion Layer Model (DLM) mass transfer correlation, for all of the heat sinks inside containment. This heat transfer methodology was reviewed and approved for use in the Kewaunee containment DBA analyses (Reference 5). The DLM correlation does not require the user to specify a revaporization input value, as was done in previous analyses using the Uchida correlation.

The direct heat transfer coefficient set is used for the heat sinks representing floors. The submerged conductors are essentially insulated for the vapor after the pool develops. Insulated surfaces are modeled with a constant (0.0 Btu/hr-ft^2/°F)

Modeling Sump Recirculation

The calculated containment peak pressure and temperature occur long before the transfer to sump recirculation. However, a sump recirculation model consisting of simplified RHRS and CCW system models, was added to the Ginna containment model to calculate the long-term LOCA containment pressure and temperature response.

The recirculation system is actuated after a low RWST level signal. The RHR heat exchanger cools the water from the containment sump. The RHR system injects the cooled water into the RCS to cool the core. The RHR heat exchanger is cooled with CCW water and service water provides the ultimate heat sink, cooling the CCW heat exchangers.

Boundary Conditions

LOCA Mass and Energy Release

The LOCA mass and energy release methodology generates the releases from both sides of the break (or two flow paths: mass and energy exiting from the vessel side of the break; and mass and energy exiting from the steam generator side of the break). The LOCA transient M&E releases are calculated as separate flow paths (for the first 3,600 seconds) and input to the GOTHIC containment model via boundary conditions. The break mass and enthalpy are input to the containment model through forcing functions on flow boundary conditions. The M&E releases from the boundary conditions are analyzed for Ginna out to 3,600 seconds i.e., time at which all energy in the primary heat structures and steam generator secondary system is released/depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F). LR section 2.6.3.1, M&E Release Analysis for Postulated Loss of Coolant Accidents, describes the LOCA long-term M&E release methodology (Reference 2). The boundary conditions are linked to functions that define the mass break flow and the enthalpy of the break flow.

The liquid portion of the break flow is release as drops with an assumed diameter of 100 microns (0.00394 in). This is consistent with the methodology approved for Kewaunee (Reference 5) and is based on data presented in Reference 7.

The long-term post one hour mass and release (boil-off from the core at the decay heating rate) calculations are performed through user defined functions by GOTHIC. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC. After primary system and secondary system energy have been released (depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F), the M&E release to the containment is assumed to be from long-term steaming of decay heat. A flow boundary condition is defined to provide the long-term boil-off M&E release to containment. The mass flow rate and enthalpy of the flow is calculated using GOTHIC control variables.

The long-term boil-off calculation used the American Nuclear Society (ANS) Standard 5.1 decay heat model (+2σ uncertainty) for the determination of long-term boil-off from core. Table 2.6.3-4 lists the decay heat curve used. This assumption is consistent with the long-term M&E methodology documented in Reference 8.

Containment Recirculation Fan Coolers

The containment fan coolers are modeled with a cooler component. There are two trains of containment safeguards available, with two fan coolers per train. Consistent with the application of single-failure criterion presented in LR section 2.6.3.1.2.1.2, Application of Single-Failure Criterion, an inherent assumption is that offsite power is lost with the pipe rupture. This results in the actuation of the EDGs, powering the two trains of safeguards equipment. Operation of the EDG delays the operation of the safeguards equipment that is required to mitigate the transient.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards based on the postulated single failure of an EDG. This assumption results in the loss-of-one train of safeguards equipment. Thus the remaining equipment is conservatively modeled as: two CRFCs and one containment spray pump. The other case assumes maximum safeguards, which assumes both EDGs are available. With the maximum safeguards case the limiting single failure assumption postulated is the failure of one containment spray pump. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

The fan coolers in the containment evaluation model are modeled to actuate on the containment high pressure setpoint with a biased high uncertainty, (6 psig), and begin removing heat from containment after a specified 44-second delay. The heat removal rate per containment fan cooler is given as a function of containment steam saturation temperature is presented in Table

2.6.1-2. The heat removal rate is read into a GOTHIC function and a multiplier, based on the number of fan coolers running is used to calculate the heat removal rate from containment.

Containment Spray System

The containment spray is modeled with a boundary condition. As previously identified in the fan cooler modeling discussion, Ginna has two trains of containment safeguards available, with one spray pump per train. Consistent with the application of single-failure criterion presented in LR section 2.6.3.1.2.1.2, Application of Single-Failure Criterion, an inherent assumption is that offsite power is lost with the pipe rupture. This results in the actuation of the EDG, powering the two trains of safeguards equipment. Operation of the EDG delays the operation of the safeguards equipment that is required to mitigate the transient.

Relative to single failure criterion with respect to a LOCA event, one spray pump is considered inoperable, whether due to a EDG failure (minimum safeguards case) or as the limiting single failure in the maximum safeguards case.

The containment spray is modeled to actuate on the containment high-high pressure setpoint with a biased high uncertainty (33.5 psig) and to begin injecting 104°F water after a specified 28.5 second delay. The containment spray flow is 1,300 gpm per spray pump in the injection phase. The spray flowrate is modeled in GOTHIC as a function of time. The containment spray is credited only during the injection phase of the transient and is terminated during the transition to cold-leg recirculation (i.e., at 2652 seconds).

Accumulator Nitrogen Gas Modeling

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas was released at a rate of 212.3 lbm/seconds, beginning at 44.17 seconds (average accumulator tank water volume empty time) and ending at 64.17 seconds.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The analysis performed to assess the containment response to the limiting LOCA resulting from operation at EPU conditions does not add any new components or introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The analytical results associated with operating at EPU conditions do not add any new or previously unevaluated materials to the plant systems. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

Main Steamline Break

Sixteen steamline break cases were analyzed varying the initial reactor power and the assumed single failure. The M&E release methodology utilizes the RETRAN code, documented in WCAP-14882-P-A. The containment response analysis uses the GOTHIC, version 7.2, computer code. The analysis included the effects of the EPU to 1817MWt, a shutdown margin of 1.3%, and the benefit of the modification to automatically close the main feedwater isolation valves (MFIVs) on a safety injection signal. All the cases consider the largest possible break, a double-ended rupture immediately downstream of the flow restrictor at the outlet of the steam generator. This conservatively bounds the plant response to any smaller break. This break location also limits the break size to the 1.4 ft^2 cross-sectional are of the flow restrictor.

The current limiting steamline break containment pressure case, as described in UFSAR section 6.2.1.2.3, is a full-power double-ended break with a main feedwater regulating valve failure. The new analysis credits feedwater isolation due to the automatically actuated MFIVs with stroke time of 30 seconds.

The analysis also considered more limiting conditions associated with the EPU and a reduction in the minimum required shutdown margin at end of cycle condition for a spectrum of cases. With the MFIV plant modification, the limiting case definition changes to a double-ended break initiated at 70% power with a vital bus failure.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The analysis performed to assess the containment response to the limiting MSLB resulting from operation at EPU conditions does not add any new components or introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The analytical results associated with operating at EPU conditions do not add any new or previously unevaluated materials to the plant systems. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

2.6.1.3 Primary Containment Functional Design Results

2.6.1.3.1 Loss-of-Coolant Accident

The containment pressure, steam temperature, and water (sump) temperature profiles for the DEHL case (peak pressure case) are shown in Figures 2.6.1-1 through 2.6.1-3. Table 2.6.1-6 provides the transient sequence of events for the DEHL transient. The results of the DEPS (long-term EQ transient) are shown in Figures 2.6.1-4 through 2.6.1-6. Table 2.6.1-7 presents sequence of events for the DEPS transient. Table 2.6.1-8 provides the containment pressure and temperature results relative to peak containment conditions and also at 24 hours for EQ support and the acceptance limits for these parameters.

A review of the results presented in Table 2.6.1-8, shows that the analysis margin (analysis margin is the difference between the calculated peak pressure and temperature and the acceptance limits) is maintained, although slightly reduced. The current licensing containment response basis (UFSAR 6.2.1.2.2.6) results for containment peak pressure and temperature for a LOCA event was 52 psig and 275°F, respectively. From the containment response analysis, performed in support of the Ginna EPU program, the containment peak pressure and temperature is 54.21 psig and 282.42°F.

As indicated in Table 2.6.5-1, the peak temperature @ 24 hours of 159.4°F exceeds the acceptance limit of 152°F for a short time. The impact of the 24 hour peak temperature exceeding the acceptance criteria on equipment qualification is evaluated in LR section 2.3.1, Equipment Qualification of Electrical Equipment.

LOCA Containment Response Transient Description Double Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel generator to start resulting in one train of ECCS and containment safeguards equipment being available. The containment heat removal systems that are assumed available are one RHR heat exchanger, two CCW heat exchangers, one containment spray pump (injection phase), and two containment recirculation fan coolers. Further loss of offsite power delays the actuation times of the safeguards equipment due to the time required for diesel startup after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of mass and energy to the containment with a resultant rapid increase in both the containment pressure and temperature. This rapid rise in containment pressure actuates the containment HIGH pressure signal at 0.38 seconds and a containment HIGH-HIGH pressure signal at 4.18 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy, reaching the blowdown peak pressure of 50.59 psig at 13.02 seconds, and then decreasing slightly as the end of blowdown occurs at 13.4 seconds (pressure of 50.55 psig). The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. Since the mass and energy release during this period is low, pressure continues to decrease slightly. At approximately 44.9 seconds the accumulators have emptied, and the pressure increases in response to the loss of steam condensation in the RCS loops and the introduction of the accumulator nitrogen gas to containment out to a second peak which occurred at 70.1 seconds.

During this period the containment spray (32.79 seconds) and containment recirculation fan coolers (44.4 seconds) have also started and are removing heat. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the continued action of the containment recirculation fan coolers and the containment injection spray leads to a slowly decreasing containment pressure out to the end of reflood, which occurs at 222.1 seconds.

At this juncture, by design of WCAP-10325-P-A (Reference 8) mass and energy release evaluation model, energy removal from the steam generator secondary side begins at a very high rate, resulting in a rise in containment pressure from 222.1 seconds out to 1,110 seconds when the ultimate peak pressure of 53.88 psig is reached. Energy continues to be removed from the secondary side of the broken loop and intact loop steam generators until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. This point is reached at 1,117.1 and 1,228 seconds for the broken loop and intact loop steam generators, respectively. Energy removal from the secondary side of the steam generators continues by way of intermediate pressure equilibration stages until the final depressurization, when the secondary reaches the mandatory reference temperature of T_{sat} at 14.7 psia, and 212°F, at

3,600 seconds. The heat removal of the broken loop and intact loop steam generators are calculated separately. The intermediate equilibration stages are met at 1,227.2 seconds for the broken loop steam generator and 1,333 seconds for the intact loop steam generator. After the peak containment pressure is reached and during the steam generator depressurization period, the mass and energy release is reduced since the large energy removal has been accomplished. Containment pressure slowly decreases through the initiation of cold leg recirculation at 2,652 seconds. At this time, the emergency core cooling system (ECCS) is realigned for sump recirculation resulting in an increase in safety injection temperature (due to the delivery from the hot sump and a reduction in steam condensation). Also at 2,652 seconds the containment injection phase spray is terminated from the refueling water storage tank. Without crediting recirculation spray, the containment pressure and temperature will begin to increase out to approximately 3,600 seconds. At this time, the energy removal from the two operating containment recirculation fan coolers exceed the energy release and the pressure and temperature turn around. This trend continues to the end of the transient at 2.592E+6 seconds.

The LOCA containment response analysis has been performed as part of the EPU program for Ginna. As illustrated in the LR section 2.6.1.2.4, Primary Containment Functional Design Results, all cases were well below the containment acceptance limits of 60 psig and 286°F. In addition, the long term DEPS case was well below 50% of the peak containment pressure value within 24 hours. Based on the results, all applicable SRP criteria have been met.

2.6.1.3.2 Main Steamline Break

For the current limiting steamline break case, with credit for FWIV closure within 30 seconds, the peak containment pressure is reduced by 8.1 psig.

A new limiting steamline break was determined to be a double-ended break initiated at 70% power with a vital bus failure. The resultant peak containment pressure is 59.6 psig which is acceptable because it is below the containment design pressure of 60.0 psig. This represents a gain in margin of 0.2 psig compared to the current analysis documented in the UFSAR.

2.6.1.4 Primary Containment Functional Design References

1. Letter from M.G. Korsnick (Ginna) to D.M. Skay (NRC), Subject: License Amendment Request Regarding Main Feedwater Isolation Valves, R.E. Ginna Nuclear Power Plant, dated April 29, 2005.

2. NAI 8907-06, Rev. 15, *GOTHIC Containment Analysis Package Technical Manual, Version 7.2*, September 2004.

3. NAI-8907-09, Rev. 8, *GOTHIC Containment Analysis Package Qualification Report*, Version 7.2, September 2004.

4. *GOTHIC Containment Analysis Package*, Version 5.0, prepared for EPRI, RP 4444-1, December 1995.

5. Docket No. 50-305, Amendment No. 169, Facility Operating License No. DPR-43 (TAC No. MB6408) For Kewaunee Nuclear Power Plant, September 29, 2003.– Issuance of Amendment, 9/29/03, Enclosure 2 – Safety Evaluation.

6. NAI 8907-02, Rev. 16, *GOTHIC Containment Analysis Package User Manual*, Version 7.2, September 2004.

7. AICHE Journal Volume 8, #2, *Sprays formed by Flashing Liquid Jets, Brown and York*, May 1962.

8. WCAP-10325-P-A, May 1983 (Proprietary) and WCAP-10326-A (Nonproprietary), *Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version.*

9. Letter from M.G. Korsnick (Ginna) to Document Control Desk (NRC), Subject: Transmittal of Revised Analysis Associated with the License Amendment Request Regarding Main Feedwater Isolation Valves, R.E. Ginna Nuclear Power Plant, dated July 1, 2005.

Table 2.6.1-1 Containment Response Analysis Parameters	
Parameter	**Value**
Essential Service Water Temperature (°F)	85
RWST Water Temperature (°F)	104
Initial Containment Temperature (°F)	120
Initial Containment Pressure (psia)	15.7
Initial Relative Humidity (%)	20
Net Free Volume (ft^3)	1,000,000
Reactor Containment Fan Coolers	
Total	4
Analysis Maximum	4
Analysis Minimum	2
Containment High Setpoint (psig)	6.0
Delay Time (sec) Without Offsite Power	44.0
Containment Spray Pumps	
Total	2
Analysis Maximum Safeguards	1
Analysis Maximum Safeguards	1
Flowrate (gpm) Injection Phase (per pump)	1,300
Containment High High Setpoint (psig)	33.5
Delay Time (sec) Without Offsite Power (1 spray pump)	28.5
ECCS Recirculation Switchover, sec Minimum Steam Generator Maximum Steam Generator	 2,652 1,446
Containment Spray Termination Time, (sec) Minimum Safeguards Maximum Safeguards	 2,652 1,446
Containment ECCS Sump Recirculation Flow, (gpm) Minimum Safeguards Maximum Safeguards	 1,000 3,000

Table 2.6.1-2 Containment Recirculation Fan Cooler Heat Removal Capability As a Function of Containment Steam Saturation Temperature	
Containment Temperature (°F)	Heat Removal Rate [Btu/sec] Per Reactor Containment Fan Cooler
85	0
120	398
220	8,839
240	10,375
260	11,911
280	13,446
286	13,907

Table 2.6.1-3 LOCA Containment Response Analysis Recirculation System Alignment Parameters	
Residual Heat Removal System	
RHR Heat Exchangers	
Modeled in analysis[a]	1
Recirculation switchover time, sec	
Minimum safeguards	2,652
Flowrate, gpm	
Tubeside (includs 200 gpm pump re-circulation)	1,200
Shellside	1,800
Component Cooling Water Heat Exchangers	
Modeled in analysis	1
Flowrate, gpm	
Shellside[a]	1,800
Tubeside[a] (service water)	5,000
Additional heat loads, BTU/hr	0.0
Notes: a. Minimum heat removal data representing 1 EDG	

Table 2.6.1-4 Containment Structural Heat Sink Input					
Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)	Thickness (ft)
1	Insulated Containment Wall	36,285	SS	0.019	0.00158
			Gap	0.010	
			Insulation	1.250	0.1042
			Gap	0.010	
			Steel	0.375	0.03125
			Gap	0.021	
			Concrete	42.000	3.5
2	Uninsulated Containment Wall	12,370	Overcoat	0.008	
			Primer	0.002	
			Steel	0.375	0.03125
			Gap	0.021	
			Concrete	30.000	2.5
3	Basement Floor	6,576	Overcoat	0.005	
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	24.000	2.0
4	Wet Sump Wall A	8.2	Overcoat	0.004	
			Primer	0.002	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	36.00	3.0
5	Dry Sump Wall A	2,052.8	Overcoat	0.004	
			Primer	0.002	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	36.0	3.0

Table 2.6.1-4 (cont.) Containment Structural Heat Sink Input					
Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)	Thickness (ft)
6	Sump Floors	366	Overcoat	0.005	
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.025	0.0208
			Gap	0.021	
			Concrete	12.000	1.0
7	Walls of Sump B	189	Overcoat	0.005	
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	12.000	1.0
8	Outer Refueling Cavity Wall	6,132	Overcoat	0.005	
			Concrete	35.280	2.94
9	Inner Refueling Cavity Wall	5,609	SS	0.250	0.0208
			Gap	0.021	
			Concrete	24.000	2.0
10	Bottom Refueling Cavity	1,143	SS	0.250	0.0208
			Gap	0.021	
			Concrete	48.0	4.0
11	Loop Compartments	18,846	Overcoat	0.005	
			Concrete	16.938	1.4115
12	Floor of Intermediate Level	9,672	Overcoat	0.005	
			Concrete	3.000	0.25
13	Operating Deck	15,570	Overcoat	0.005	
			Concrete	12.000	1.0
14	Thick Crane Structure	7,225	Overcoat	0.004	
			Primer	0.002	
			Steel	0.750	0.0625
15	Crane Structure	3,374	Overcoat	0.004	

Table 2.6.1-4 (cont.) Containment Structural Heat Sink Input					
Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)	Thickness (ft)
			Primer	0.002	
			Steel	0.415	0.03455
16	I-Beam	7,678	Overcoat	0.004	
			Primer	0.002	
			Steel	0.260	0.0217
17	Thick I-Beam	5,536	Overcoat	0.004	
			Primer	0.002	
			Steel	0.703	0.0586
18	Crane Support	342	Overcoat	0.004	
			Primer	0.002	
			Steel	2.000	0.16667
19	Crane Beams	236	Overcoat	0.004	
			Primer	0.002	
			Steel	1.440	0.12
20	Grating and Misc	14,000	Overcoat	0.004	
			Primer	0.002	
			Steel	0.062	0.005208

Table 2.6.1-5 Material Properties for Containment Structural Heat Sink		
Material	**Conductivity (Btu/hr-ft-°F)**	**Specific Heat (Btu/lbm-°F)**
Concrete	0.81	31.5
Carbon Steel	28.0	54.4
Insulation	0.0208	1.11
Stainless Steel	8.8	54.6
Organic Coating	0.1	20.0
Inorganic Primer	1.0	20.0
Air (gap)	0.0174	0.241

Table 2.6.1-6 Double-Ended Hot Leg Break Sequence of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip, SG Throttle Valve Closure and Loss of Offsite Power are Assumed
1.94	Compensated Pressurizer Pressure for Reactor Trip (1,915 psia) Reached
2.9	Low-Pressurizer Pressure SI Setpoint (1,715 psia) Reached - Feedwater Isolation Signal
6.35	Broken Loop Accumulator Begins Injecting Water
6.39	Intact Loop Accumulator Begins Injecting Water
14.9	Feedwater Isolation Valves Closed
15.02	Peak Temperature Occurs (280.1°F)
15.52	Peak Pressure Occurs (54.21 psig)
16.0	End of Blowdown Phase
30.0	Transient Modeling Terminated

Table 2.6.1-7 Double-Ended Pump Suction Break Sequence of Events (Minimum Safeguards)	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip, SG Throttle Valve Closure and Loss of Offsite Power are Assumed
0.38	Containment HIGH Pressure Setpoint (20.7 psia; 6.0 psig;) Reached
1.83	Compensated Pressurizer Pressure Reactor Trip (1,915 psia) Reached
3.1	Low Pressurizer Pressure SI Setpoint (1,695 psia) Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
4.18	Containment HIGH-HIGH Pressure Setpoint (33.5 psig; Analysis Value) Reached
6.46	Broken Loop Accumulator Begins Injecting Water
6.55	Intact Loop Accumulator Begins Injecting Water
13.4	End of Blowdown Phase
13.4	Accumulator Mass Adjustment for Refill Period
15.1	Feedwater Isolation Valves Closed
32.78	Containment Spray Pump (RWST) Begins
35.1	Pumped Safety Injection Begins (Includes 32 Second Diesel Delay)
43.4	Broken Loop Accumulator Water Injection Ends
44.43	Containment Fan Coolers Actuate
44.9	Intact Loop Accumulator Water Injection Ends
222.1	End of Reflood for Minimum Safeguards Case
1,110.0	Containment Peak Pressure and Temperature Occurs (53.88 psig; and 282.4°F)
1,117.1	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration When the Secondary Temperature is the Saturation (T_{sat}) At Containment Design Pressure of 74.7 psia
1,227.2	M&E Release Assumption: Broken Loop SG Equilibration at Containment Pressure of 64.7 psia
1,228.0	M&E Release Assumption: Intact Loop SG Equilibration When the Secondary Temperature is the Saturation (T_{sat}) at Containment Design Pressure of 74.7 psia
1,333.0	M&E Release Assumption: Intact Loop SG Equilibration at Containment of 54.7 psia
2,652.0	Switchover to Cold Leg Recirculation Begins
2,652.0	Containment Spray Terminated
2.592E+6	Transient Modeling Terminated

Table 2.6.1-8 LOCA Containment Response Results				
Case	Peak Press. @ Time	Peak Temp. @ Time	Peak Press. (psig) @ 24 hours	Peak Temp. (°F) @ 24 hours
DEHL	54.21 psig @ 15.52 sec	280.1°F @ 15.02 sec	---	---
DEPS-Minimum Safeguards	53.88 psig @ 1,110. sec	282.4°F @ 1,110. sec	7.77	159.4

Containment Pressure – Acceptance Limits		
	Peak Pressure	Pressure @ 24 hours
Pressure	60 psig	UFSAR Figure 6.1-2

Containment Temperature – Acceptance Limits		
	Peak Temperature	Temperature @ 24 hours
Temperature	286°F	152 (UFSAR Figure 6.1.-1)

Jun/22/2005 08:37:46
GOTHIC Version 7.2(QA) - Sept 2004
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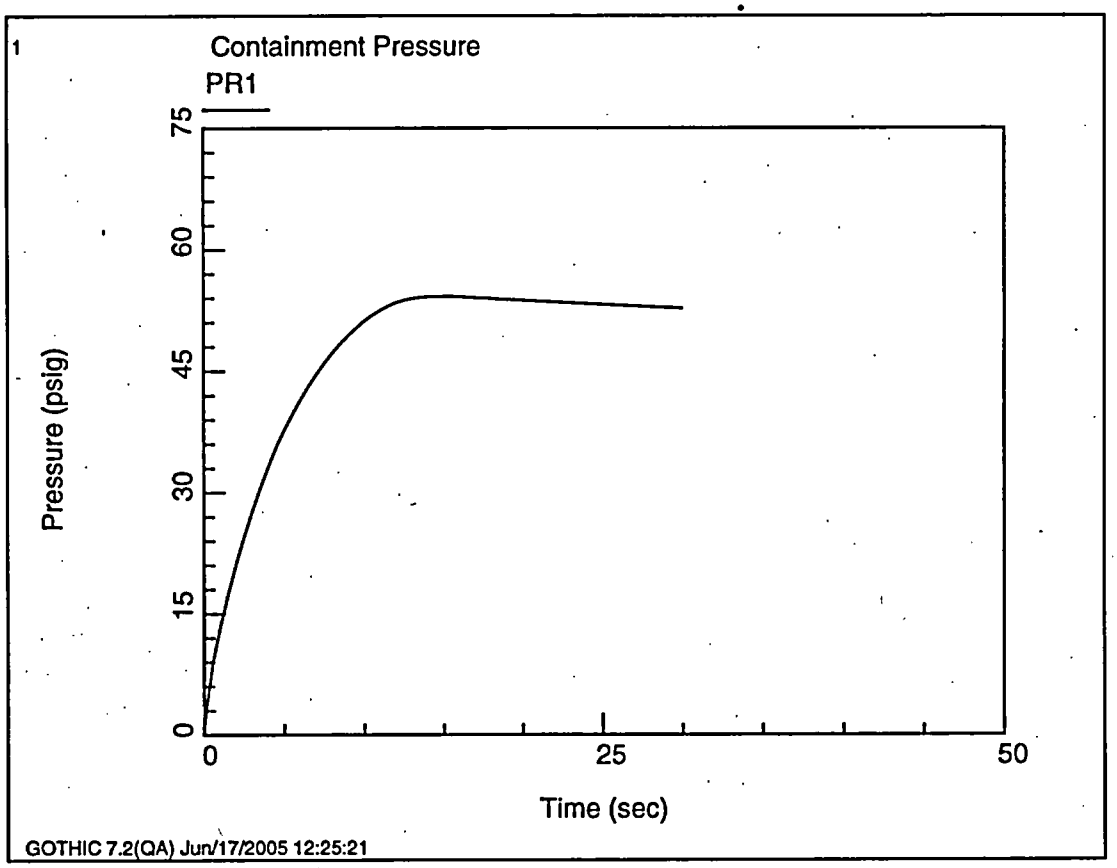

Figure 2.6.1-1

Containment Pressure - Double-Ended Hot Leg Break

Jun/22/2005 08:38:51
GOTHIC Version 7.2(QA) - Sept 2004
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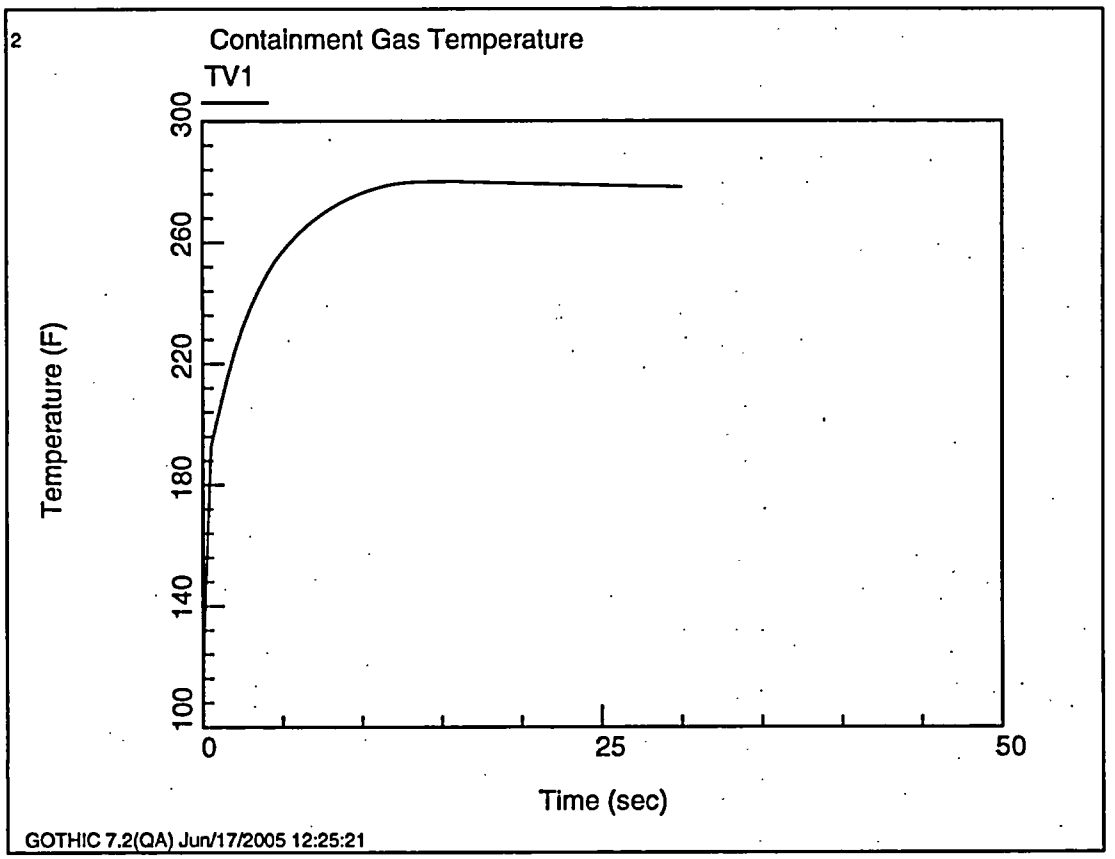

Figure 2.6.1-2

Containment Temperature – Double-Ended Hot Leg Break

Jun/22/2005 08:40:10
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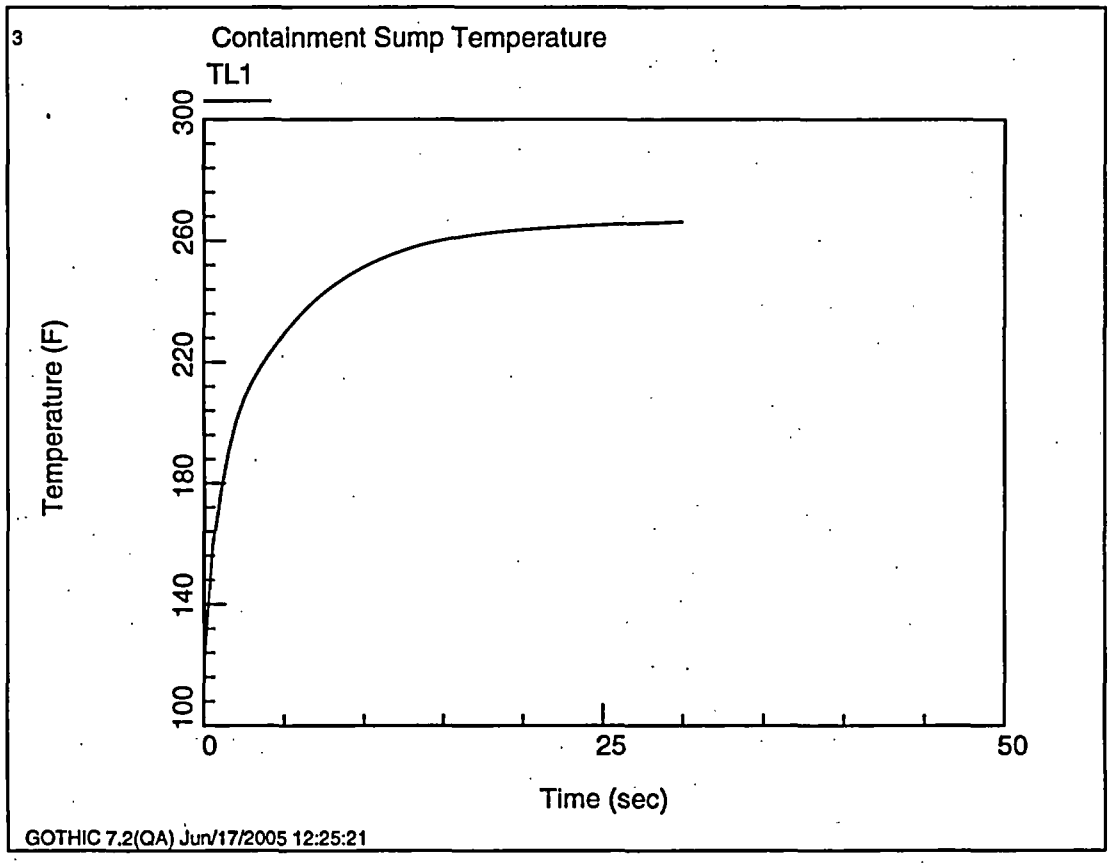

Figure 2.6.1-3

Containment Sump Temperature – Double-Ended Hot Leg Break

Jun/20/2005 07:08:03
GOTHIC Version 7.2(QA) - Sept 2004
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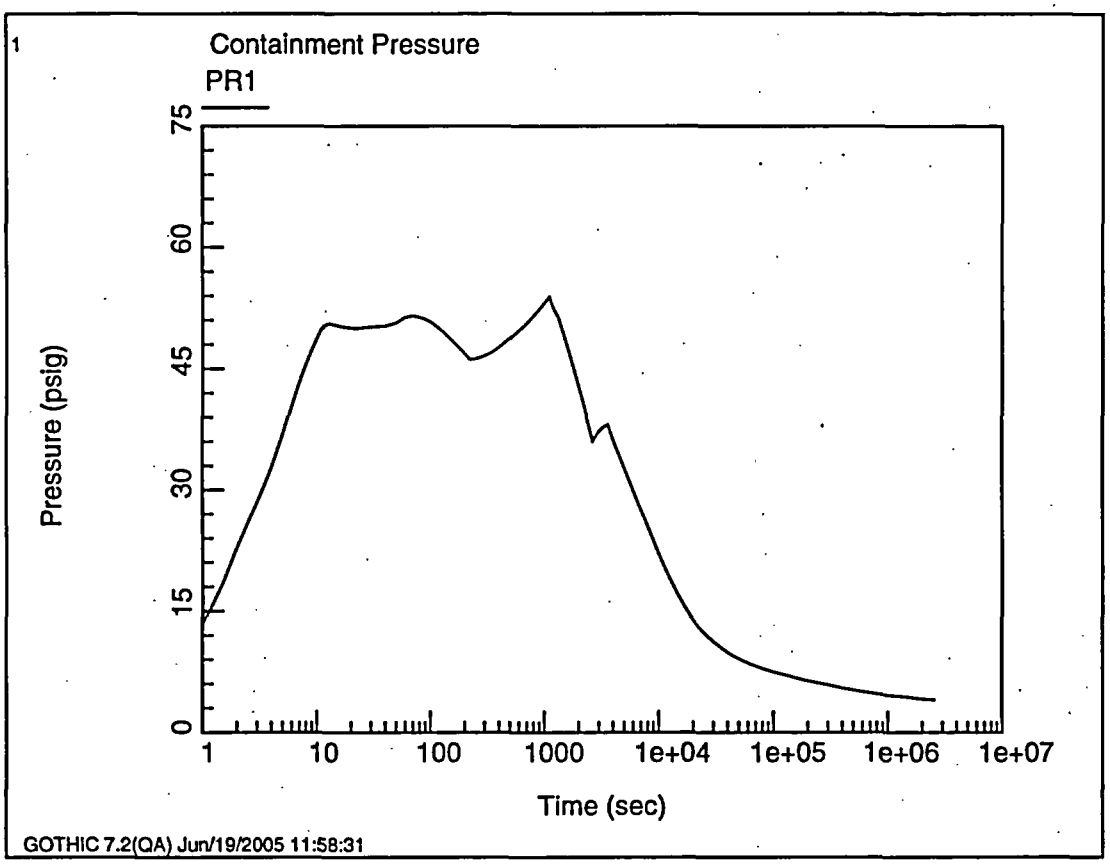

Figure 2.6-1-4

Containment Pressure – Double-Ended Pump Suction Break

Jun/20/2005 07:10:38
GOTHIC Version 7.2(QA) - Sept 2004
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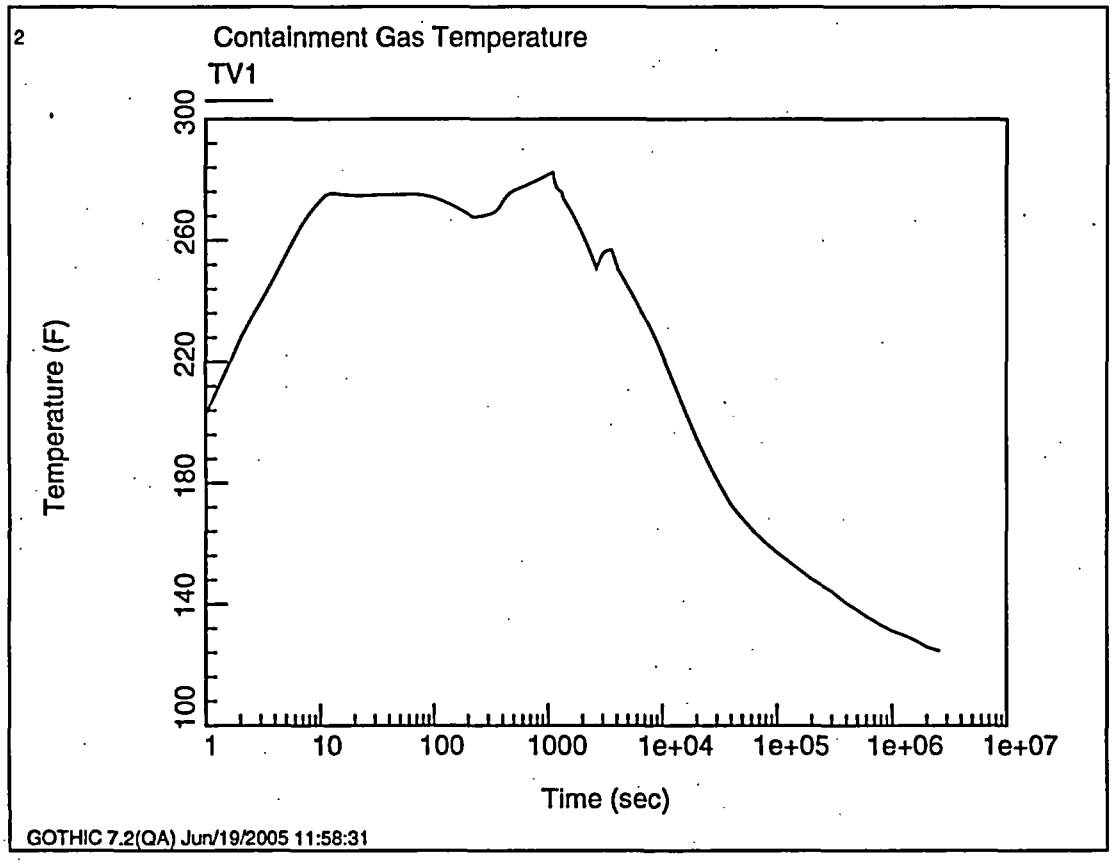

Figure 2.6.1-5

Containment Temperature – Double-Ended Pump Suction Break

Jun/20/2005 07:13:22
GOTHIC Version 7.2(QA) - Sept 2004
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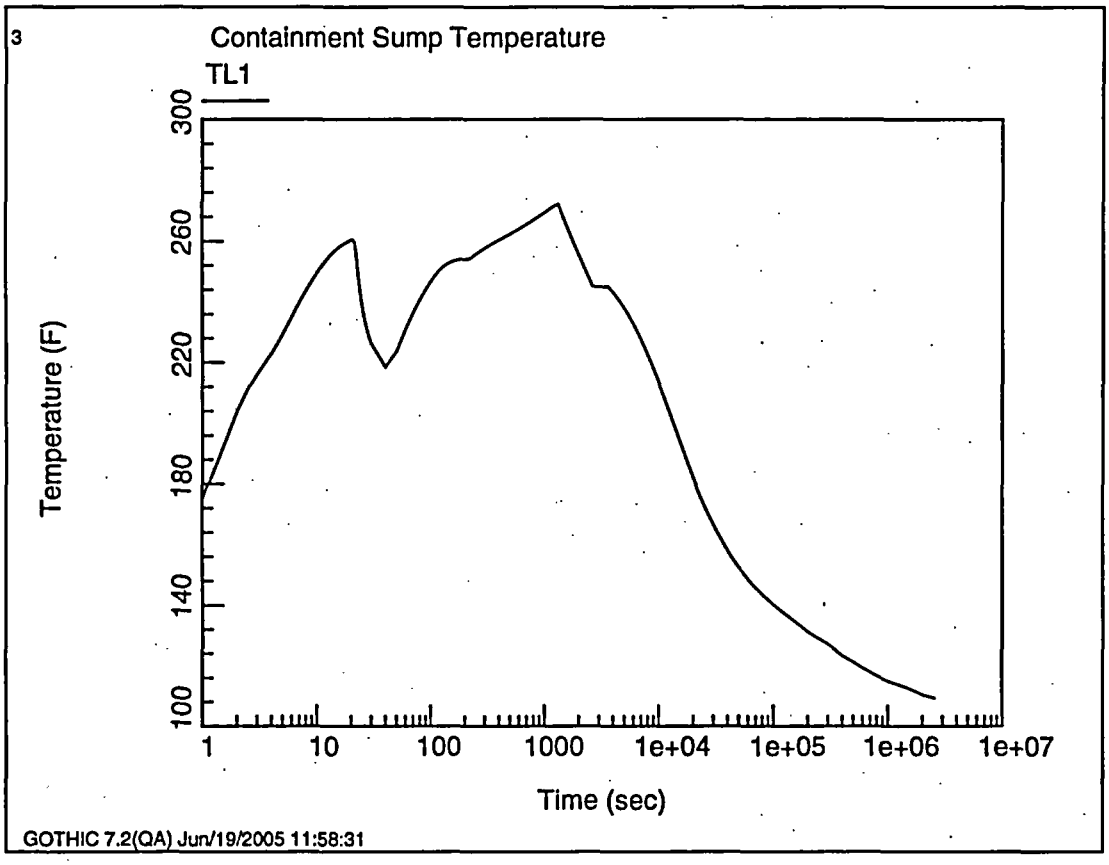


Figure 2.6.1-6

Containment Sump Temperature – Double-Ended Pump Suction Break

2.6.1.5 Conclusion

The Ginna staff has reviewed the assessment of the containment pressure and temperature transient and concludes that it has adequately accounted for the increase of M&E that would result from the proposed EPU. The Ginna staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The Ginna staff also concludes that the containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-13, GDC-16, GDC-38, GDC-50, and GDC-64 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to containment functional design.

2.6.2 Subcompartment Analyses

2.6.2.1 Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The Ginna Nuclear Power Plant, LLC (Ginna) staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The Ginna staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions and the resulting increase in pressurization.

The NRC's acceptance criteria for subcompartment analyses are based on

- GDC-4, insofar as it requires that structures, systems and components (SSCs) important-to-safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects, and

- GDC-50, insofar as it requires that the containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments.

Specific review criteria are contained in NRC SRP section 6.2.1.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy containment subcompartment structural design relative to conformance to:

- GDC-4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 – Environmental and Missile Design Bases. As described in this UFSAR section, a review of postulated pipe breaks inside containment was conducted as part of the SEP (SEP Topic III-5.A) including dynamic effects such as pipe whip and jet impingement. The

evaluation of postulated high energy piping failures inside containment is discussed in UFSAR sections 3.6.1.3.2 and 6.2.1.3. As discussed in UFSAR section 5.4.11.1, asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping is not a design basis for the Ginna Station. In addition, Ginna Station applies leak-before-break (LBB) methodology to eliminate from consideration the dynamic effects which would result from certain high energy pipe breaks as discussed in LR section 2.1.6.

- GDC-50 is described in Ginna UFSAR 3.1.2.5.1, General Design Criterion 50 – Containment Design Basis. As described in this UFSAR section, the containment structure is designed with margin to accommodate the temperature and pressure conditions associated with the LOCA and main steam line break without loss of function. An evaluation of containment internal structures relative to post-accident pressure differentials and thermal gradients is discussed in UFSAR section 6.2.1.3.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's SSCs have been evaluated for plant license renewal. Plant system and component materials of construction, operating history, and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, NUREG-1786, dated May 2004

Containment internal structures are included within the scope of license renewal as discussed in SER section 2.4.1. The passive, long-lived internal structures are subject to existing aging management programs as described in SER section 3.5.

2.6.2.2 Technical Evaluation

2.6.2.2.1 Introduction

The containment subcompartments are evaluated for their structural response to potential increases in pressure differentials and thermal gradients resulting from postulated accidents which are conservatively assumed to initiate at EPU operating conditions.

2.6.2.2.2 Description of Analyses and Evaluations

The uncontrolled release of pressurized high-temperature reactor coolant, termed a LOCA, would result in the release of steam and water into the containment. This, in turn, would result in increases in the local subcompartment pressures and an increase in the global containment pressure and temperature. Both long-term and short term effects on the containment resulting from a

postulated LOCA were considered using the condition for Ginna at the EPU core power (see LR Table 1-1). The EPU analyses were performed using the Westinghouse LOCA Mass and Energy (M&E) Release Model for Containment Design described in WCAP-10325-P-A (reference 1).

The short-term M&E LOCA-related M&E releases were used as input to the subcompartment analyses. The analyses were performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high energy line pipe rupture within that subcompartment. Short-term M&E release calculations are performed to support analysis of reactor coolant loop compartments (UFSAR section 6.2.1.3.2), the concrete around and under the reactor vessel (UFSAR section 6.2.1.3.4) and the concrete structures around the steam generators (UFSAR section 6.2.1.3.4). Since Ginna is approved for LBB for the RCS loops and RHR branch lines, and has applied for LBB for all remaining 10 inch branch lines, the LBB methodology was used to qualitatively demonstrate that any changes associated with the proposed EPU are offset by the LBB benefit (i.e., the application of smaller reactor coolant system (RCS) nozzle breaks for subcompartment analyses). This demonstrates that the current licensing bases for these subcompartments remains bounding. The critical mass flux correlation utilized in the SATAN computer program was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term release. The evaluation showed that the design basis release would remain bounding due to LBB.

Short-Term LOCA M&E Releases

As noted in the Ginna UFSAR, short-term LOCA M&E release calculations were performed to support analysis of the reactor coolant loop compartment, the concrete around and under the reactor vessel, and the concrete structures around the steam generators. The breaks assumed in the current licensing basis for these structures are primary loop breaks (a longitudinal split of area equivalent to the cross-section of a reactor coolant pipe, i.e., 4.5 ft^2). Input parameters and assumptions used for the analyses are described in LR section 2.6.3.1.2.1.

Ginna has been approved by the NRC for application of LBB methodology as part of the NRC staff's resolution of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant Systems", and subsequent submittals per GDC-4 (see LR section 2.1.6). With the elimination of the large RCS breaks for calculating dynamic effects, the only break locations that need to be considered are the largest remaining branch lines off of the primary loop piping. These branch lines are the pressurizer spray line and the upper plenum injection lines into the vessel, the pressurizer surge line and the accumulator line.

Even the releases from these smaller 10-inch breaks are considerably lower than would result from the large RCS breaks. Using a double-ended break in the largest branch line rather than the current single-ended split in the RCS reduces the break size by a factor of 4. A reduction of this magnitude in pipe break size has a significant beneficial impact on the subcompartment loadings. This results

in a peak compartment pressure reduction by a factor of 2.76 and a differential pressure reduction across an adjacent subcompartment wall by a factor of 3.86. Using the EPU pressure and temperature conditions (i.e., increased pressure and decreased temperature), the lower temperature increases the short-term LOCA M&E release. Using the critical mass flux equation, it was determined that the release could increase by 9.7%. Therefore, from a very conservative perspective, the current design basis releases could increase by a factor of 1.10 due to RCS initial pressure/temperature condition effects.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Program

In addition to the evaluations described in the UFSAR, containment subcompartment analyses considerations were evaluated for plant License Renewal. No systems or components are being added or modified as the result of re-evaluation of subcompartment analyses for EPU conditions. The subcompartment analyses described in this LR section involve only analytical techniques and results that do not introduce new functions for existing components that would change the license renewal boundaries. Therefore, no new aging effects requiring management are identified with respect to containment subcompartment analyses.

2.6.2.3 Results

The Ginna containment building employs a generally open design which diminishes the effects of differential pressure/temperature effects on subcompartment walls (see drawings 33013-2101, 2113, 2131 and 2132).

The effect of eliminating the large RCS breaks through application of LBB methodology, and substituting a 10-inch break, reduces the effective break area by a factor of 4. The penalty associated with assumed M&E release from the proposed EPU operating conditions is a release increase of a factor of 1.10. Therefore, the benefit of applying LBB methodology to postulated break locations more than offsets any penalty to the current licensing basis short-term M&E release and subcompartment analyses associated with the proposed EPU. Furthermore, NRC approval of the LBB submittals for the connecting 10-inch branch lines (see LR section 2.6.1) would further increase available margin.

The combination of an open containment design, application of LBB methodology to RCS and connected piping 10-inches and greater, and minimal changes in primary or secondary pressures and temperatures associated with EPU operating conditions results in EPU having no deleterious effects on subcompartment design parameters.

2.6.2.4 Reference

1. WCAP-10325-P-A, May 1983 (Proprietary) and WCAP-10326-A (Nonproprietary), *Westinghouse LOCA Mass and Energy Release Model for Containment Design*, March 1979.

2.6.2.5 Conclusion

The Ginna staff has reviewed the subcompartment assessment and the change in predicted pressurization resulting from the increased M&E release. The Ginna staff concludes that safety-related containment SSCs will continue to be protected from the dynamic effects resulting from pipe breaks and that subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this, the Ginna staff concludes that the Ginna Station will continue meet its current licensing basis as recently modified by approved LBB methodology with respect to the requirements of GDC-4 and GDC-50 for the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 M&E Release Analysis for Postulated Loss-of-Coolant Accidents

2.6.3.1.1 Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. Ginna Nuclear Power Plant, LLC's (Ginna) review covered the energy sources that are available for release to the containment and the Mass & Energy (M&E) release rate calculations for the initial blowdown phase of the accident.

The NRC's acceptance criteria for M&E release analyses for postulated loss-of-coolant accidents (LOCAs) are based on:

- GDC-50, insofar as it requires that sufficient conservatism is provided in the M&E release analysis to ensure that containment design margin is maintained
- 10CFR50, Appendix K, insofar as it identifies energy sources during a LOCA

Specific review criteria are contained in the SRP, Section 6.2.1.3.

Ginna Current Licensing Basis

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), section 3.1, the GDC used during the licensing of Ginna Station predates those provided in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of assumptions regarding energy sources available for release to the containment and the M&E release rate calculations relative to conformance to:

- GDC-50 is described in Ginna UFSAR section 3.1.2.5.1, General Design Criterion 50 – Containment Design Basis. As described in this UFSAR section, the containment structure was designed with margin to accommodate the temperature and pressure conditions associated with the loss-of-coolant

accident and main steam line break, without loss of function. The evolution of the containment analysis licensing basis is discussed in LR section 2.6.1.1. For purposes of evaluating the integrity of the containment as a whole and the integrity of structures internal to the containment (sub-compartments), the effects of M&E releases are examined for long-term releases and short-term releases, respectively. The containment functional design requirement is discussed in UFSAR SECTION 6.2.1.1.1. The containment integrity evaluation (long-term releases) is described in UFSAR section 6.2.1.2. LR section 2.6.1 discusses containment LOCA response analysis. Short-term M&E release calculations are performed to support the reactor coolant loop compartment (UFSAR section 6.2.1.3.2), the concrete around and under the reactor vessel (UFSAR section 6.2.1.3.4), and concrete structures around the steam generators (UFSAR section 6.2.1.3.4). The current licensing basis for these structures is a longitudinal split in a primary loop pipe of an equivalent cross-sectional area of 4.5 ft^2. LR section 2.6.2 discusses the sub-compartment analysis.

- 10CFR50, Appendix K, is described in Ginna UFSAR section 15.6.4.2 for the loss-of-coolant accident (LOCA) analyses. Five cases of large break LOCA are analyzed which conform to the modeling requirements of 10CFR50, Appendix K. These analyses identify all appropriate sources of energy available for the large break LOCA cases considered and demonstrate conformance to the acceptance criteria of 10CFR50.46 to ensure that fuel design limits are not exceeded.

2.6.3.1.2 Long-Term LOCA M&E Releases

2.6.3.1.2.1 Technical Evaluation

The evaluation/generation of the design basis long-term LOCA M&E release data was completed to support the EPU operation.

2.6.3.1.2.1.1 Introduction

The long-term LOCA M&E releases are described in the Ginna UFSAR, Section 6.2.1.2.2. The M&E release rates described in this section form the basis of further computations to evaluate the containment response following the postulated LOCA (UFSAR section 6.2.1.2.2.6) and to ensure that containment design margin is maintained.

The uncontrolled release of pressurized high-temperature reactor coolant, termed a LOCA, will result in the release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures and an increase in the global containment pressure and temperature. Both long-term and short-term effects on the containment resulting

from a postulated LOCA were considered using the operating conditions for Ginna at the EPU uprated core power.

The long-term LOCA M&E releases were analyzed for Ginna out to 3600 seconds, i.e., the time at which all energy in the primary heat structures and steam generator secondary system is released / depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F). The long-term post-one hour releases (boil-off from the core at the decay heating rate) were calculated by the GOTHIC code. The pre-3600-second and past 3600-second M&E releases were inputs to the containment integrity analysis (discussed in LR section 2.6.1, Primary Containment Functional Design). To demonstrate the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA), the long-term LOCA M&E releases were analyzed to 30 days and used as input to the containment integrity analysis. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure, and maintaining the Environmental Qualification (EQ) conditions within acceptable limits.

The EPU analyses were performed using the Westinghouse LOCA M&E Release Model for Containment Design, March 1979 Version, described in WCAP-10325-P-A (reference 1). The NRC review and approval letter is included with reference 1. LR section 2.6.3.1.2, Long-Term LOCA M&E Releases, discusses the long-term LOCA M&E releases generated for the EPU program.

The short-term LOCA-related M&E releases were used as input to the subcompartment analyses (see LR section 2.6.2, Subcompartment Analyses). Those analyses were performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. Short-term M&E release calculations are performed to support analysis of reactor coolant loop (RCL) compartments (UFSAR, Section 6.2.1.3.2), the concrete around and under the reactor vessel (UFSAR, Section 6.2.1.3.4), and the concrete structures around the steam generator (UFSAR, Section 6.2.1.3.4). Since Ginna is approved for leak-before-break (LBB), the LBB methodology was used to qualitatively demonstrate that any changes associated with the EPU are offset by the LBB benefit (i.e., of using the smaller reactor coolant system (RCS) nozzle breaks). This demonstrates that the current licensing bases for these subcompartments remain bounding. The critical mass flux correlation utilized in the SATAN computer program (reference 2) was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term release. The evaluation showed that the design basis releases would remain bounding due to LBB. LR section 2.6.3.1.3, Short-Term LOCA M&E Releases, discusses the short-term LOCA M&E releases generated for the EPU program. LR section 2.6.2, Subcompartment Analyses, discusses the short-term evaluation conducted for this program.

2.6.3.1.2.1.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters and Assumptions

The M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures were chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance (+4°F) was then added. The RCS pressure in this analysis is based on a nominal value of 2250 psia, plus an uncertainty allowance (+60 psi). All input parameters are chosen consistent with accepted analysis methodology.

Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed in the following paragraphs. Tables 2.6.3.1-1 through 2.6.3.1-3 present key data assumed in the analysis.

The core-rated power of 1811 MWt, adjusted for calorimetric error (i.e., 102% of 1775 MWt), was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and dead band was reflected in the initial RCS temperature. As previously discussed, the initial RCS pressure in this analysis was based on a nominal value of 2250 psia, plus an allowance that accounted for the measurement uncertainty on pressurizer pressure. The selection of 2310 psia as the limiting pressure is considered to affect blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point where it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally, the RCS has a higher fluid density at the higher pressure (assuming a constant temperature), and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting condition for the long-term M&E release calculations.

The selection of the fuel design features for the long-term M&E release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (that is, to maximize the core-stored energy). The core-stored energy that was selected for the 14x14 422V+ fuel product bounds the core-stored energy for 14x14 optimized fuel assembly (OFA) fuel product and also the transition core. The core-stored energy is based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core-stored energy calculations account for appropriate uncertainties associated with the models in the PAD code (such as calibration of the thermal

model, pellet densification model, or clad creep model). In addition, the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (such as pellet as-built density). The total uncertainty for fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel type, power level, and burnup. Thus, the analysis very conservatively accounts for the stored energy in the core.

The RCS volume is increased by 3%, which is composed of a 1.6% allowance for thermal expansion and a 1.4% allowance for uncertainty.

A uniform steam generator tube plugging (SGTP) level of 0% was modeled. This assumption maximized the reactor coolant volume and fluid release by including the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0% tube plugging assumption maximized heat transfer area and, therefore, the transfer of secondary heat across the steam generator tube. Additionally, this assumption reduced the RCL resistance, which reduced the ΔP upstream of the break for the pump suction breaks and increased break flow. Thus, the analysis very conservatively modeled the effects related to SGTP.

The secondary-to-primary heat transfer is maximized by assuming conservative heat transfer coefficients. This conservative energy transfer is ensured by maximizing the initial internal energy of the inventory in the steam generator secondary side. This internal energy is based on full-power operation plus uncertainties.

Following a large break LOCA inside containment, the safety injection system (SIS) operates to reflood the RCS. The first phase of the SIS operation is the passive accumulator injection. Two accumulators are assumed available to inject. When the RCS depressurizes below 714.7 psia the accumulators begin to inject. The accumulator injection temperature was conservatively modeled high at 120°F. Relative to the active pumped emergency cool cooling system (ECCS) operation, the M&E release calculation considered configurations, component failures, and offsite power assumptions to conservatively bound respective alignments. The cases include a minimum safeguards case (two high-head SI (HHSI) pumps and one low-head SI (LHSI pump, see Table 2.6.3.1-2), and a maximum safeguards case, (three HHSI pumps and two LHSI pumps, see Table 2.6.3.1-3). In addition, the containment backpressure is assumed to be equal to the containment design pressure. This assumption was shown in reference 1 to be conservative for the generation of M&E energy releases.

In summary, the following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment:

- Maximum expected operating temperature of the RCS (100% full-power operation)

- Allowance for RCS temperature uncertainty (+4.0°F)
- Margin in RCS volume of 3% (which is composed of 1.6-percent allowance for thermal expansion, and 1.4% allowance for uncertainty)
- Core power of 1811 MWt
- Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer)
- Allowance in core-stored energy for effect of fuel densification
- An allowance for RCS initial pressure uncertainty (+60 psi)
- A total uncertainty for fuel temperature calculation based on a statistical combination of effects and dependent upon fuel type, power level, and burnup
- A maximum containment backpressure equal to design pressure (60 psig)
- SGTP level (0% uniform)

  Maximizes reactor coolant volume and fluid release

  Maximizes heat transfer area across the steam generator tubes

  Reduces RCL resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Thus, based on the previously discussed conditions and assumptions, an analysis of the Ginna Station was performed for the release M&E from the RCS in the event of LOCA at 1811 MWt core power.

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the M&E release rates for each break analyzed. An inherent assumption in the generation of the M&E release is that offsite power is lost with the pipe rupture. This results in the actuation of the emergency diesel generators (EDGs), required to power the safety injection system. Operating the EDG delays the operation of the SIS that is required to mitigate the transient.

Two cases have been analyzed to assess the effects of a single failure. The first case assumed minimum safeguards SI flow based on the postulated single failure of an EDG. This assumption results in the loss of one train of safeguards equipment. Thus the remaining train was conservatively modeled as: two HHSI pumps and one LHSI pump. The other case assumed

maximum safeguards SI flow based on no postulated failures that could impact the amount of ECCS flow. The maximum safeguards case was modeled as: three HHSI pumps and two LHSI pumps. The single failure assumption postulated is the failure of one containment spray pump. However, this has no impact on the amount of ECCS flow and, therefore, no impact on the M&E release portion of the analysis. The analysis of the cases described provided confidence that the effect of credible single failures is bounded

Decay Heat Model

American Nuclear Society (ANS) Standard 5.1 was used in the LOCA M&E release model for Ginna for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee (NUPPSCO) in October 1978 and subsequently approved. The official standard was issued in August 1979. Table 2.6.3.1-4 lists the decay heat curve used in the Ginna EPU Program M&E release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E release analysis include the following:

- The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- The fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from American ANS Standard 5.1 (reference 5).
- The fuel has been assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission has been assumed to be 200 MWV/fission.
- Two sigma uncertainty (two times the standard deviation) have been applied to the fission product decay.

Based upon NRC staff review, (Safety Evaluation Report of the March 1979 evaluation model (reference 1), use of the ANS Standard-5.1, November 1979 decay heat model, was approved for the calculation of M&E releases to the containment following a LOCA.

Acceptance Criteria

Although Ginna is not a Standard Review Plan (SRP) plant, for completeness, the SRP long-term cooling criterion is also examined. An LBLOCA is classified as an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the SRP section 6.2.1.3, the relevant requirements are as following:

- 10CFR50, Appendix A
- 10CFR50, Appendix K, paragraph I.A

To meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

2.6.3.1.2.1.3 Description of Analyses and Evaluations

Description of Analyses

The evaluation model (EM) used for the long-term LOCA M&E release calculations is the 1979 model described in WCAP-10325-P-A (reference 1). This EM has been reviewed and approved by the NRC. The approval letter is included with reference 1.

This report section presents the long-term LOCA M&E releases generated in support of the Ginna EPU program. These M&E releases were used in the containment integrity analysis and qualification temperature evaluation (LR section 2.6.1, Primary Containment).

The M&E release rates described in this section form the basis of further computations to evaluate the containment following the postulated accident. Discussed in this section are the long-term LOCA M&E releases for the hypothetical double-ended pump suction (DEPS) rupture with minimum safeguards, and maximum safeguards and DEHL rupture cases. The DEPS minimum safeguard break was analyzed with two service water pumps in operation. Only the results for the limiting peak pressure case and the limiting long-term EQ case are presented herein and in the UFSAR. The M&E release for these two cases are shown in Tables 2.6.3.1-5 through 2.6.3.1-13. These two cases are used for the long-term containment response analyses in LR section 2.6.1, Primary Containment Functional Design and LR section 2.6.5, Containment Heat Removal.

LOCA M&E Release Phases

The containment system receives M&E releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA M&E analysis, is typically divided into four phases.

- Blowdown – the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state

- Refill – the period of time when the lower plenum is being filled by the accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment M&E releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient water to completely fill the lower plenum. This allows an uninterrupted release of M&E to containment. Thus, the refill period is conservatively neglected in the M&E release calculation.

- Reflood – the period of time that begins when water from the lower plenum enters the core and ends when the core is completely quenched

- Post-Reflood (FROTH) – the period of time following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken-loop steam generator cools, the break flow becomes two-phase.

Computer Codes

The WCAP-10325-P-A (reference 1) M&E release evaluation model consists of M&E release versions of the following codes: SATANV1, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA M&E releases for Ginna.

SATAN VI calculates the blowdown phase, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, M&E flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and cools the core. The most important feature of WREFLOOD is the steam/water mixing model.

The FROTH code models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to the containment design pressure to the end of the transient. It also compiles a summary of data for the entire transient, including formal instantaneous M&E release tables and M&E balance tables with data at critical times.

Break Size and Location

Generic studies have been performed and documented in reference 1 with respect to the effect of postulated break size on the LOCA M&E releases. The double-ended guillotine (DEG) break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for a pipe rupture for M&E release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The break locations analyzed for this program are the DEPS rupture (10.48 ft^2) and the DEHL rupture (9.174 ft^2). Break M&E releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown phase. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown M&E release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid that exits the core vents directly to containment, bypassing the steam generators. As a result, the reflood M&E releases are reduced significantly as compared to either the pump suction or cold-leg break location releases where the core exit mixture must pass through the steam generators before venting through the break. For the hot-leg break, generic studies have confirmed that there is no reflood peak (that is, from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the M&E releases for the hot-leg break blowdown phase were calculated and presented in this report. In addition, since none of the powered safety systems were assumed to be operational during the initial blowdown phase, the service water system would not impact the DEHL break.

The cold-leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold-leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core

heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that of the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold-leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot-leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment. Thus, only the DEHL and DEPS cases were used to analyze long-term LOCA containment integrity.

M&E Release Data

Blowdown M&E Release Data

The SATAN VI code was used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermo-dynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated breakflow is incorporated into the analysis. The methodology for the use of this model is described in WCAP-10325-P-A (reference 1).

Table 2.6.3.1-5 presents the calculated M&E release for the blowdown phase of the DEHL break. For the DEHL break M&E release tables, break path 1 refers to the M&E exiting from the reactor vessel side of the break; break path 2 refers to the M&E release exiting from the steam generator side of the break. Table 2.6.3.1-8 presents the calculated M&E releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the M&E release tables refers to the M&E exiting from the steam generator side of the break. Break path 2 refers to the M&E exiting from the pump side of the break.

Reflood M&E Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models: one for the contents of the reactor vessel and one for the RCLs. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped SI and accumulators, reactor coolant pump (RCP) performance, and steam generator release are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code has the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer

water levels, fluid thermo-dynamic conditions (pressure, enthalpy, and density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, that is, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the WCAP-10325-P-A (reference 1) M&E release evaluation model in recent analyses, for example, D. C. Cook Docket Unit 1 (reference 3). Even though the WCAP-10325-P-A (reference 1) model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented reference 3). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (that is, thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. These data were generated in 1/3-scale tests (reference 4), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a pressurized water reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in WCAP-10325-P-A (reference 1). For all these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture. For this break, there are two flow paths available in the RCS by which M&E can be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the RCP. Steam that is not condensed by ECCS injection in the intact RCS loop passes around the downcomer and through the broken-loop cold leg and RCP in venting to

containment. This steam also encounters ECCS injection water as it passes though the broken-loop cold leg, complete mixing occurs, and a portion of it is condensed. It is this portion of steam that is condensed that is credited in this analysis. This assumption is justified based upon the postulated break location and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in WCAP-10325-P-A (reference 1) and operating license Amendment No. 126 for D. C. Cook Unit 1 (reference 3).

Table 2.6.3.1-9 presents the calculated M&E releases for the reflood phase of the pump suction double-ended rupture with minimum safeguards.

The transient response of the principal parameters during reflood are given in Table 2.6.3.1-10 for the DEPS cases.

Post-Reflood M&E Release Data

The FROTH code (reference 2) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The M&E releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken-loop and intact-loop steam generators. During this phase of the transient, the RCS has equilibrated with containment pressure. However, the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure, after this point the EPITOME code completes the calculation to the time of switchover to cold-leg recirculation.

The methodology for the use of this model is described in WCAP-10325-P-A (reference 1). The M&E release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the M&E release available to containment is generated directly from core boil off/decay heat.

Table 2.6.3.1-11 presents the two-phase post-reflood M&E release data for the pump suction double-ended cases with minimum ECCS assumptions.

Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the calculation for steam generator cooldown by removing steam generator secondary energy at different rates (that is, first and second stage rates). The first stage rate is applied until the steam generator reaches T_{sat} at the user-specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then, the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken-loop and intact-loop steam generators is calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature, and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches T_{sat} at the containment design pressure. The constant heat-removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user-specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first-stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second-stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, that is, 14.7 psia and 212°F (the M&E balance Tables 2.6.3.1-6 and 2.6.3.1-7 have this point labeled as "Available Energy").

Sources of M&E

The sources of mass considered in the LOCA M&E release analysis are given in Tables 2.6.3.1-6 and 2.6.3.1-12. These sources are:

- The RCS water
- Accumulator water (both inject)
- Pumped injection (SI)

The energy inventories considered in the LOCA M&E release analysis are given in Tables 2.6.3.1-7 and 2.6.3.1-13. The energy sources are the following:

- RCS water
- Accumulator water (both inject)
- Pumped injection (SI)
- Decay heat
- Core-stored energy
- RCS metal (includes steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary: feedwater pump coastdown after the signal to close the flow control valve)

The analysis used the following energy reference points:

- Available energy: 212°F: 14.7 psia (energy available that could be released)
- Total energy content: 32°F; 14.7 psia (total internal energy of the RCS)

The M&E inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End-of-blowdown time
- End-of-refill time
- End-of-reflood time
- Time of broken-loop steam generator equilibration to pressure setpoint
- Time of intact-loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

The energy release from the zirc-water reaction is considered as part of the WCAP-10325-P-A (reference 1) methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point were the zirc-water reaction is significant. This is in contrast to the 10CFR50.46 analyses, which are biased to calculate high fuel-rod-cladding temperatures and therefore a non-significant zirc-water reaction. For the LOCA M&E calculation, the energy created by the zirc-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the M&E releases, and is therefore already included in the LOCA M&E release.

The sequence of events for the LOCA transients are shown in Tables 2.6.3.1-14 and 2.6.3.1-15.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The analysis performed to assess the containment response to the limiting LOCA resulting from operation at EPU conditions does not add any new components or introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The analytical results associated with operating at EPU conditions do not add any new or previously unevaluated materials to the plant systems. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

2.6.3.1.2.1.4 Results

The LOCA M&E releases from accident initiation to the depressurization down to atmospheric pressure at 3600 seconds, that is, 14.7 psia and 212°F where applicable, have been provided for the DEHL and for the DEPS break cases.

The M&E release transients for the limiting transients are presented in Tables 2.6.3.1-5 and 2.6.3.1-8.

The results of this analysis (M&E release rate transients) were used in the containment integrity analysis (see LR section 2.6.1, Primary Containment Functional Design).

The consideration of the various energy sources listed in LR section 2.6.3.1.2.1.3, Description of Analyses and Evaluations, for the long-term M&E release analysis provides assurance that all available sources of energy have been included in this analysis. By addressing all available sources of energy as well as the limiting break size and location and the specific modeling of each phase of the long-term LOCA transient, the review guidelines presented in SRP section 6.2.1.3 have been satisfied.

2.6.3.1.2.1.5 M&E Release Analysis for Postulated LOCA References

1. WCAP-10325-P-A, May 1983 (Proprietary) and WCAP-10326-A (Nonproprietary), *Westinghouse LOCA Mass and Energy Release Model for Containment Design*, March 1979.

2. WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary) and WCAP-8312-A, Rev. 2 (Nonproprietary) *Topical Report Westinghouse Mass and Energy Release Data Containment Design*.

3. Docket No. 50-315, Amendment No. 126, *Facility Operating License No. DPR-58 (TAC No. 71062)*, for D. C. Cook Nuclear Plant Unit 1, June 9, 1989.

4. EPRI 294-2, *Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary*, WCAP-8423, Final Report, June 1975.

5. ANSI/ANS-5.1 1975, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

2.6.3.1.3 Short-Term LOCA M&E Releases

2.6.3.1.3.1 Technical Evaluation

An evaluation was conducted to determine the effect of the Ginna EPU program on the short-term LOCA-related M&E releases that support the analysis of subcompartments discussed in the Ginna UFSAR section 6.2.1.3. LR section 2.6.2 discusses sub-compartment analysis in more detail.

2.6.3.1.3.1.1 Introduction

The containment internal structures are designed for a pressure buildup that could occur following a postulated LOCA. If a LOCA were to occur in these relatively small volumes, the pressure would build up at a faster rate than the overall containment, thus imposing a differential pressure across the walls of the compartments. The evaluation of the containment internal structures is discussed in section 6.2.1.3 of the Ginna UFSAR.

Short-term LOCA M&E release calculations are performed to support analysis of the RCL compartments (UFSAR section 6.2.1.3.2), the concrete around and under the reactor vessel (UFSAR section 6.2.1.3.4), and the concrete structures around the steam generator (UFSAR section 6.2.1.3.4). The breaks assumed in the current licensing basis for these structures are large primary loop pipe breaks (a longitudinal split of area equivalent to the cross-sectional area of a reactor coolant pipe, i.e., 4.5 ft^2). These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short-pressure pulse (generally less than 3 seconds) that accompanies a LOCA within the region.

Ginna has been approved for LBB methods as part the NRC staff's resolution of unresolved safety issue A-2 (reference 1). With the elimination of the large RCS breaks, the only break locations that need to be considered are the largest branch lines off of the primary loop piping. These branch lines are the pressurizer surge line, the pressurizer spray line, and the accumulator lines. The releases from these smaller breaks are considerably lower than would result from the large RCS breaks. Note that requests to eliminate the remaining 10 inch attached lines from the postulated dynamic effects due to leak-before-break have been submitted to the NRC for review and approval.

2.6.3.1.3.1.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters and Assumptions

The short-term LOCA M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures were chosen to bound the temperature range of all operating cases, and a temperature uncertainty allowance (-4°F) was then included. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+60 psi). All input parameters are chosen consistent with accepted analysis methodology. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with increasing pressures and decreasing temperatures, the short-term LOCA releases are expected to increase due to changes associated with the current RCS conditions.

Increased power has no impact on the short-term releases because of the duration of the event (i.e., ~1.0 second). Only changes in the initial RCS pressure and temperature conditions would affect the results.

For the M&E releases, the core-stored energy and flow behavior through the core have the potential of changing as a result of a fuel change. However, any changes to the flow characteristics past the fuel are assumed small, and as such, would have an insignificant impact on the short-term LOCA M&E releases. Any possible change in the core-stored energy does not adversely affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to the short-term LOCA M&E releases. This change does not create conditions more limiting than those assumed in the analyses. Any change in core-stored energy would have no affect on the releases because of the short duration of the postulated accident.

Therefore, the only effect that needs to be addressed is the change in RCS coolant temperatures and RCS coolant pressure.

In summary, the following assumptions were employed to ensure that the M&E releases were conservatively calculated, thereby maximizing mass release to containment subcompartments

- RCS vessel outlet temperature goes from 609.8°F to 600.3°F
- RCS vessel/core inlet temperature goes from 552.5°F to 528.3°F
- Allowance for RCS temperature uncertainty is ±4.0°F
- Allowance for RCS pressure uncertainty is ± 60 psi.

Acceptance Criteria

Although Ginna is not a Standard Review Plan (SRP) plant, for completeness the SRP long-term cooling criterion is also examined. An LBLOCA is classified as an ANS Condition IV event – an infrequent fault. To satisfy the NRC acceptance criteria presented in the SRP section 6.2.1.3, the relevant requirements are as following:

- The NRC's NUREG-0800, Section 6.2.1.3, "M&E Release Analysis for Postulated Loss-of-Coolant Accidents" subsection II, Part 3a provides guidance on NRC's expectations for what must be included in A LOCA M&E release calculation, if that calculation is to be acceptable. The Westinghouse M&E models described in WCAP-8264-P-A, Rev. 1 (reference 2) have been found by the NRC to satisfy those expectations.

2.6.3.1.3.1.3 Description of Analysis and Evaluations

Description of Analysis

Short-term releases are linked directly to the critical mass flux, which increases with increasing pressures and decreasing temperatures. The short-term LOCA releases are expected to increase due to changes associated with the current RCS conditions. Short-term blowdown transients are characterized by a peak M&E release rate that occurs during a subcooled condition; thus the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA M&E release analyses (reference 2). This correlation was used to conservatively evaluate the impact of the deviations in the RCS inlet and outlet temperature for the EPU program. Therefore, using lower temperatures maximizes the short-term LOCA M&E releases.

Short-term LOCA M&E release calculations are performed to support analysis of the RCL compartments (UFSAR section 6.2.1.3.2), the concrete around and under the reactor vessel (UFSAR section 6.2.1.3.4), and the concrete structures around the steam generator (UFSAR section 6.2.1.3.4). The breaks assumed in the current licensing basis for these structures are large primary loop pipe breaks (a longitudinal split of area equivalent to the cross-sectional area of a reactor coolant pipe, i.e., 4.5 ft^2).

Using a double-ended break in the largest branch line, rather than the current single-ended split in the RCS, reduces the break size by a factor of 4. A reduction of this magnitude in pipe break size has a significant beneficial impact on the subcompartment loadings. For example, based upon available sensitivities, the peak break compartment pressure was reduced by a factor of 2.76, and the peak differential across an adjacent wall was reduced by a factor of 3.86.

Using the EPU pressure and temperature (P/T) conditions (i.e., increased pressure and decreased temperature), the lower temperature increases the short-term LOCA M&E release. For conservatism, uncertainties were applied only to the EPU data to maximize the possible

increase. Using the critical mass flux equation, it was determined that the release could increase by 9.7%. Therefore, from a very conservative perspective, the current design basis releases could increase by a factor of 1.10 due to RCS initial condition (P/T) effects.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The analysis performed to assess the containment response to the limiting LOCA resulting from operation at EPU conditions does not add any new components or introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The analytical results associated with operating at EPU conditions do not add any new or previously unevaluated materials to the plant systems. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

2.6.3.1.3.1.4 Short-Term LOCA M&E Releases Results

In summary, the effect of eliminating the large RCS breaks and instead considering the branch nozzles is a factor of 4 reduction in the break area, whereas the penalty associated with the uprate is a release increase of only a factor of 1.10. So the benefit of LBB more than offsets any penalty to the current licensing basis short-term mass & energy and subcompartment analysis associated with uprating.

The current licensing basis LOCA M&E releases used for the subcompartment results for the RCL compartments, the reactor vessel compartment, and the steam generator compartment remain bounding for the short-term subcompartment analysis by virtue of applying LBB methods.

2.6.3.1.3.1.5 Short-Term LOCA M&E Releases References

1. NRC SER Subject: *Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04)* issued to A-2 Users Group Surry and Surry 2 included, February 1, 1984.

2. WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary) and WCAP-8312-A, Rev. 2 (Nonproprietary), *Topical Report Westinghouse Mass and Energy Release Data Containment Design.*

3. WCAP-12035, *Containment Subcompartment Analysis Utilizing Leak Before Break Technology for Watts Bar Units 1 and 2*, November 1988.

2.6.3.1.4 Conclusion

The Ginna staff has reviewed the M&E release assessment and concludes that it has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10CFR50, Appendix K. Based on this, the Ginna staff finds that the M&E release analysis will continue to meet the Ginna Station current licensing basis with respect to the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to M&E release for postulated LOCA.

Table 2.6.3.1-1
System Parameters Initial Conditions

Parameters	Value
Core Thermal Power (MWt)	1811.0
RCS Total Flow Rate (lbm/sec)	18,000.0
Vessel Outlet Temperature[a] (°F)	615.8
Core Inlet Temperature[a] (°F)	544.2
Vessel Average Temperature[a] (°F)	580.
Initial Steam Generator Steam Pressure (psia)	855.0
Steam Generator Design	BWI
SGTP (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	108,548.0
Assumed Maximum Containment Backpressure (psia)	74.7
Accumulator Water volume (ft^3) per accumulator (minimum)[b] N_2 cover gas pressure (psia) (minimum) Temperature (°F)	 1,090 714.7 120
SI Start Time, (sec) [total time from beginning of event, which includes the maximum delay from reaching the setpoint]	35.1
Auxiliary Feedwater Flow (gpm/steam generator) (Minimum Safeguards)	0
Auxiliary Feedwater Flow (gpm/steam generator) (Maximum Safeguards)	170

Notes:

Core thermal power, RCS total flow rate, RCS coolant temperatures, and steam generator secondary side mass include appropriate uncertainty and/or allowance.

a. RCS coolant temperatures include +4.0°F allowance for instrument error and deadband.

b. Does not accumulate line volume.

Table 2.6.3.1-2
SI Flow Minimum Safeguards

RCS Pressure (psia)	Total Flow (gpm)
Injection Mode (Reflood phase)	
14.7	1800
20	1776
40	1683
60	1580
80	1466
100	1335
120	1170
140	820
214.7	600
314.7	600
414.7	600
514.7	600
Recirculation Mode	
RCS Pressure (psia)	**Total Flow (gpm)**
14.7	1000

Table 2.6.3.1-3
SI Flow Maximum Safeguards

RCS Pressure (psia)	Total Flow (gpm)
Injection Mode (Reflood phase)	
14.7	4452
20	4449.8
40	4441.6
60	4433.4
80	4425.2
100	4417
114.7	4411
120	4408.7
140	4400.1
175	4385
176	994.5
214.7	978
314.7	933
414.7	885.5
514.7	838
Recirculation Mode	
RCS Pressure (psia)	**Total Flow (gpm)**
0	3000

Table 2.6.3.1-4
LOCA M&E Release Analysis Core Decay Heat Fraction

Time (sec)	Decay Heat Generation Rate (Btu/Btu)
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1000	0.022156
1500	0.019921
2000	0.018315
4000	0.014781
6000	0.013040
8000	0.012000
10,000	0.011262
15,000	0.010097
20,000	0.009350
40,000	0.007778
60,000	0.006958
80,000	0.006424
100,000	0.006021
150,000	0.005323
200,000	0.004847
400,000	0.003770
600,000	0.003201
800,000	0.002834
1,000,000	0.002580
2,000,000	0.001909
4,000,000	0.001355

Table 2.6.3.1-5 DEHL Break Blowdown M&E Release				
Time	Break Path No. 1		Break Path No. 2	
Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
.00000	.0	.0	.0	.0
.00109	45,457.3	28,785.6	45,455.5	28,783.2
.101	36,063.4	23,163.9	25,677.4	16,219.2
.201	33,800.9	21,645.9	22,336.1	14,001.1
.301	32,833.4	20,979.7	19,868.1	12,261.1
.402	31,610.9	20,196.8	18,661.6	11,310.7
.501	31,274.7	19,985.5	17,898.3	10,669.0
.601	30,698.1	19,663.8	17,385.8	10,215.2
.701	30,310.4	19,505.9	17,009.7	9874.1
.801	29,844.8	19,345.3	16,679.6	9585.4
.901	28,872.3	18,851.0	16,477.0	9387.4
1.00	27,824.6	18,308.2	16,277.0	9205.7
1.10	26,809.0	17,784.7	16,129.4	9063.7
1.20	25,851.9	17,296.6	16,022.8	8952.7
1.30	24,822.5	16,743.8	15,962.9	8873.9
1.40	23,698.5	16,111.4	15,943.0	8821.7
1.50	22,553.9	15,455.6	15,959.0	8793.5
1.60	21,455.5	14,818.7	15,992.1	8778.5
1.70	20,757.1	14,328.7	16,036.5	8773.1
1.80	19,932.5	13,942.1	16,086.6	8774.5
1.90	19,023.9	13,584.7	16,131.0	8776.2
2.00	18,017.9	13,011.0	16,162.7	8774.3
2.10	17,499.7	12,604.2	16,181.2	8768.5
2.20	17,303.4	12,315.3	16,187.3	8758.8

Table 2.6.3.1-5 (cont.) DEHL Break Blowdown M&E Release				
Time	Break Path No. 1		Break Path No. 2	
Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
2.30	17,225.3	12,096.1	16,179.1	8744.0
2.40	17,129.8	11,900.3	16,154.8	8722.8
2.50	16,983.2	11,709.9	16,112.0	8693.6
2.60	16,796.7	11,522.1	16,048.5	8655.0
2.70	16,609.4	11,349.0	15,960.9	8605.0
2.80	16,461.9	11,211.0	15,847.2	8542.3
2.90	16,360.3	11,112.7	15,685.1	8454.7
3.00	16,299.2	11,031.7	15,461.6	8335.2
3.10	16,295.3	10,969.6	15,213.9	8204.0
3.20	16,349.6	10,926.9	14,950.7	8065.9
3.30	16,431.3	10,897.5	14,671.5	7920.1
3.40	16,522.0	10,875.7	14,357.6	7756.6
3.50	16,602.5	10,854.5	14,007.6	7574.4
3.60	16,644.1	10,823.5	13,621.9	7373.8
3.70	16,628.3	10,775.4	13,206.9	7158.5
3.80	16,607.8	10,728.5	12,783.4	6939.5
3.90	16,595.2	10,686.3	12,373.6	6729.1
4.00	16,598.2	10,653.3	11,951.4	6513.2
4.20	16,575.2	10,565.0	11,147.6	6104.9
4.40	16,634.3	10,508.3	10,377.0	5714.6
4.60	13,136.0	8892.5	9626.5	5333.4
4.80	13,207.3	8900.1	8955.0	4993.5
5.00	13,012.9	8741.7	8368.2	4698.1
5.20	12,813.9	8577.5	7863.5	4445.5
5.40	12,670.8	8427.1	7430.3	4230.0
5.60	12,553.4	8322.4	7049.9	4041.3
5.80	12,434.9	8164.3	6710.5	3873.2
6.00	11,923.0	7871.6	6408.3	3724.2

Table 2.6.3.1-5 (cont.) DEHL Break Blowdown M&E Release				
Time	Break Path No. 1		Break Path No. 2	
Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
6.20	11,716.9	7710.2	6129.4	3587.0
6.40	11,461.3	7524.4	5874.6	3462.6
6.60	11,165.5	7317.4	5631.2	3344.5
6.80	10,818.8	7087.4	5402,5	3234.5
7.00	10,411.8	6835.3	5180.6	3128.4
7.20	9942.7	6561.2	4951.6	3019.3
7.40	9443.7	6281.5	4710.5	2906.0
7.60	8934.8	6001.7	4454.9	2789.6
7.80	8451.3	5723.3	4198.1	2678.4
8.00	8063.8	5505.1	3935.8	2569.4
8.20	7578.3	5262.4	3675.9	2464.0
8.40	7124.4	5024.6	3421.4	2361.8
8.60	6602.0	4755.4	3181.5	2265.4
8.80	5958.5	4470.8	2972.3	2179.5
9.00	5335.9	4203.1	2763.2	2081.1
9.20	4784.3	3944.6	2564.0	1991.7
9.40	4272.1	3704.0	2370.0	1913.0
9.60	3767.2	3477.3	2180.8	1836.8
9.80	3248.5	3244.6	2009.0	1761.3
10.0	2801.4	2893.7	1863.6	1691.6
10.2	2600.4	2674.8	1741.5	1633.2
10.4	2400.2	2491.3	1621.4	1582.6
10.6	2149.3	2317.4	1508.2	1539.8
10.8	1895.0	2135.2	1384.6	1495.4
11.0	1691.1	1956.8	1263.1	1450.1
11.2	1534.5	1782.8	1116.0	1345.1
11.4	1369.9	1621.5	969.5	1190.9
11.6	1213.2	1466.9	866.5	1070.0

Table 2.6.3.1-5 (cont.) DEHL Break Blowdown M&E Release				
Time	Break Path No. 1		Break Path No. 2	
Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
11.8	1082.4	1323.6	822.7	1018.1
12.0	955.9	1179.0	794.5	984.4
12.2	818.9	1022.2	766.4	950.4
12.4	620.6	776.5	734.9	912.3
12.6	488.8	614.0	702.3	872.7
12.8	409.4	516.8	663.7	825.5
13.0	367.6	464.9	580.1	722.0
13.2	343.4	434.6	464.0	578.8
13.4	325.5	412.3	373.7	467.2
13.6	309.1	392.0	283.4	355.2
13.8	308.1	391.2	198.2	249.3
14.0	293.1	372.5	142.4	179.9
14.2	285.7	363.3	93.3	118.4
14.4	280.3	356.6	71.3	91.1
14.6	270.0	343.7	75.4	96.5
14.8	253.2	322.5	83.3	106.7
15.0	235.9	300.8	79.9	102.2
15.2	221.3	282.6	54.7	70.2
15.4	171.1	219.0	51.3	66.3
15.6	122.0	156.7	.0	.0
15.8	59.6	77.1	.0	.0
16.0	.0	.0	.0	.0

Notes:
Path 1: M&E exiting from the reactor vessel side of the break.
Path 2: M&E exiting from the steam generator side of the break.

Table 2.6.3.1-6
DEHL Break Mass Balance

Time (Seconds)		0.00	16.00	16.00
		Mass (thousand lbm)		
Initial	In RCS and ACC	404.07	404.07	404.07
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
*****Total Available*****		404.07	404.07	404.07
Distribution	Reactor Coolant	265.88	46.01	52.17
	Accumulator	138.19	103.32	97.17
	Total Contents	404.07	149.33	149.33
Effluent	Break Flow	.00	254.73	254.73
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	254.73	254.73
*****Total Accountable*****		404.07	404.06	404.06

Table 2.6.3.1-7
DEHL Break Energy Balance

Time (seconds)		0.00	16.00	16.00
		Energy (million Btu)		
Initial Energy	In RCS, ACC, S GEN	427.66	427.66	427.66
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	2.83	2.83
	Heat from Secondary	.00	5.53	5.53
	Total Added	.00	8.36	8.36
Total Available		427.66	436.02	436.02
Distribution	Reactor Coolant	154.08	10.64	11.19
	Accumulator	12.41	9.28	8.72
	Core Stored	15.33	6.53	6.53
	Primary Metal	84.08	79.25	79.25
	Secondary Metal	44.27	42.64	42.64
	Steam Generator	117.49	121.50	121.50
	Total Contents	427.66	269.85	269.85
Effluent	Break Flow	.00	165.84	165.84
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	165.84	165.84
Total Accountable		427.66	435.69	435.69

Table 2.6.3.1-8 DEPS Break Blowdown M&E Release (Minimum Safeguards Cases)				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
.00000	.0	.0	.0	.0
.00104	83,075.6	44,493.8	39,106.8	20,905.6
.00202	39,932.0	21,348.0	39,666.9	21,204.2
.00301	39,324.4	21,344.2	39,452.6	21,088.8
.00413	39,918.0	21,341.4	39,221.9	20,964.4
.102	39,667.4	21,279.0	19,428.5	10,276.2
.201	40,334.9	21,785.9	21,138.0	11,297.9
.301	41,046.6	22,365.5	22,325.7	11,940.5
.401	41,812.0	23,023.0	22,699.8	12,146.4
.501	42,557.9	23,712.9	22,462.6	12,025.3
.601	42,991.0	24,243.6	22,060.0	11,816.2
.701	42,937.6	24,490.8	21,663.8	11,609.4
.801	42,201.1	24,320.8	21,345.8	11,443.8
.901	41,118.0	23,926.3	21,114.6	11,323.4
1.00	39,939.2	23,454.5	20,913.9	11,217.9
1.10	38,683.6	22,918.9	20,691.3	11,099.4
1.20	37,305.8	22,291.5	20,423.2	10,955.4
1.30	35,738.3	21,529.6	20,115.6	10,789.6
1.40	33,973.1	20,632.7	19,874.6	10,659.6
1.50	32,557.4	19,953.8	19,660.2	10,544.2
1.60	31,505.1	19,524.5	19,397.4	10,402.6
1.70	30,427.3	19,100.0	19,069.8	10,225.6
1.80	29,158.6	18,567.8	18,718.6	10,035.6
1.90	27,565.7	17,846.1	18,356.7	9840.0

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release (Minimum Safeguards Case)				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
2.00	23,148.7	15,213.2	17,985.0	9639.2
2.10	19,124.9	12,792.0	17,590.3	9426.1
2.20	16,707.8	11,353.2	17,189.4	9210.1
2.30	14,799.5	10,160.7	16,877.7	9043.1
2.40	13,461.3	9311.9	16,663.0	8929.2
2.50	12,634.0	8793.4	16,485.8	8836.1
2.60	12,104.7	8465.4	16,071.9	8614.5
2.70	11,692.0	8209.9	15,623.7	8375.1
2.80	11,302.6	7970.5	15,300.4	8203.9
2.90	10,929.3	7751.4	15,033.5	8063.6
3.00	10,572.0	7555.3	14,795.0	7938.5
3.10	10,221.3	7373.2	14,577.3	7824.9
3.20	9889.7	7208.0	14.377.3	7720.8
3.30	9574.2	7052.2	14,195.7	7626.9
3.40	9286.6	6909.7	14,011.0	7531.2
3.50	9035.6	6784.2	13,834.1	7439.8
3.60	8816.0	6672.8	13,664.1	7352.4
3.70	8626.3	6574.2	13,887.8	7480.5
3.80	8462.3	6487.1	14,040.1	7567.5
3.90	8321.8	6410.3	14,001.3	7550.3
4.00	8195.7	6338.5	14,018.8	7564.6
4.20	7986.7	6214.3	13,988.7	7557.5
4.40	7809.8	6097.7	13,849.8	7490.3
4.60	7637.8	5971.2	13,567.9	7344.5
4.80	7460.0	5822.1	13,223.6	7165.2

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release (Minimum Safeguards Case)				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
5.00	7496.8	5807.6	12,823.5	6956.2
5.20	7692.0	5984.7	12,482.3	6780.3
5.40	7139.5	5933.9	11,988.8	6520.8
5.60	6460.8	5639.7	11,465.2	6244.2
5.80	6129.1	5403.6	11,047.4	6024.4
6.00	5924.6	5193.2	10,617.1	5797.5
6.20	5746.3	4983.2	10,200.1	5577.9
6.40	5561.9	4766.3	9761.2	5345.6
6.60	5369.0	4538.9	9338.9	5121.2
6.80	5183.2	4318.0	8946.7	4894.6
7.00	4988.8	4099.7	8598.3	4648.5
7.20	4788.6	3896.0	8327.2	4412.1
7.40	4641.0	3729.8	8415.4	4356.4
7.60	4529.0	3588.7	8242.6	4180.1
7.80	4431.0	3477.1	8230.8	4101.3
8.00	4325.1	3379.8	7817.0	3835.6
8.20	4212.2	3292.9	7763.0	3753.9
8.40	4091.5	3213.3	7154.9	3415.6
8.60	3966.3	3139.5	7156.5	3370.7
8.80	3832.9	3070.7	6731.2	3134.3
9.00	3693.6	3007.4	6485.7	2984.6
9.20	3547.2	2950.1	6293.3	2860.8
9.40	3391.9	2899.7	5905.4	2653.9
9.60	3225.8	2853.2	5595.1	2485.0
9.80	3050.9	2813.2	5396.6	2366.5
10.0	2862.9	2777.7	5148.1	2228.5

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release (Minimum Safeguards Case)				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
10.2	2658.3	2745.7	4854.9	2074.4
10.4	2415.5	2690.7	4556.4	1918.8
10.6	2091.8	2522.3	4329.3	1785.7
10.8	1747.9	2154.5	4131.7	1659.5
11.0	1506.4	1866.6	3909.9	1525.9
11.2	1316.9	1636.5	3643.7	1381.4
11.4	1123.5	1400.5	3338.8	1230.2
11.6	949.2	1185.5	2994.9	1072.5
11.8	805.2	1007.2	2685.8	938.1
12.0	682.7	855.0	2368.2	809.6
12.2	567.3	711.1	2042.8	683.2
12.4	463.6	581.7	1695.8	554.4
12.6	368.7	462.9	1320.4	422.4
12.8	263.0	330.5	910.1	286.1
13.0	157.4	198.1	478.2	148.7
13.2	47.2	59.6	70.8	22.0
13.4	.0	.0	.0	.0

Table 2.6.3.1-9 DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
13.41	.0	.0	.0	.0
13.9	.0	.0	.0	.0
14.1	.0	.0	.0	.0
14.2	.0	.0	.0	.0
14.3	.0	.0	.0	.0
14.4	64.6	76.3	.0	.0
14.5	16.2	19.2	.0	.0
14.7	8.7	10.2	.0	.0
14.8	13.4	15.8	.0	.0
14.9	18.6	22.0	.0	.0
15.0	23.8	28.1	.0	.0
15.1	29.4	34.8	.0	.0
15.2	33.9	40.0	.0	.0
15.3	38.1	45.0	.0	.0
15.4	42.2	49.8	.0	.0
15.5	46.1	54.5	.0	.0
15.6	49.9	58.9	.0	.0
15.7	53.5	63.2	.0	.0
15.8	56.9	67.2	.0	.0
15.9	59.7	70.6	.0	.0
16.0	62.5	73.8	.0	.0
16.1	65.1	76.9	.0	.0
16.2	67.7	80.0	.0	.0
16.3	70.1	82.9	.0	.0
16.4	72.6	85.8	.0	.0

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
16.5	74.9	88.5	.0	.0
17.5	95.7	113.1	.0	.0
18.5	112.9	133.5	.0	.0
19.5	127.6	150.8	.0	.0
20.1	136.3	161.1	.0	.0
20.5	141,1	166.9	.0	.0
21.5	163.1	193.0	662.8	81.2
22.5	264.7	313.5	2622.0	324.2
23.5	269.9	319.6	2677.8	334.0
24.5	266.6	315.0	2638.3	330.2
25.0	264.9	313.7	2616.6	328.1
25.5	263.1	311.6	2595.0	325.9
26.5	259.7	307.6	2552.1	321.7
27.5	256.4	303.7	2510.1	317.5
27.5	253.3	299.9	2469.2	313.4
29.5	250.2	296.3	2429.4	309.4
30.5	247.3	292.8	2390.7	305.5
31.5	244.5	289.5	2353.2	301.8
32.5	241.8	286.3	2316.7	298.1
33.5	239.3	283.3	2,281.3	294.5
34.5	236.8	280.3	2246.9	291.1
35.5	245.5	290.7	2390.3	300.6
36.5	243.2	287.9	2358.2	297.4
36.6	243.0	287.6	2355.0	297.1
37.5	240.9	285.2	2326.9	294.3

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
Time sec	Break Path No. 1		Break Path No. 2	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
38.5	238.8	282.7	2296.5	291.2
39.5	236.7	280.2	2266.8	288.3
40.5	234.7	277.8	2238.0	285.4
41.5	232.8	275.5	2209.8	282.5
42.5	230.9	273.3	2182.4	279.8
43.0	230.0	272.2	2168.9	278.4
43.5	196.4	232.4	1054,5	178.2
44.5	170.7	201.9	1092.0	171.4
45.6	204.6	242.1	177.4	96.6
46.6	202.3	239.4	176.5	95.5
47.6	200.0	236.6	175.5	94.4
48.6	197.6	233.9	174.6	93.2
49.6	195.3	231.1	173.6	92.1
50.6	192.9	228.2	172.7	90.9
51.6	190.5	225.4	171.7	89.8
52.6	188.2	222.6	170.8	88.6
53.6	185.8	219.9	169.9	87.5
54.6	183.5	217.1	169.0	86.4
55.6	181.2	214.3	168.0	85.3
56.6	178.8	211.5	167.1	84.1
57.6	176.4	208.7	166.2	83.0
57.9	175.7	207.9	165.9	82.7
58.6	174.1	205.9	165.3	81.9
59.6	171.7	203.1	164.4	80.7
60.6	169.3	200.3	163.4	79.6
61.6	166.9	197.4	162.5	78.5

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
62.6	164.5	194.6	161.6	77.4
63.6	162.1	191.7	160.7	76.2
64.6	159.7	188.9	159.7	75.1
65.6	157.3	186.0	158.8	74.0
66.6	154.9	183.2	157.9	72.9
67.6	152.4	180.3	157.0	71.8
68.6	150.0	177.4	156.1	70.7
69.6	147.6	174.5	155.2	69.5
70.6	145.2	171.7	154.2	68.4
71.6	142.7	168.8	153.3	67.3
72.6	140.3	165.9	152.4	66.2
73.6	137.9	163.1	151.5	65.2
74.6	135.5	160.2	150.6	64.1
75.5	133.3	157.6	149.8	63.1
76.6	130.7	154.5	148.9	61.9
78.6	125.9	148.8	147.1	59.8
80.6	121.1	143.2	145.4	57.8
82.6	116.4	137.6	143.7	55.7
84.6	111.7	132.1	142.1	53.7
86.6	107.2	126.7	140.5	51.8
88.6	102.6	121.3	138.9	49.9
90.6	98.2	116.1	137.4	48.1
92.6	93.9	111.0	136.0	46.3
94.6	89.6	106.0	134.6	44.6
96.6	85.5	101.0	133.2	43.0
98.4	81.9	96.8	132.0	41.6

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
98.6	81.5	96.3	131.9	41.4
100.6	77.6	91.7	130.7	39.9
102.6	73.9	87.3	129.5	38.5
104.6	70.2	83.0	128.4	37.2
106.6	66.8	78.9	127.3	35.9
108.6	63.9	75.5	126.3	34.7
110.6	62.4	73.7	125.3	33.5
112.6	60.9	72.0	124.4	32.4
114.6	59.5	70.4	123.5	31.3
116.6	58.2	68.8	122.6	30.3
118.6	56.9	67.3	121.7	29.3
120.6	55.7	65.8	120.9	28.3
122.6	54.5	64.4	120.1	27.4
124.6	53.4	63.1	119.4	26.5
126.6	52.4	61.9	118.7	25.6
128.6	51.4	60.7	118.0	24.8
130.6	50.4	59.6	117.3	24.0
131.2	50.2	59.3	117.1	23.8
132.6	49.6	58.6	116.7	23.3
134.6	48.7	57.6	116.1	22.6
136.6	47.9	56.7	115.5	21.9
138.6	47.2	55.8	115.0	21.3
140.6	46.5	55.0	114.5	20.7
142.6	45.9	54.2	114.0	20.2
144.6	45.3	53.5	113.6	19.6
146.6	44.7	52.8	113.2	19.1

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
148.6	44.2	52.2	112.8	18.7
150.6	43.7	51.7	112.4	18.2
152.6	43.3	51.1	112.0	17.8
154.6	42.8	50.6	111.7	17.4
156.6	42.5	50.2	111.4	17.0
158.6	42.1	49.8	111.1	16.7
160.6	41.8	49.4	110.8	16.4
162.6	41.5	49.0	110.6	16.1
164.6	41.2	48.7	110.3	15.8
166.6	41.0	48.4	110.1	15.5
168.6	40.8	48.2	109.9	15.3
170.6	40.6	47.9	109.7	15.1
172.6	40.4	47.7	109.5	14.9
173.9	40.3	47.6	109.4	14.7
174.6	40.2	47.5	109.4	14.7
176.6	40.1	47.3	109.2	14.5
178.6	39.9	47.2	109.1	14.3
180.6	39.8	47.1	109.0	14.2
182.6	39.7	46.9	108.8	14.0
184.6	39.6	46.8	108.7	13.9
186.6	39.6	46.7	108.6	13.8
188.6	39.5	46.7	108.5	13.6
190.6	39.4	46.6	108.4	13.5
192.6	39.4	46.6	108.4	13.4
194.6	39.4	46.5	108.3	13.4
196.6	39.3	46.5	108.2	13.3

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
Time sec	Break Path No. 1		Break Path No. 2	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
198.6	39.3	46.5	108.1	13.2
200.6	39.3	46.4	108.1	13.1
202.6	39.3	46.4	108.0	13.1
204.6	39.3	46.4	108.0	13.0
206.6	39.3	46.4	107.9	13.0
208.6	39.3	46.4	107.9	12.9
210.6	39.3	46.5	107.8	12.9
212.6	39.3	46.5	107.8	12.8
214.6	39.4	46.5	107.8	12.8
216.6	39.4	46.5	107.8	12.7
218.6	39.4	46.6	107.7	12.7
220.6	39.4	46.6	107.7	12.7
222.1	39.5	46.6	107.7	12.7

Table 2.6.3.1-10
DEPS - Minimum SI Principle Parameters During Reflood

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total	Injection Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
13.4	151.3	.000	.000	.00	.00	.500	.0	.0	.0	.00
14.1	150.7	22.832	.000	.57	1.07	.000	3960.8	3960.8	.0	89.79
14.3	150.2	25.061	.000	.98	1.14	.000	3939.9	3939.9	.0	89.79
14.31	150.1	24.834	.000	1.08	1.15	.000	3929.5	3929.5	.0	89.79
15.0	150.1	2.660	.167	1.39	2.37	.391	3856.6	3856.6	.0	89.79
15.1	150.2	2.662	.190	1.40	2.58	.429	3844.4	3844.4	.0	89.79
15.7	150.4	2.607	.292	1.50	3.86	.519	3787.3	3787.3	.0	89.79
16.5	150.7	2.534	.393	1.62	5.53	.555	3714.6	3714.6	.0	89.79
20.1	152.3	2.831	.598	2.00	12.79	.595	3426.9	3426.9	.0	89.79
22.5	153.4	4.086	.666	2.24	15.81	.721	3202.9	3202.9	.0	89.79
24.5	154.4	3.948	.690	2.45	15.83	.722	3080.8	3080.8	.0	89.79
25.0	154.6	3.905	.693	2.51	15.83	.721	3053.2	3053.2	.0	89.79
30.5	157.6	3.586	.715	3.01	15.83	.712	2784.8	2784.8	.0	89.79
34.5	159.9	3.438	.720	3.34	15.83	.705	2621.5	2621.5	.0	89.79
35.5	160.4	3.516	.723	3.42	15.83	.713	2775.8	2574.5	.0	88.50
36.6	161.1	3.483	.723	3.51	15.83	.711	2736.1	2534.7	.0	88.48
43.0	164.9	3.323	.726	4.00	15.83	.703	2529.0	2326.7	.0	88.37

Table 2.6.3.1-10 (cont.)
DEPS - Minimum Safety Injection Principle Parameters During Reflood

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total	Injection Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
44.5	165.7	2.749	.717	4.11	15.83	.639	1369.2	1163.7	.0	87.12
45.6	166.3	3.093	.724	4.18	15.73	.674	203.8	.0	.0	72.03
50.6	169.5	2.928	.723	4.52	14.87	.672	204.1	.0	.0	72.03
57.9	174.5	2.691	.720	5.00	13.74	.668	204.6	.0	.0	72.03
66.6	180.9	2.407	.716	5.52	12.59	.663	205.1	.0	.0	72.03
75.5	187.7	2.116	.710	6.00	11.63	.654	205.5	.0	.0	72.03
86.6	196.2	1.766	.702	6.53	10.74	.637	206.0	.0	.0	72.03
98.4	204.8	1431	.691	7.00	10.13	.610	206.3	.0	.0	72.03
114.6	215.7	1.121	.679	7.54	9.76	.577	206.6	.0	.0	72.03
131.2	224.9	.960	.672	8.00	9.69	.574	206.6	.0	.0	72.03
152.6	234.5	.834	.667	8.53	9.85	.574	206.5	.0	.0	72.03
173.9	242.6	.774	.666	9.00	10.18	.577	206.5	.0	.0	72.03
198.6	250.6	.747	.668	9.52	10.65	.582	206.5	.0	.0	72.03
222.1	257.4	.740	.672	10.00	11.13	.587	206.5	.0	.0	72.03

Table 2.6.3.1-11 DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
222.1	89.4	113.9	140.8	51.8
227.1	89.2	113.7	140.7	51.7
232.1	89.0	113.4	140.6	51.6
237.1	88.8	113.1	140.5	51.4
242.1	89.9	114.5	140.4	51.3
247.1	89.7	114.2	140.3	51.2
252.1	89.4	113.9	140.2	51.0
257.1	89.2	113.6	140.0	50.9
262.1	89.0	113.3	139.9	50.8
267.1	88.7	113.0	139.8	50.6
272.1	88.5	112.7	139.7	50.5
277.1	88.3	112.4	139.6	50.4
282.1	89.4	113.9	139.5	50.2
287.1	89.1	113.6	139.4	50.1
292.1	88.9	113.3	139.3	50.0
297.1	88.7	112.9	139.1	49.8
302.1	88.4	112.6	139.0	49.7
307.1	88.2	112.3	138.9	49.6
312.1	87.9	112.0	138.8	49.4
317.1	89.0	113.4	138.7	49.3
322.1	88.8	113.1	138.6	49.2
327.1	88.5	112.8	138.5	49.0
332.1	88.3	112.5	138.3	48.9
337.1	88.1	112.2	138.2	48.7
342.1	87.8	111.9	138.1	48.6

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
347.1	87.6	11.6	138.0	48.5
352.1	88.6	112.9	137.9	48.3
357.1	88.4	112.6	137.8	48.2
362.1	88.1	112.3	137.6	48.0
367.1	87.9	112.0	137.5	47.9
372.1	87.6	111.6	137.4	47.8
377.1	87.4	111.3	137.3	47.6
382.1	87.1	111.0	137.2	47.5
387.1	88.2	112.3	137.0	47.3
392.1	87.9	112.0	136.9	47.2
397.1	87.7	111.7	136.8	47.1
402.1	87.4	111.4	136.7	46.7
407.1	87.2	111.2	136.6	46.8
412.1	87.1	110.9	136.5	46.7
417.1	86.5	110.2	137.8	48.2
422.1	86.7	110.5	137.9	48.4
427.1	86.5	110.2	137.8	48.2
432.1	86.3	110.0	137.6	48.1
437.1	86.2	109.8	137.5	47.9
442.1	86.0	109.5	137.4	47.8
447.1	85.8	109.3	137.3	47.6
452.1	86.9	110.7	137.1	47.5
457.1	86.7	110.4	137.0	47.3
462.1	86.5	110.2	136.9	47.2

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
467.1	86.3	110.0	136.8	47.0
472.1	86.1	109.7	136.6	46.9
477.1	85.9	109.5	136.5	46.7
482.1	85.7	109.2	136.4	46.6
487.1	86.8	110.6	136.3	46.4
492.1	86.6	110.3	136.1	46.3
497.1	86.4	110.1	136.0	46.1
502.1	86.2	109.8	135.9	46.0
507.1	86.0	109.6	135.7	45.8
512.1	85.8	109.3	135.6	45.7
517.1	86.8	110.6	135.5	45.5
522.1	86.6	110.4	135.3	45.3
527.1	86.4	110.1	135.2	45.2
532.1	85.0	108.3	136.7	46.9
537.1	84.8	108.0	136.5	46.8
542.1	85.8	109.3	136.4	46.6
547.1	85.6	109.1	136.3	46.4
552.1	85.4	108.8	136.1	46.3
557.1	85.2	108.5	136.0	46.1
562.1	85.0	108.3	135.8	45.9
567.1	86.0	109.5	135.7	45.7
572.1	85.7	109.2	135.5	45.6
577.1	85.5	109.0	135.4	45.4
582.1	85.3	108.7	135.3	45.2
587.1	85.1	108.4	135.1	45.1

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
592.1	84.9	108.1	135.0	44.9
597.1	85.8	109.4	134.8	44.7
602.1	85.6	109.1	134.7	44.6
607.1	85.4	108.8	134.5	44.4
612.1	85.2	108.6	134.4	44.2
617.1	85.0	108.3	134.2	44.0
622.1	84.8	108.1	135.6	45.7
627.1	84.6	107.8	135.5	45.5
632.1	84.4	107.6	135.3	45.3
637.1	84.2	107.3	135.2	45.1
642.1	85.2	108.5	135.0	44.9
647.1	85.0	108.2	134.8	44.7
652.1	84.7	108.0	134.7	44.6
657.1	84.5	107.7	134.5	44.4
662.1	84.3	107.4	134.4	44.2
667.1	85.2	108.6	134.2	44.0
672.1	85.0	108.3	134.0	43.8
677.1	84.8	108.0	133.9	43.6
682.1	84.6	107.7	133.7	43.4
687.1	84.4	107.5	135.0	45.0
692.1	84.1	107.2	134.9	44.8
697.1	83.9	106.9	134.7	44.6
702.1	84.8	108.0	134.5	44.4
707.1	84.5	107.7	134.3	44.2
712.1	84.3	107.4	134.2	44.0

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
717.1	84.1	107.1	134.0	43.7
722.1	84.9	108.2	133.8	43.5
727.1	84.7	107.9	133.6	43.3
732.1	84.4	107.5	133.5	43.1
737.1	84.2	107.2	133.3	42.9
742.1	83.9	106.9	134.5	44.4
747.1	83.7	106.6	134.3	44.1
752.1	83.4	106.3	134.1	43.9
757.1	84.2	107.3	134.0	43.7
762.1	84.0	107.0	133.8	43.5
767.1	83.7	106.6	133.6	43.3
772.1	84.5	107.6	133.4	43.0
777.1	84.2	107.3	133.2	42.8
782.1	83.9	106.9	133.0	42.6
787.1	84.7	107.8	132.8	42.3
792.1	83.4	106.2	134.0	43.7
797.1	84.1	107.1	133.8	43.5
802.1	83.8	106.8	133.6	43.2
807.1	83.5	106.4	133.3	43.0
812.1	84.2	107.3	133.1	42.7
817.1	83.9	107.0	132.9	42.5
822.1	83.7	106.6	132.7	42.2
827.1	84.3	107.4	132.5	42.0
832.1	83.1	105.8	133.6	43.3
837.1	83.7	106.7	133.4	43.0

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
842.1	83.4	106.3	133.2	42.8
847.1	83.1	105.9	132.9	42.5
852.1	83.7	106.7	132.7	42.2
857.1	83.4	106.3	132.5	42.0
862.1	84.0	107.0	132.3	41.7
867.1	82.8	105.4	133.3	42.9
872.1	83.4	106.2	133.1	42.6
877.1	83.0	105.8	132.8	42.3
882.1	83.6	106.5	132.6	42.0
887.1	83.2	106.0	132.3	41.8
892.1	83.7	106.7	132.1	41.5
897.1	83.4	106.2	133.0	42.6
902.1	83.0	105.7	132.8	42.3
907.1	83.5	106.3	132.5	42.0
912.1	83.1	105.8	132.2	41.7
917.1	83.5	106.4	132.0	41.4
922.1	83.1	105.8	132.9	42.5
927.1	82.7	105.3	132.6	42.1
932.1	83.1	105.8	132.3	41.8
937.1	83.4	106.3	132.1	41.5
942.1	83.0	105.7	131.8	41.1
947.1	82.5	105.1	132.6	42.1
952.1	82.8	105.5	132.3	41.8
957.1	83.1	105.9	132.0	41.4
962.1	82.6	105.2	131.7	41.0

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
967.1	82.9	105.6	132.5	42.0
972.1	82.3	104.9	132.2	41.6
977.1	82.5	105.1	131.9	41.2
982.1	82.7	105.3	131.5	40.8
987.1	82.1	104.6	132.3	41.8
992.1	82.2	104.7	131.9	41.3
997.1	83.0	105.7	131.6	40.9
1002.1	83.0	105.8	131.2	40.5
1007.1	82.3	104.9	131.9	41.3
1012.1	82.3	104.8	131.6	40.9
1017.1	82.9	105.6	131.2	40.4
1022.1	82.1	104.6	131.8	41.2
1027.1	82.6	105.2	131.4	40.7
1032.1	82.3	104.8	132.1	41.4
1037.1	82.6	105.2	131.6	40.9
1042.1	82.8	105.5	131.2	40.4
1047.1	82.3	104.9	131.7	41.1
1052.1	82.3	104.9	131.3	40.5
1057.1	82.2	104.7	131.8	41.1
1062.1	82.5	105.0	131.3	40.6
1067.1	82.0	104.5	131.8	41.4
1072.1	82.4	104.9	131.2	40.5
1077.1	81.9	104.3	131.7	41.0
1082.1	82.1	104.6	131.1	40.3
1087.1	81.7	104.1	131.5	40.7

Table 2.6.3.1-11 (cont.) DEPS Break Minimum SI Post-Reflood M&E Release				
	Break Path No. 1		Break Path No. 2	
Time sec	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
1092.1	82.3	104.9	130.9	40.0
1097.1	82.1	104.6	131.2	40.4
1102.1	81.9	104.3	131.4	40.7
1107.1	81.9	104.4	130.7	39.9
1112.1	81.5	103.9	130.9	40.1
1117.1	39.0	49.7	167.5	50.4
1227.2	48.4	60.4	158.1	47.5
1227.8	48.4	60.4	158.1	47.1
1333.0	48.4	60.4	158.1	47.1
1333.1	46.9	53.9	159.7	13.7
2652.0	40.6	46.8	165.9	14.8
2652.1	40.6	46.8	94.6	19.2
3600.0	37.7	43.3	97.6	19.7

Notes:
Path 1: M&E exiting from the reactor vessel side of the break.
Path 2: M&E exiting from the steam generator side of the break.

Table 2.6.3.1-12 DEPS Break Mass Balance Minimum Safeguards								
Mass Balance								
	Time (seconds)	0.00	13.40	13.40	222.07	1227.16	1333.01	3600.00
Mass (thousand lbm)								
Initial	In RCS and ACC	404.07	404.07	404.07	404.07	404.07	404.07	404.07
Added Mass	Pumped Injection	.00	.00	.00	38.50	246.09	267.95	669.1
	Total Added	.00	.00	.00	38.50	246.09	267.95	669.1
*** Total Available ***		404.07	404.07	404.07	442.57	650.16	672.03	1073.18
Distribution	Reactor Coolant	265.88	17.13	42.46	76.21	76.21	76.21	76.21
	Accumulator	138.19	116.17	90.84	.00	.00	.00	.00
	Total Contents	404.07	133.30	133.30	76.21	76.21	76.21	76.21
Effluent	Break Flow	.00	270.77	270.77	366.36	585.95	607.75	1008.90
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	270.77	270.77	366.36	585.95	607.75	1008.90
*** Total Accountable ***		404.07	404.07	404.07	442.57	662.15	683.96	1085.11

Table 2.6.3.1-13 DEPS Break Energy Balance Minimum Safeguards								
Energy Balance								
Time (seconds)		0.00	13.40	13.40	222.07	1227.16	1333.01	3600.00
Energy (million BTU)								
Initial Energy	In RCS, ACC, Steam Generator	427.66	427.66	427.66	427.66	427.66	427.66	427.66
Added Energy	Pumped Injection	.00	.00	.00	2.77	17.73	19.30	55.80
	Decay Heat	.00	2.39	2.39	15.29	57.65	61.45	129.66
	Heat From Secondary	.00	6.52	6.52	6.52	6.52	6.52	6.52
	Total Added	.00	8.90	8.90	24.58	81.90	87.27	191.97
*** Total Available ***		427.66	436.57	436.57	452.24	509.56	514.93	619.63
Distribution	Reactor Coolant	154.08	4.68	6.96	19.94	19.94	19.94	19.94
	Accumulator	12.41	10.43	8.16	.00	.00	.00	.00
	Core Stored	15.33	10.34	10.34	2.77	2.67	2.63	1.81
	Primary Metal	84.08	80.67	80.67	67.66	40.43	39.01	27.08
	Secondary Metal	44.27	43.57	43.57	41.74	24.25	23.10	16.12
	Steam Generator	117.49	124.75	124.75	118.29	65.10	62.14	42.84
	Total Contents	427.66	274.45	274.45	250.40	152.39	146.84	107.79
Effluent	Break Flow	.00	161.80	161.80	198.08	356.73	357.83	502.77
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	161.80	161.80	198.08	356.73	357.83	502.7
*** Total Accountable ***		427.66	436.25	436.25	448.47	509.12	504.66	610.57

Table 2.6.3.1-14
DEHL Break Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power Are Assumed
1.94	Compensated Pressurizer Pressure for Turbine Trip – 1915 psia Reached
2.9	Low-Pressurizer Pressure SI Setpoint – 1715 psia Reached - Feedwater Isolation Signal
6.35	Broken Loop Accumulator Begins Injecting Water
6.39	Intact Loop Accumulator Begins Injecting Water
14.9	Feedwater Isolation Valves Closed
16.0	End-of-Blowdown Phase
16.0	Accumulator Mass Adjustment for Refill Period
16.0	Transient Modeling Terminated

Table 2.6.3.1-15
DEPS Break Minimum Safeguards Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip, and Loss-of-Offsite Power Are Assumed
1.83	Compensated Pressurizer Pressure Turbine Trip – 1915 psia Reached
3.1	Low-Pressurizer Pressure SI Setpoint – 1715 psia Reached – Feedwater Isolation Signal
6.46	Broken-Loop Accumulator Begins Injecting Water
6.55	Intact-Loop Accumulator Begins Injecting Water
13.4	End-of-Blowdown Phase
13.4	Accumulator Mass Adjustment for Refill Period
15.1	Feedwater Isolation Valves Closed
35.1	Pumped SI Begins after 32-Second Diesel Delay
43.4	Broken-Loop Accumulator Water Injection Ends
44.9	Intact-Loop Accumulator Water Injection Ends
222.1	End-of-Reflood Phase
1117.1	M&E Release Assumption: Broken-Loop Steam Generator Equilibration at Containment Design Pressure of 74.7 psia
1227.2	M&E Release Assumption: Broken-Loop Steam Generator Equilibration at Containment pressure of 64.7 psia
1228.0	M&E Release Assumption: Intact Loop Steam Generator Equilibration at Containment Design Pressure of 74.7 psia
1333.0	M&E Release Assumption: Intact Loop Steam Generator Equilibration at Containment pressure of 54.7 psia
2652.0	Switchover to Cold Leg Recirculation
3600.0	Time of Full Depressurization (i.e., to 14.7 psia)

2.6.3.2 Mass and Energy Release Analysis for Secondary System Pipe Ruptures

2.6.3.2.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) review covered the energy sources that are available for release to the containment, the mass and energy release rate calculations, and the single-failure analyses performed for steam and feedwater line isolation provisions, which would limit the flow of steam or feedwater to the assumed pipe rupture. The NRC's acceptance criteria for mass and energy release analysis for secondary system pipe ruptures are based on:

- GDC-50, insofar as it requires that the margin in the design of the containment structure reflects consideration of the effects of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the values of input parameters.

Specific review criteria are contained in the SRP, Section 6.2.1.4 and other guidance provided in Matrix 6 of RS-001.

Ginna Current Licensing Basis

As noted in *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 3.1, the GDC used during the licensing of Ginna Station predates those provided in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the GDC is discussed in the Ginna UFSAR, Sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the Systematic Evaluation Program review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of Ginna Station design relative to:

- GDC-50 is described in Ginna UFSAR section 3.1.2.5.1, General Design Criterion 50 – Containment Design Basis which states that the containment structure was designed with margin to accommodate the temperature and pressure conditions associated with the loss-of-coolant accident and main steam line break, without loss of function. The evolution of the containment analysis licensing basis is discussed in LR section 2.6.1.1.

2.6.3.2.2 Technical Evaluation

The steamline break mass and energy release inside containment event was analyzed for the EPU program operation. The analysis considered the EPU, and a plant modification to add automatic actuators to the feedwater isolation valves (FIVs). The modification would allow use of the FIVs in lieu of the main feedwater pump discharge valves to provide isolation capability to the steam generators in the event of a steam line break, assuming failure of a feedwater regulator valve (FRV). The FIV plant modification was shown to be a benefit to the containment pressure response, lowering the peak containment pressure by 8.1 psi for the postulated failure of an FRV. The steamline break analysis is described in detail in the License Amendment Request already submitted regarding using the FIVs in lieu of the main feedwater pump discharge valves to isolate steam generators (Reference: Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), *License Amendment Request Regarding Main Feedwater Isolation Valves*, dated April 29, 2005, as supplemented by Reference: Letter from Mary G. Korsnick (Ginna) to Document Control Desk (NRC), *Transmittal of Revised Analysis Associated with the License Amendment Request Regarding Main Feedwater Isolation Valves*, dated July 1, 2005).

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The analysis performed to assess the containment response to the limiting MSLB resulting from operation at EPU conditions does not add any new components or introduce any new functions for existing components that would change the license renewal system evaluation boundaries, except for the FIVs. The new actuators will be added to the scope for license renewal, in accordance with 10CFR50.47(b). The analytical results associated with operating at EPU conditions do not add any new or previously unevaluated materials to the plant systems. System component internal and external environments remain within the parameters previously evaluated. A review of internal and industry operating experience has not identified the need to modify the basis for Aging Management Programs to account for the effects of EPU. Thus no new aging effects requiring management are identified.

2.6.3.2.3 Results

With the FIV plant modification, the limiting case definition changed to a double-ended rupture initiated at 70% power with a vital bus failure. The peak containment pressure is 59.6 psig, which is acceptable because it is below the containment design pressure of 60.0 psig. This represents a gain in margin of 0.2 psi compared to the current analysis of record documented in the UFSAR.

2.6.3.2.4 Conclusion

Ginna has reviewed the mass and energy release assessment for the postulated secondary system pipe ruptures and finds that the analysis adequately addresses the effects of the proposed EPU. Based on this, Ginna concludes that the analysis meets the Ginna Station current licensing basis requirements with respect to GDC-50 for ensuring that the analysis is conservative (i.e., that the analysis includes sufficient margin). Therefore, the Ginna finds the proposed EPU acceptable with respect to mass and energy release for postulated secondary system pipe ruptures.

2.6.4 Combustible Gas Control in Containment

2.6.4.1 Regulatory Evaluation

Following a loss-of-coolant accident (LOCA), hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excess hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered

- The production and accumulation of combustible gases,
- The capability to prevent high concentrations of combustible gases in local areas,
- The capability to monitor combustible gas concentrations, and
- The capability to reduce combustible gas concentrations.

The Ginna staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated.

The NRC's acceptance criteria for combustible gas control in containment are based on

- 10CFR50.44, insofar as it requires that certain plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere,
- GDC-5, insofar as it requires that SSCs important-to-safety not be shared among nuclear power plants unless it can be shown that sharing will not significantly impair their ability to perform their safety functions,
- GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained,
- GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit periodic inspections, and
- GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing.

Specific review criteria are contained in NRC SRP section 6.2.5.

Ginna Current Licensing Basis

Ginna Nuclear Power Plant, LLC (Ginna) has installed dual hydrogen recombiners the function of which was to limit hydrogen concentrations in the containment vessel following a loss-of-coolant accident (LOCA). The Ginna UFSAR currently includes various sections which discuss post-LOCA hydrogen control and the Hydrogen Recombiner System design and operation.

On September 16, 2003, the NRC amended 10CFR50.44 to eliminate certain requirements for hydrogen recombiners and hydrogen purge systems and relaxed the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with risk significance. In order for Ginna Station to adopt the provisions of the amended rule, a license amendment request was submitted to the NRC for approval of changes to Ginna Technical Specifications 3.3.3 and 3.6.7 on August 6, 2004 (reference 1) which was later supplemented on March 14, 2005 (reference 2). The amendment request was prepared in accordance with the NRC-approved Technical Specification Task Force (TSTF) Traveler 447, Revision 1, as a consolidated line item improvement process change. On May 5, 2005, the NRC approved the requested changes to the Ginna Station Technical Specifications which are associated with the 10CFR50.44 rule change (reference 3).

The NRC-approved Technical Specification changes eliminated the need for hydrogen recombiners at the Ginna Station and conformance to GDC-41, GDC-42 and GDC-43 with respect to the containment combustible gas control system. GDC-5 was not applicable because Ginna is a single unit installation.

Based on the NRC-approved changes and the low safety significance of post-LOCA combustible gas generation in large, dry pressurized water reactor containment buildings, such as Ginna Station, the existing UFSAR information will be classified as historical and thus not updated for EPU purposes. However, the capability to monitor post-accident hydrogen concentration in containment is retained, consistent with the requirement of 10CFR50.44(b)(4)(ii), but the components necessary to monitor hydrogen no longer need to be classified as safety-related as previously recommended by Regulatory Guide 1.97.

2.6.4.2 Technical Evaluation

None Required.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The Ginna Station containment combustible gas control system was evaluated for plant license renewal. This evaluation and conclusions are presented in NRC License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, NUREG-1786, dated May 2004.

Portions of the containment combustible gas control system were determined to be within the scope of license renewal as discussed in NUREG-1786, section 2.3.2.4. For

those components within the scope of license renewal, the programs to manage the effects of aging were identified and evaluated in NUREG-1786 section 3.2.2.4.4.

The NRC license renewal evaluations were performed when components of the containment combustible gas control system were included as engineered safety features (ESFs). The license amendment discussed above, which eliminated the system as an ESF system and classification as safety-related, was approved by the NRC in May 2005. Therefore, these system components are being removed from the scope of license renewal, and evaluation for the impact of the proposed EPU on license renewal evaluations is not considered necessary.

2.6.4.3 Results

Not Applicable.

2.6.4.4 References

1. Letter from Mary G. Korsnick (Ginna, LLC) to Robert L. Clark (NRC), "Application for Technical Specification Improvement to Eliminate Requirements for Hydrogen Recombiners and Hydrogen Monitors Using Consolidated Line Item Improvement Process," dated August 6, 2004.

2. Letter from Mary G. Korsnick (Ginna, LLC) to Donna M. Skay (NRC), "Replacement Pages Associated with the Application for Technical Specification Improvement to Eliminate Requirements for Hydrogen Recombiners and Hydrogen Monitors Using the Consolidated Line Item Improvement Process," dated March 14, 2005.

3. Letter from Donna M. Skay (NRC) to Mary G. Korsnick (Ginna, LLC), "Amendment Eliminating Requirements for Hydrogen Recombiners and Hydrogen Monitors Using the Consolidated Line Item Improvement Process (TAC No. MC4195), dated May 5, 2005.

4. Federal Register, Volume 68, Page 54123, September 16, 2003.

2.6.4.5 Conclusion

The Ginna staff concludes that, based on the license amendment approved by the NRC on May 5, 2005, the containment combustible gas control system and its components are no longer classified as ESF or safety-related. The Ginna staff further concludes that post-LOCA hydrogen generation at the proposed EPU conditions need not be further evaluated. The Ginna staff concludes that reliable equipment is provided to continuously monitor post-accident hydrogen concentration, consistent with the requirements of 10CFR50.44. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

2.6.5.1 Regulatory Evaluation

The Containment Recirculation Fan Cooler (CRFC) system, spray system, and residual heat removal (RHR)system remove heat from the containment atmosphere and from the water in the containment sump. The Ginna Nuclear Power Plant, LLC (Ginna) staff review in this area focused on the effects of the Extended Power Uprate (EPU) on the analyses of the available net positive suction head (NPSH) to the containment RHR pumps, and the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers (HXs).

The NRC's acceptance criteria for containment heat removal are based on:

- GDC-38, insofar as it requires that the containment heat removal system to be capable of rapidly reducing the containment pressure and temperature following a loss-of-coolant accident (LOCA), maintaining them at acceptably low levels.

Specific review criteria are contained in the SRP section 6.2.2 as provided in Matrix 6 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna UFSAR, section 3.1, the general design criteria used during the licensing of the Ginna Station predate those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station are published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of the Ginna Station Containment Heat Removal System design relative to conformance to:

GDC-38 is described in UFSAR section 3.1.4.2.9, General Design Criterion 38 – Containment Heat Removal. As described in this UFSAR section, two systems based on different principles are provided to remove heat from containment following postulated accidents to maintain pressure below the containment design pressure. These systems are designed with suitable redundancy and electric power separation to assure performance of its intended function considering a single active failure. These systems are discussed in UFSAR section 6.2.2. The containment spray system and the containment recirculation fan cooling (CRFC) system,

operating in combination, have been shown to be capable of removing sufficient heat from the containment atmosphere following an accident condition to maintain the containment pressure below the containment design limit. The functional performance assumptions for the containment heat removal systems are inputs to the containment accident analyses. The evolution of the containment analysis licensing basis is discussed in LR section 2.6.1.1.

Containment sump NPSH calculations assume the sump water to be saturated, relative to containment conditions (i.e., no credit is taken for containment overpressure). Sufficient NPSH margin exists to support the required flow rates for design basis accidents while removing the necessary containment and decay heat.

In addition to the evaluations described in the UFSAR, the Ginna Station Containment Heat Removal Systems were evaluated for plant License Renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, NUREG-1786, dated May 2004.

With respect to the above SER, the components of the containment heat removal systems that are within the scope of license renewal are identified in SER section 2.3.2.2, Containment Spray, and section 2.3.3.9, Containment Ventilation.

2.6.5.2 Technical Evaluation

This section discusses the containment heat removal systems modeled in the containment integrity analysis for a postulated LOCA event (LR section 2.6.1) in support of EPU program operation. The steamline break mass and energy (M&E) release inside containment event was also analyzed for the EPU program. LR section 2.6.3.2, "Mass and Energy Analysis for Secondary System Pipe Ruptures," discusses the analysis.

2.6.5.2.1 Introduction

The containment heat removal systems are described in Ginna UFSAR, section 6.2.2. Two means of removing heat from the containment atmosphere are provided: the CRFC units and the containment spray system. As shown in UFSAR, Sections 6.2.1.2.3 through 6.2.1.2.3.3 for containment integrity, at least one train of each of these systems is required to provide sufficient steam-condensing capacity to ensure against containment overstress, and to remove that portion of the residual heat and chemical reaction heat released to the containment atmosphere.

The service water (SW) system is the heat sink for the CRFCs. Operation of the minimum required SW pumps will provide sufficient cooling water for the minimum required number of CRFC units.

The EPU increases the heat available to be released into containment, and thus, subsequent heat loads on the containment heat removal systems.

The purpose of the containment integrity LOCA analyses is to demonstrate that the containment, containment structures, and containment cooling safeguards systems are adequate to mitigate the consequences of a hypothetical rupture of a large-break reactor coolant system (RCS) pipe. The impact of LOCA M&E releases on the containment pressure and temperature are addressed to ensure that the containment pressure and temperature remain below their design limits. The containment heat removal systems must also be capable of limiting the post-accident containment conditions such that Environmental Qualification (EQ) acceptance limits at the EPU program conditions are met.

The containment heat removal systems consist of the CRFCs and the containment spray system. These systems perform the following:

- Remove heat from the containment atmosphere.
- Limit offsite radiation by reducing the pressure differential between containment and the external environment (including scrubbing the containment atmosphere) following a LOCA so that offsite doses and control room doses are within the guidelines of 10CFR100 and GDC 19 (see LR section 2.7, "Habitability, Filtration, and Ventilation" for further discussion).

The CRFCs are post-accident safety equipment required to remove heat from the containment atmosphere to control containment temperature and prevent over-pressurization of the containment building. Heat is removed from the containment atmosphere via heat transfer through the SW flowing through the fan coolers. Heat transfer rates through the CRFCs are based upon containment and the SW temperature and SW flow rate. The CRFC heat removal is also dependent on air flow rates.

The CRFC system consists of four containment fan cooler units. The four fan cooler units are arranged as two redundant safety trains, each train containing two fan cooler units. The containment fan coolers are modeled in GOTHIC as a cooler component. Consistent with the application of single-failure criterion presented in LR section 2.6.3.1.2.2.2, "Application of Single-Failure Criterion," an inherent assumption is that offsite power is lost with the pipe rupture. This results in the actuation of the emergency diesel generators (EDGs), powering the two trains of safeguards equipment. Operation of the safeguards equipment required to mitigate the transient is delayed until the EDG has started and accepting loads.

The containment spray system is an active post-accident safety system required to remove heat from the containment atmosphere to control containment temperature and prevent over-pressurization of the containment building, and to remove containment iodine (see LR section 2.7, Habitability, Filtration, and Ventilation). The containment spray system consists of two

redundant trains. During the initial operation of the system (i.e., the injection-phase mode), the containment spray pumps take suction from the refueling water storage tank (RWST) and sprays borated water into containment. Heat is removed from the containment atmosphere via heat transfer to the spray droplets.

Relative to the single failure criterion with respect to a LOCA event, one spray pump is considered inoperable, whether due to an EDG failure (minimum safeguards case) or as the limiting single failure in a maximum safeguards case.

The primary function of the RHR system in the containment evaluation model is remove heat energy from the core and Reactor Coolant System (RCS). The system can also be used with the Safety Injection System (SIS). The RHR system is modeled in the long term LOCA containment recirculation phase of the transient. The RHR system can operate in conjunction with the containment spray system and the CRFC for long term residual heat removal due to core decay heat generation and cooldown of the RCS. (See LR section 2.6.3.1.2.1.2, "Input Parameters, Assumptions, and Acceptance Criteria" and section 2.6.1.2.3, "Boundary Conditions" relative to decay heat modeling). The RHR system is discussed in LR section 2.8.4.4, "Residual Heat Removal System."

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC. GOTHIC version 7.2 was used for the LOCA containment response analysis.

2.6.5.2.2 Input Parameters, Assumptions, and Acceptance Criteria

2.6.5.2.2.1 Input Parameters and Assumptions

Design Basis Accident

The accident modeled in the containment integrity analysis assumes a loss of offsite power coincident with a double ended rupture of the RCS piping between the steam generator outlet and the reactor coolant pump (RCP) inlet.

Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the M&E release rates for the RCP double-ended pump suction (DEPS) break. An inherent assumption in the generation of the M&E release is that offsite power is lost. This results in the actuation of the EDGs, required to power the safety-grade equipment.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an EDG (a complete train of safeguards equipment). This results in the loss of one safety injection pump, one RHR pump, one component cooling water (CCW) pump, two of four CRFCs, and one of two containment spray pumps.

Table 2.6.1-1 provides the key parameters used in the containment integrity analysis relative to the CRFC and containment spray systems. Table 2.6.1-2 provides the CRFCs heat removal capability.

The containment RHR system components credited for the limiting transient scenario are: one (1) RHR pump and heat exchanger and one (1) CCW pump.

2.6.5.2.2.2 Acceptance Criteria

As specified 10CFR50, Appendix A, GDC-38, a system to remove heat from the reactor containment should be provided. The system safety function is to reduce rapidly, consistent with the functioning of other associated system, the containment pressure and temperature following any LOCA, and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities must be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

In order to satisfy the requirement of GDC-38, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the containment leakage criteria for doses at 24 hours.)

Note that Ginna is not an SRP plant. For completeness, however, the SRP long-term cooling criterion is also examined.

The containment analysis acceptance criteria for Ginna are the design pressure of 60 psig and design temperature of 286°F (UFSAR section 6.2.1.2.2.6). Also the Ginna UFSAR containment

temperature profile for Equipment Qualification, Figure 6.1-1, shows that the containment temperature at 24 hours is 152°F.

2.6.5.2.2.3 Description of Analyses and Evaluations

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards based on the postulated single failure of an EDG. This assumption results in the loss of one train of safeguards equipment. Thus, the remaining equipment is conservatively modeled as two (2) CRFCs and one (1) containment spray pump. The other case assumes maximum safeguards, which assumes both EDGs are available. With the maximum safeguards case, the limiting single-failure assumption postulated is the failure of one (1) containment spray pump. The analysis of the cases described provides confidence that the effect of credible single failures is bounded. Operation of the safeguards equipment required to mitigate the transient is delayed until the EDG has started and is accepting loads.

The CRFCs in the containment evaluation model are modeled to actuate on the containment high-pressure setpoint with a biased high uncertainty (6 psig) and begin removing heat from containment after a specified 44-second delay. The post-accident heat removal following a postulated LOCA is based on minimum heat removal capability duty curves. Table 2.6.1-2 provides the CRFC heat removal values applicable to the limiting LOCA M&E release containment response transient.

The containment spray is modeled to actuate on the containment high-high pressure setpoint with a biased high uncertainty to provide margin for instrument uncertainty (33.5 psig) and begin injecting 104°F water after a specified 28.5 second delay. The containment spray flow is 1300 gpm per spray pump in the injection phase. The spray flowrate is modeled in GOTHIC as a function of time. For the Ginna design basis LOCA M&E release containment response transient, the containment spray is credited only during the injection-phase of the transient and is terminated during the transition to cold-leg recirculation (i.e., at 2652 seconds).

The spray drop diameter is modeled as injected in the form of drops with an assumed diameter of 1000 microns (0.0394 inches).

The heat removing capabilities of two CRFCs and one containment spray are considered in the containment integrity analysis, i.e., containment peak pressure and temperature response. Conservative CRFC performance parameters are used as inputs to the containment analyses. The current containment spray system flowrates and water temperatures are used as inputs for the containment analyses. These parameters are incorporated into the containment analyses to demonstrate acceptable performance.

The RHR system in the containment evaluation model is modeled in the long term LOCA containment recirculation phase of the transient. The RHR system operates in conjunction with the containment spray system and the CRFCs for long term residual heat removal due to core

decay heat generation and RCS cooldown. For the Ginna EPU the containment spray pumps are terminated at cold leg recirculation switchover. (See section 2.6.3.1.2.1.2, "Input Parameters, Assumptions, and Acceptance Criteria" and section 2.6.1.2.3, "Boundary Conditions relative to decay heat modeling"). In the containment integrity analysis the RHR system is credited at the onset of cold leg recirculation (i.e., at 2652 seconds from start of the event; activated after a RWST low level signal). The following considerations were applied for the containment integrity analyses:

- Tube plugging is addressed for both the RHR and CCW heat exchangers.
- The GOTHIC heat exchanger models were benchmarked together as a coupled system
- The long-term boil-off calculation used the American Nuclear Society (ANS) Standard 5.1 decay heat model (+2σ uncertainty) for the determination of long-term boil-off from core.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Components of the Containment Heat Removal Systems that are within the scope of License Renewal are described in NUREG-1786, sections 2.3.2.2 and 2.3.3.10. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, sections 3.2.2.4.2 and 3.3. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the Containment Heat Removal Systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.6.5.3 Results

Relative to the safety function of the CRFC and containment spray systems, Table 2.6.1-8 of LR section 2.6.1, "Primary Containment Functional Design," shows continued compliance with the acceptance limits.

The containment analysis acceptance criteria are defined as: the design pressure and design temperature for Ginna, UFSAR section 6.2.1.2.2.6, are 60 psig and 286°F, respectively. Also, the Ginna UFSAR containment temperature profile for Equipment Qualification, Figure 6.1-1, shows that the containment temperature at 24 hours is 152°F.

The limiting long term LOCA containment response transient is from a DEPS break. An analysis of the effects of the single failure criteria has been performed on the M&E release rates for the RCP

suction (DEPS) break. An inherent assumption in the generation of the M&E release is that offsite power is lost. This results in the actuation of the EDGs which are required to power the safety-grade equipment (i.e. for containment heat removal). Table 2.6.5-1 presents the containment temperature results as compared to Ginna UFSAR Figure 6.1.1, "Design-Basis Accident, Containment Temperature Profile."

As shown in Table 2.6.5-1, the peak temperature of 159.4°F @ 24 hours exceeds the acceptance limit of 152°F for a short time. The impact of the 24 hour peak temperature exceeding the acceptance criteria on equipment qualification is evaluated in LR section 2.3.1, "Equipment Qualification of Electrical Equipment."

In terms of pump NPSH, the uprate analytical results are bounded by the peak qualification parameters. No changes to safeguards systems' flow rates are involved. The higher calculated sump water temperature at 24 hours of 159.4°F is subcooled, and has no effect on NPSH margin. Therefore, sufficient NPSH will remain available for the RHR pumps at uprate conditions, with no credit for containment overpressure.

2.6.5.4 Reference

1. Letter from D. M. Crutchfield, NRC to J. E. Maier, RG&E, Subject: SEP Evaluation Report on Topics VI-D and VI-3, dated April 12, 1982.

Table 2.6.5-1 Containment Temperature Profile					
Time	**10 seconds**	**1 hour**	**2.8 hours**	**5.6 hours**	**1 day**
Containment Temperature (°F) **UFSAR** **(Figure 6.1-1**	286	286	250	219	152
Containment Temperature (°F) **EPU Analysis**	273	256.8	221.9	194.6	159.4

2.6.5.5 Conclusion

The Ginna staff has reviewed the containment heat removal systems assessment and concludes that it has adequately addressed the effects of the proposed EPU. The Ginna staff finds that the systems will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-38 for rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Pressure Analysis for Emergency Core Cooling System Performance Capability

2.6.6.1 Regulatory Evaluation

Following a loss-of-coolant accident (LOCA), the emergency core cooling system (ECCS) will supply water to the reactor vessel to reflood the reactor core and thereby cool the core. The core flooding rate will increase with increasing containment pressure. The Ginna Nuclear Power Plant, LLC (Ginna) staff reviewed analyses of the minimum containment pressure that could exist during the period of time following a LOCA until the core is reflooded to confirm the validity of the containment pressure used in ECCS performance capability studies. Ginna's review included assumptions made regarding heat removal systems, structural heat sinks, and other heat removal processes that have the potential to reduce the pressure.

The NRC's acceptance criteria for the pressure analysis for ECCS performance capability are based on:

- 10CFR50.46, insofar as it requires the use of an acceptable emergency core cooling system evaluation model that realistically describes the behavior of the reactor during LOCAs, or an emergency core cooling system evaluation model developed in conformance with 10CFR50, Appendix K.

Specific review criteria are contained in the SRP, Section 6.2.1.5 as provided in Matrix 6 of RS-001, Revision 0.

Ginna Current Licensing Basis

Ginna UFSAR section 15.6.4.2.4.2 describes the methodology used to analyze the large break LOCA. The current ECCS containment backpressure analysis for a large break LOCA was performed using the COCO computer code (reference 1) as sanctioned by the current Large Break LOCA evaluation model (reference 2). As noted in UFSAR section 15.6.4.2.4.2, the containment backpressure used in the WCOBRA/TRAC hydraulic calculations was conservatively low and included the effect of all pressure reducing systems and processes.

2.6.6.2 Technical Evaluation

This section discusses the containment backpressure analysis used in the large break LOCA analysis to support an extended power uprate (EPU).

2.6.6.2.1 Introduction

The system hydraulic transient for a large break LOCA is influenced by the containment pressure transient response to the M&E released from the reactor coolant system (RCS) by the

LOCA. In the best estimate ECCS evaluation model using the automated statistical treatment of uncertainty method (ASTRUM) (reference 3), the containment pressure transient is provided as a boundary condition to the system hydraulic transient. The containment pressure transient applied is to be conservatively low and includes the effect of the operation of all pressure reducing systems and processes. The COCO computer code (reference 1) is used to generate the containment pressure response to the M&E release from the break from a reference WCOBRA/TRAC transient. This containment pressure curve is then used to determine an appropriate input to the WCOBRA/TRAC code as sanctioned by the large break LOCA evaluation model (reference 3).

2.6.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

2.6.6.2.2.1 Input Parameters and Assumptions

Table 2.6.6-1 provides the general parameters used in the ECCS containment backpressure boundary condition analysis. Table 2.6.6-2 provides the containment recirculation fan coolers (CRFCs) heat removal rate used in the ECCS containment backpressure boundary condition analysis. Table 2.6.6-3 provides the structural heat sink data used in the ECCS containment backpressure boundary condition analysis. Ginna and Westinghouse have ongoing processes which ensure that the values and ranges used in the ECCS containment backpressure analysis for a large break LOCA conservatively bound the values and ranges of the plant as-operated for those parameters.

2.6.6.2.2.2 Acceptance Criteria

As specified in 10CFR50, Appendix K: The containment backpressure boundary condition analysis is acceptable if the containment pressure used for evaluating the cooling effectiveness during reflood does not exceed a pressure calculated conservatively for this purpose. The calculation should include the effects of operation of all installed pressure reducing systems and processes.

2.6.6.2.3 Description of Analyses and Evaluations

The Containment Backpressure Analysis for a Large Break LOCA was performed for the proposed EPU using the COCO computer code (reference 1) as sanctioned by the large break LOCA evaluation model (reference 3). The application of this code is consistent with Westinghouse Emergency Core Cooling System Evaluation Model Summary, WCAP-8339 Appendix A (Non-Proprietary), June 1974 (reference 4). This analysis reflects the Ginna specific parameters as discussed in section 2.6.6.2.2. The result of this analysis is discussed in Section 2.6.6.2.4.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The NRC issued its Ginna License Renewal Safety Evaluation Report (SER), NUREG-1786, in May 2004. The ECCS system components whose performance is relied upon to support the inputs, assumptions, and results of the containment backpressure analysis for a large break LOCA are discussed in SER section 2.3.2, "Engineered Safety Features Systems." EPU activities do not add any new components nor do they introduce any new functions for existing ECCS components that would change the license renewal evaluation boundaries. The ECCS performance capability described in this section for the proposed EPU involves analytical techniques and methodology which are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of 10CFR50.46. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to ECCS performance capability.

2.6.6.3 Results

Figure 2.6.6-1 provides a plot of the containment pressure curve used as an input into the WCOBRA/TRAC computer code and the containment pressure curve calculated by the COCO computer code. The containment pressure curve used as an input to the WCOBRA/TRAC code for the thermal-hydraulic calculations is at a lower pressure than the containment pressure curve calculated by the COCO computer code. As such, the containment pressure curve used in the Large Break LOCA analysis is considered acceptable.

Since the RCS has more energy at uprated power conditions when compared to the non-uprated power condition, the energy release associated with a large break LOCA at uprated conditions will be increased. Therefore, the minimum containment pressure response for a large break LOCA at uprated power conditions will be greater than that of the non-uprated conditions.

2.6.6.4 References

1. F. M. Bordelon and E. T. Murphy, Containment Pressure Analysis Code (COCO), WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

2. S. I. Dederer et. al., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection, WCAP-10924-P-A, Volume 2, Revision 2, and Addenda (Proprietary Version), December 1988.

3. M. E. Nissley et. al., Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), WCAP-16009-P-A (Proprietary Version), WCAP-16009-NP-A (Non-Proprietary Version), January 2005.

4. F. M. Bordelon et. al., Westinghouse Emergency Core Cooling System Evaluation Model Summary, WCAP-8339 Appendix A (Non-Proprietary), June 1974.

2.6.6.5 Conclusion

Ginna has reviewed the minimum containment pressure analysis and concludes that the analysis has adequately accounted for plant operation at the EPU power level and were performed using acceptable analytical models. Ginna further concludes that the evaluation has demonstrated that the containment pressure curve used in the large break LOCA analysis is considered acceptable. Based on this, Ginna concludes that the requirements in 10CFR50.46 regarding emergency core cooling system performance will continue to be met following implementation of the proposed EPU. Therefore, Ginna finds the EPU acceptable with respect to minimum containment pressure for emergency core cooling system performance.

Table 2.6.6-1

Parameters for ECCS Containment Backpressure Analysis

Parameter	Value
Containment Physical Description	
• Maximum Net Free Volume (ft^3)	1.066×10^6
Containment Initial Conditions	
• Minimum Operating Pressure (psia)	14.5
• Minimum Operating Temperature (°F)	90
Relevant Temperatures	
• Minimum Refueling Water Storage Tank (RWST) Temperature (°F)	50
• Minimum Service Water Temperature (°F)	30
• Minimum Outside Temperature (°F)	-20
Containment Spray System and SI Spill	
• Maximum Number of Pumps Operating	2
• Maximum Runout Flow Rate (gpm from ALL pumps)	3600
• Minimum Initiation Time (sec)	9
• Maximum Safety Injection Spill Flow Rate (gpm)	465
Safeguards Containment Recirculation Fan Coolers (CRFCs)	
• Maximum Number of Fan Coolers Operating	4
• Minimum Post Accident Initiation Time of Fan Coolers (sec)	0

Table 2.6.6-2

Containment Recirculation Fan Cooler Heat Removal Rate for ECCS Containment Backpressure Analysis

Containment Temperature (°F)	Heat Removal Rate for One CRFC (BTU / sec)
30	0
220	32472
286	50500

Table 2.6.6-3

Structural Heat Sink Data for ECCS Containment Backpressure Analysis

Wall Description	Thickness (in)	Material	Area (ft^2)
• Insulated Portion of Dome and Containment Wall	0.019 1.250 0.375 42.0	Stainless Steel Insulation Carbon Steel Concrete	36285
• Uninsulated Portion of Dome	0.375 30.0	Carbon Steel Concrete	12370
• Basement Floor	24.0 0.250 24.0	Concrete Carbon Steel Concrete	7230
• Walls of Sump A	0.25 36.0	Carbon Steel Concrete	2270
• Floor of Sump A	24.0 0.250 12.0	Concrete Carbon Steel Concrete	280
• Walls of Sump B	24.0 0.25 12.0	Concrete Carbon Steel Concrete	210
• Floor of Sump B	24.0 0.250 12.0	Concrete Carbon Steel Concrete	120

Table 2.6.6-3

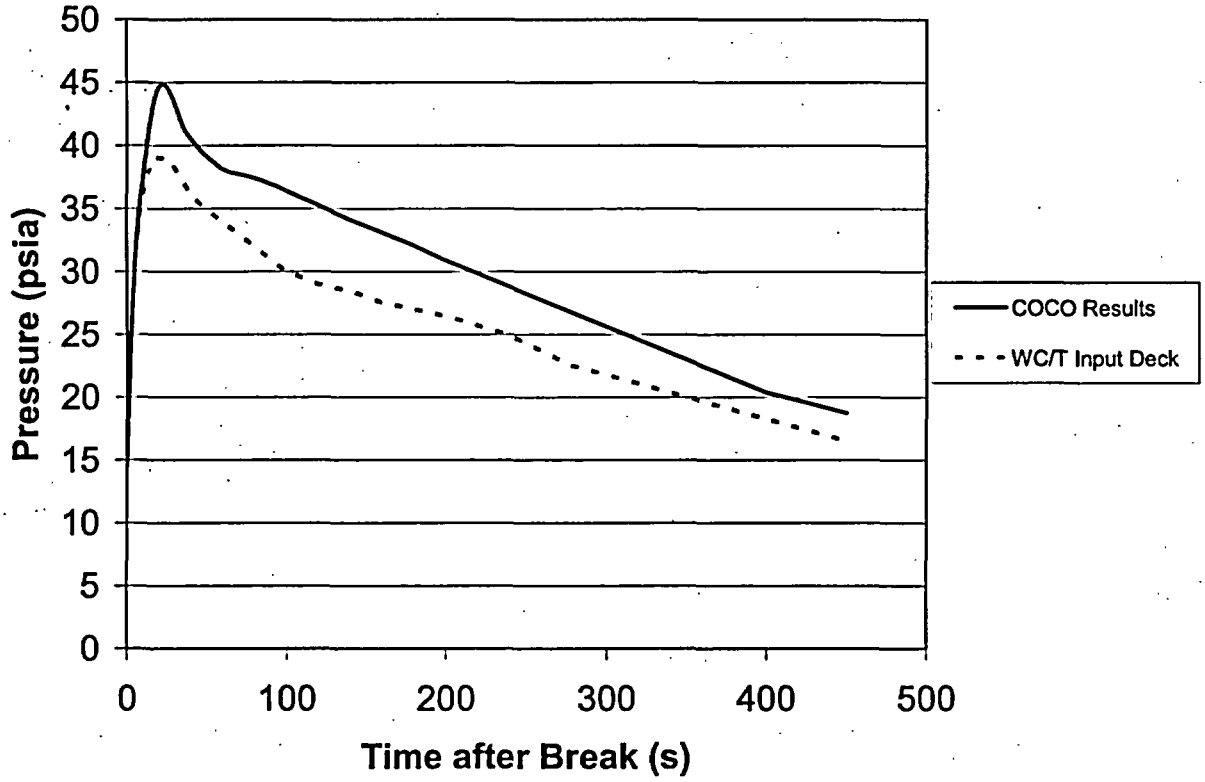
(continued)

Structural Heat Sink Data for ECCS Containment Backpressure Analysis

Wall Description	Thickness (in)	Material	Area (ft^2)
• Inside of Refueling Cavity	0.250 30.0	Stainless Steel Concrete	6170
• Bottom of Refueling Cavity	0.250 24.0	Stainless Steel Concrete	1260
• Area on Outside of Refueling Cavity Walls	0.250 30.0	Stainless Steel Concrete	6750
• Area Inside of Loop Compartment	30.0	Concrete	10370
• Intermediate Level Floor Area	6.0	Concrete	5320
• Operating Floor	24	Concrete	6500
• 1.48 inch thick I-beam	1.48	Carbon Steel	2000
• 0.94 inch thick I-beam	0.94	Carbon Steel	630
• 0.52 inch thick I-beam	0.52	Carbon Steel	4220
• 0.61 inch thick I-beam	0.61	Carbon Steel	1190
• Cylindrical Supports for Steam Generator and Reactor Coolant Pumps	0.5	Carbon Steel	470
• Plant Crane Support Columns	0.75 2.0	Carbon Steel Carbon Steel	4030 380
• Beams Used for Crane Structure	0.79 1.52 1.44	Carbon Steel Carbon Steel Carbon Steel	1220 1910 260
• Structure on Operating Floor	24	Concrete	2060
• Grating, Stairs, Miscellaneous Steels	0.125	Carbon Steel	7000

Figure 2.6.6-1

COCO Calculated Containment Backpressure (using mass and energy releasesfrom a reference WCOBRA/TRAC transient) and WCOBRA/TRAC Input Containment Backpressure Versus Time After Break



2.7.1.1 Fire Protection

2.7.1.1.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The Ginna Nuclear Power Plant, LLC (Ginna) staff review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire.

The NRC's acceptance criteria for the fire protection program are based on:

- 10CFR50.48 and associated Appendix R to 10CFR50, insofar as they require the development of a fire protection program to ensure, among other things, the capability to safely shut down the plant

- GDC-3, insofar as it requires that:

  a) Structures, systems, and components important-to-safety be designed and located to minimize the probability and effect of fires,

  b) Noncombustible and heat resistant materials be used, and

  c) Fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important-to-safety .

- GDC-5, insofar as it requires that SSCs important-to-safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Specific review criteria are contained in Standard Review Plan Section 9.5.1, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety

Assessment Report (IPSAR), completed August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis. Specifically, the adequacy of Ginna Station design relative to conformance to:

- GDC - 3 is described in Ginna UFSAR section 9.5.1, "Fire Protection Systems." This UFSAR section discusses conformance to GDC-3 of 10CFR50, Appendix A; Branch Technical Position (BTP) 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants;" Atomic Industrial Forum (AIF) GDC-3; 10CFR50.48, "Fire Protection;" and 10CFR50, Appendix R, "Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979." The discussion in UFSAR section 9.5.1 provides a consolidation of the information provided in UFSAR section 3.1.1.1.3 for AIF GDC-3 and UFSAR section 3.1.2.1.3 for NRC GDC-3 as presented in 10CFR50, Appendix A. As described in UFSAR section 9.5.1, fire detection and fighting systems of appropriate capacity and capability are provided to minimize the adverse effects of fire on safety-related SSCs . Fire prevention in all areas of the plant is provided by structure and component design, which optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Sensing devices include both ionization chambers (smoke detectors) and temperature detectors. Fire-fighting equipment includes fixed (automatic water suppression) in appropriate areas. In addition, automatically initiated Halon 1301 provides a total flooding system in the relay room and computer room. Appropriate hoses and portable fire-fighting equipment are placed throughout the plant. The fire protection system is designed in accordance with the standards of the National Fire Protection Association and is based generally on the recommendations of the Nuclear Energy Property Insurance Association.

- GDC – 5 is described in Ginna UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components, which states that Ginna is a single unit installation. Therefore, there are no shared SSCs.

As addressed in Ginna UFSAR section 9.5.1.1.1, the design criterion used during the licensing of Ginna Station was General Design Criterion 3, included in the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967. The design of the fire protection system was reviewed in 1972 on the basis of GDC 3 of Appendix A to 10CFR50, which was promulgated after the licensing of Ginna Station. It was determined that the requirements of GDC 3 were appropriately met by the plant design.

As addressed in Ginna UFSAR section 9.5.1.1.2, in May 1976 NRC Branch Technical Position (BTP) 9.5-1 was published for comment as Regulatory Guide 1.120, and in August 1976, Appendix A to BTP 9.5-1 was published for use by plants docketed prior to July 1, 1976. The design of the Ginna Station fire protection system was reviewed against the criteria of BTP 9.5-1 in a submittal to the NRC in February 1977. The submittal included a fire hazards analysis and several proposed design modifications in compliance with the regulatory guidance. A safety evaluation report was issued by the NRC in February 1979, with supplements in December 1980, February 1981, and June 1981.

As addressed in Ginna UFSAR section 9.5.1.4:

- The requirements for protecting safe shutdown systems and their respective components and associated circuits are specified in 10CFR50, Appendix R and NRC Generic Letter 81-12, "Fire Protection Rule." Ginna submitted an evaluation report in January 1984 describing alternative safe shutdown capability in accordance with Appendix R, section III.G. The report was revised in October 1984 and in January 1985. The revisions included a request for specific exemptions from the retrofit requirements of Appendix R, section III.G. In safety evaluation reports (SERs) of February 1985 and March 1985, the NRC accepted the alternative safe shutdown proposals and granted the requested exemptions. Thus, all areas of the plant either meet the retrofit requirements of Appendix R (as exempted), or are provided with acceptable alternative safe shutdown capability. In 1998, Ginna notified the NRC that one of the exemptions granted in the safety evaluation report of March 1985 was no longer required.

- Subsequent to the implementation of the Appendix R modifications during the 1986 refueling outage, the Ginna Station alternative safe shutdown report was revised (March 1986) to incorporate deviations from the original design and compliance methods. The updated safe shutdown report is included in the Ginna Station Fire Protection Program Report.

The Ginna Station Fire Protection Program Report summarizes the licensing basis for the program, as follows: The program has been developed to comply with and is based upon the requirements of GDC 3 of 10CFR50, Appendix A; 10CFR50.48(a); and Ginna's commitment to implement 10CFR50,Appendix R, sections III.G (Fire Protection of Safe Shutdown Capability), III.J (Emergency Lighting), and III.O (Reactor Coolant Pump Oil Collection System), and BTP 9.5-1, Appendix A. The requirements contained in 10CFR50, Appendix R, section III.L (Alternative and Dedicated Shutdown Capability) are applicable to areas where alternate shutdown capability is selected.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station Fire Protection Program was evaluated for plant license renewal. The evaluation is documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

The fire protection program / systems are addressed in sections 2.3.3.6 and 3.3.2.3.2 of the SER. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them.

2.7.1.1.2 Technical Evaluation

2.7.1.1.2.1 Introduction

Fire Protection Program Report

The Ginna Station Fire Protection Program Report consolidates a detailed summary of the Ginna Station regulatory-required Fire Protection Program into a single document, and, as such, embodies the fire protection program. The Report documents Ginna's Fire Protection Plan, Fire Hazards Analysis, and Appendix R Safe Shutdown Analysis (herein referred to as the "Safe Shutdown Analysis"). It also includes a summary of Ginna's conformance to the BTP 9.5-1, Appendix A, a summary of the commitments made in demonstrating conformance to 10CFR50, Appendix R, a summary of Ginna's fire protection review and actions relative to industry operating experience, a review of compliance with selected NFPA codes, and a summary of fire protection engineering evaluations.

The program description / evaluation in this section addresses:

- "Fire Protection," which includes elements of the Fire Protection Program associated with the Fire Protection Plan and the Fire Hazards Analysis.

- "Safe Shutdown Analysis," which includes program elements associated with the Appendix R Safe Shutdown Analysis.

Fire Protection

The Fire Protection Plan describes the controls associated with the Ginna Station Fire Protection Program. The plan describes the features necessary to implement the Fire Protection Program, including administrative controls, personnel requirements for fire prevention and manual fire suppression activities, and fire protection systems and features, including fire detection and automatic and manually operated suppression systems. The administrative controls include controls to minimize the amounts of combustibles to which a safety-related / safe shutdown area may be exposed, control of hot work, impairment monitoring, etc.

The Fire Hazards Analysis includes evaluations of the fire areas / fire zones at Ginna, physical characteristics of required fire barriers, and combustible loading and fire severity in each fire area / fire zone.

Safe Shutdown Analysis

The Safe Shutdown Analysis identifies the analysis methodology used to demonstrate compliance with 10CFR50, Appendix R, sections III.G, III.J, III.O, and portions of III.L. Included is a description of the approved Appendix R exemptions for those plant areas where literal compliance with Appendix R, section III.G was not required.

The following Safe Shutdown Analysis topics are addressed below:

- Safe shutdown systems / components
- Alternative shutdown capability
- Appendix R compliance strategies
- Time critical tasks
- Other supporting analyses / evaluations
- Safe shutdown procedures

Safe Shutdown Systems / Components

The Safe Shutdown Analysis identifies the specific systems credited with achieving safe and/or alternative shutdown (e.g., auxiliary feedwater (AFW) system, residual heat removal (RHR) system). The safe shutdown components, which represent the minimum set of components used to achieve Appendix R safe / alternative shutdown (e.g., main steam atmospheric relief valves), are identified in a plant database.

Alternative Shutdown Capability

For the following Ginna fires areas, compliance with the provisions of 10CFR50, Appendix R, section III.G.2 cannot be achieved, and therefore an alternative shutdown capability is provided for the following plant areas:

- Control complex
- Cable tunnel
- Auxiliary building basement / mezzanine
- Battery rooms 1A / 1B
- Emergency diesel generator 1B area
- Screenhouse

The alternative shutdown methods provide the capability to achieve cold shutdown conditions within 72 hours and maintain them thereafter. The systems and equipment comprising the alternative shutdown methods are capable of being powered by either on-site or off-site electrical power sources.

Alternative shutdown methods include the following:

- Local control of one charging pump to provide makeup / increase in reactor coolant system pressure.
- Local control of turbine-driven AFW pump lube oil pump and turbine.
- Local control of one emergency diesel generator.
- Local connection between standby AFW system and underground yard fire water hydrant using fire hose to provide alternative AFW in event of loss of service water system.
- Local connection between underground yard fire water hydrant and emergency diesel generator using fire hose to provide alternative diesel generator cooling in event of loss of the service water system.

Appendix R Compliance Strategies

The Safe Shutdown Analysis provides a description of the safe shutdown methods for each plant fire area. Although other equipment may be used to bring the plant to a safe shutdown condition following a fire, the Safe Shutdown Analysis shows that at least one train of safe shutdown equipment is available to achieve safe shutdown given a fire in the defined fires areas at Ginna Station.

In the event of a fire in the auxiliary building basement / mezzanine, auxiliary building operating floor / intermediate level, or the screenhouse, the RHR system or its associated support systems may be damaged by fire. If the RHR system is rendered inoperable, water solid steam generator operation can be employed for cooldown to Cold Shutdown conditions. A design analysis shows that, for current plant conditions, the plant can be cooled from 260°F to less than 200°F in 12 hours using water solid steam generator operation.

In the event of a fire in the control complex, alternative shutdown capability is provided by local operation of the turbine-driven AFW pump. A time validation performed for the fire procedure for this area showed that the operators were controlling flow to the steam generators using the turbine-driven pump discharge control valve at 20 minutes after reactor shutdown. Analysis of steam generator dryout time for current plant conditions shows that it would take greater than 50 minutes for both steam generators to boil dry assuming no feedwater addition.

Time-Critical Tasks

The Safe Shutdown Analysis identifies three "time-critical" tasks. A description of these tasks and their basis follows:

- Closure of the RHR pump suction valve from the refueling water storage tank (RWST) after reactor shutdown for an Appendix R fire.

  A design analysis determines the time to drain inventory in the RWST for the scenario in which spurious operation of one of the RHR pump suction valves causes flow into containment. Closure of the RHR pump suction valve from the RWST within 28 minutes after reactor shutdown terminates the drainage of the RWST and assures that makeup to the RWST is not required for an Appendix R fire in which the pressurizer power operated relief valves (PORVs) are not available for primary system depressurization.

- Restoration of charging flow after reactor shutdown for an Appendix R fire.

  A design analysis determines the time for pressurizer level to decrease from the level at hot zero power (35%) to the bottom level tap, due to total reactor coolant system leakage, including reactor coolant pump leakage. Restoration of charging flow within 36 minutes after reactor shutdown assures that pressurizer level will not decrease below the level indicating range.

- Stopping charging flow after reactor shutdown for an Appendix R fire.

  A design analysis documents that the worst case for a loss of charging pumps suction would be a fire event which caused a spurious closure of the volume control tank suction air-operated valve, with the RWST suction air-operated valve remaining closed. The analysis determines that tripping the charging pumps from the control room within 1 minute after reactor shutdown will prevent any pump degradation due to loss of pump suction.

As addressed in the Safe Shutdown Analysis, as part of the development of the current fire procedures, these time critical tasks have been validated in the plant, utilizing plant operators.

Other Supporting Analyses / Evaluations

Supporting analyses / evaluations include the following:

- A design analysis documents the capability of the underground yard (city water) fire water loop to provide an alternate supply to the service water header. The limiting conditions in this analysis include a fire in the screen house which renders the motor and diesel driven fire pumps, along with the service water pumps, inoperable. The analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby auxiliary feedwater pumps, and two standby auxiliary feedwater room coolers.

- An effectiveness review of the reactor coolant pump lube oil collection system concludes that the system is adequately designed to contain the lube oil spills from the various leak sites located on the reactor coolant pump motors.

Safe Shutdown Procedures

Various procedures have been developed to guide the operators during alternative shutdown operations. These include operational and repair procedures to achieve cold shutdown. Operational procedures include alternative shutdown procedures for the fire areas requiring alternative shutdown capability, identified above. Operational procedures also address alternate water supplies to the auxiliary feedwater pumps and alternate cooling for the emergency diesel generators.

Acceptance Criteria

The overall acceptability of the Fire Protection Program at Ginna Station is based upon a defense-in-depth approach to demonstrate that in case of fire, the plant can be safely shutdown and maintained in a safe shutdown condition. The key attributes to this defense-in-depth approach are to provide

- Fire protection by structure and component design and location and by the application of appropriate administrative controls to minimize the probability and effects of fires,

- Fire detection capability by appropriate fire sensing devices, and

- Fire fighting and suppression capability through personnel training and provision of suitable portable and fixed fire fighting equipment of appropriate capacity and capability.

2.7.1.1.2.2 Description of Analyses and Evaluations

Fire Protection

Apart from plant modifications, the EPU does not affect the following elements of the fire protection program:

- Addition of new combustible material.
- Fire barriers, penetrations, doors, or the plant radio system.
- Ventilation air flow patterns.
- Plant fire programs or the Fire Protection Program Report.
- Fire wrap and fire coatings on structural steel.
- Fire protection suppression or fire detection system components.
- Safety-related components within an area protected by the fire suppression system.

Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Fire Protection Program requirements.

The EPU does not affect the elements of the fire protection program related to administrative controls and fire protection responsibilities of plant personnel. Evaluation of the impact of the EPU on the probability of increased radiological release resulting from a fire is addressed in LR section 2.13, Risk Evaluation.

Safe Shutdown Analysis

Safe Shutdown Systems / Components

The EPU does not affect the minimum set of pre EPU safe shutdown systems / components, including cables, credited with achieving safe and/or alternative shutdown. However, additional equipment is added to the list to account for the effects of increased decay heat. Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

Alternative Shutdown Capability

The EPU does not affect the alternative shutdown methods, discussed in LR section 2.5.1.4.2.1. The EPU does not modify the function of any mechanical component in the alternative safe shutdown flow paths or introduce any plant equipment failure modes which will impact the ability to achieve any of the alternative shutdown functions. The EPU does not adversely affect any components or circuits that provide power, control, or indication to components required for alternative safe shutdown. Enhancements will be made to selected components or control circuits to improve alternative shut down capability. As stated above, plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

Analyses were performed to demonstrate that the plant can be cooled down from normal operating temperature to Cold Shutdown at EPU conditions for the following cases: (1)

residual heat removal system (and its supporting systems) is available, and (2) RHR system is not available. For the case where the RHR system is not available, water solid steam generator operation is used for the final phase of cooldown to Cold Shutdown.

- Case 1: RHR system is available:
  - An analysis shows that the plant can be cooled from normal operating temperature to residual heat removal system initiation conditions with one steam generator within 60 hours after reactor shutdown.
  - An analysis shows that the plant can be cooled from RHR system initiation conditions to Cold Shutdown within 72 hours after reactor shutdown, assuming single train cooldown and a cooldown start time of 60 hours after reactor shutdown.
- Case 2: RHR system is not available:
  - An analysis provides time vs. temperature data for plant cooldown, using two steam generators, from normal operating temperature to conditions for initiation of water solid steam generator operation. For this analysis, at least one pressurizer PORV is available for depressurization of the primary system during the cooldown.

    For the three fire areas which may require water solid steam generator operation (Auxiliary Building basement / mezzanine, Auxiliary Building operating floor / intermediate level, and Screenhouse), in the Safe shutdown Analysis, only one train of equipment (e.g., one steam generator atmospheric relief valve) is currently credited for cooldown to the temperature at which water solid steam generator operation is initiated. Since two steam generators are needed in order to meet the Appendix R 72-hour cooldown requirement for this case at EPU conditions, actions will be taken, including update of the Safe Shutdown Analysis and applicable fire procedures, to address this change. Note that this scenario does not require that an additional single failure be postulated.
  - An analysis shows that the plant can be cooled from conditions for initiation of water solid steam generator operation to Cold Shutdown within 72 hours after shutdown, assuming initiation of water solid steam generator operation 50 hours after reactor shutdown. To facilitate drainage of water from the steam generators during water solid operation, a modification will be implemented to install drain piping which will direct drainage from each main steam header to the steam generator blowdown tank via the blowdown tank steam header.

Based on the above analyses, the requirement to achieve cold shutdown conditions within 72 hours after reactor shutdown continues to be met for EPU conditions.

Appendix R Compliance Strategies

As discussed in Section 2.5.1.4.2.1 above, in the event of a fire in the control complex, local operation of the turbine-driven auxiliary feedwater pump is used for decay heat removal, and a time validation showed that the operators were controlling auxiliary feedwater flow to the steam generators at 20 minutes after reactor shutdown. Analysis of steam generator dryout time for EPU plant conditions shows that it would take 35 minutes for both steam generators to boil dry assuming no feedwater addition . Thus, there continues to be adequate time for the operator to supply feedwater to the steam generators at EPU conditions. Actions will be taken to enhance local control of auxiliary feedwater flow to the steam generators.

The Safe Shutdown Analysis shows that at least one train of safe shutdown equipment is available to achieve safe shutdown following a fire in the defined fire areas at Ginna Station for current plant conditions. As stated above, modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Appendix R compliance methods.

For a fire in the auxiliary building basement / mezzanine, the current Shutdown Analysis states that (1) depressurization of the RCS by ambient heat losses from the pressurizer without use of a PORV may be required, and (2) water solid steam generator cooldown to provide cold shutdown capability may be required For current plant conditions, an RCS cooldown rate based on a pressurizer cooldown rate of 3.75°F per hour is used for depressurization of the RCS in the temperature range of 550°F to 350°F. Based on results of the EPU analysis for cooldown using two ARVs and water solid steam generator cooldown analysis, a pressurizer cooldown rate of greater than 3.75°F per hour will be required at EPU conditions. Therefore, to ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available.

Time-Critical Tasks

The impact of the EPU on the three "time-critical" tasks described in LR section 2.5.1.4.2.1 follows:

- Closure of the RHR pump suction valve from the RWST after reactor shutdown for an Appendix R fire.

  The analysis for this task uses as input the limiting case for total reactor coolant system leakage for a 72 hour Appendix R event. The cooldown scenario applicable to this limiting case at current plant conditions includes RCS cooldown based on a pressurizer cooldown rate of 3.75°F per hour by ambient heat loss (PORVs not available), and water solid SG cooldown at 5°F per hour. As shown in the analysis, the RCS cooldown rate is essentially the same as the pressurizer cooldown rate in the temperature range where the RCS cooldown follows RCS depressurization by ambient heat loss from the pressurizer.

As indicated under "Appendix R Compliance Strategies" above, to ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available. Since the pressurizer cooldown rate at EPU conditions will exceed 3.75°F per hour, the limiting RCS leakage determined in the current analysis will remain bounding for EPU conditions. Therefore, the calculated time for closure of the RHR pump suction valve after an Appendix R fire (28 minutes) at current conditions, which prevents drain-down of the RWST to a level that requires makeup when PORVs are not available, remains bounding for EPU conditions.

- Restoration of charging flow after reactor shutdown for an Appendix R fire.

  An analysis determines that the time for pressurizer level to decrease from the level at hot zero power (35%) to the bottom level tap, due to total reactor coolant system leakage at current plant conditions, is 36 minutes. At EPU conditions, the pressurizer level low limit at hot zero power will be changed to 20% of span. The time required for pressurizer level to decrease from the low limit at EPU conditions to the bottom level tap is determined to be 23.9 minutes. Therefore, charging flow will need to be restored within 23 minutes after reactor shutdown to assure that pressurizer level will not decrease below the indicating range at EPU conditions. Modifications in support of restoring charging flow within 23 minutes are as follows:

  - Relocation of the "A" charging pump control power transfer switch from bus 14 to the charging pump room. This will reduce the time required to transfer control power for local operation of the charging pump.

  - Installation of a bank of high pressure air cylinders with a permanent connection to the charging pump speed control system normal air supply. This will ensure a backup air supply in the event of loss of normal air supply to the speed control system.

- Stopping charging flow after reactor shutdown for an Appendix R fire.

  The EPU does not affect the analysis for this task. Therefore, this time critical task remains unchanged for EPU conditions.

Other Supporting Analyses / Evaluations

The impact of the EPU on the supporting analyses / evaluations described in LR section 2.5.1.4.2.1 follows:

- An analysis shows that the city yard fire loop has the capability to simultaneously supply water to:

    - One emergency diesel generator
    - Two standby AFW pumps
    - Two standby AFW room coolers

The emergency diesel generator flow rate is based on the minimum service water flow rate to the diesel generator coolers during the LOCA injection phase, plus margin. As addressed in LR section 2.3.3, "AC Onsite Power System," the EPU does not affect the maximum steady state loading on the emergency diesel generators during the LOCA injection phase . Therefore, the emergency diesel generator flow rate for this case, 320 gpm, is not affected by the EPU.

Note: Since some equipment used for the LOCA injection phase is not required for Appendix R cooldown (e.g., containment fans, a containment spray pump), the minimum service water flow rate to the diesel generator coolers during the LOCA injection phase would envelope the minimum flow rate needed for Appendix R cooldown.

The standby AFW pump flow rate (225 gpm per pump) is the flow rate used in the water solid steam generator cooldown analysis for current plant conditions. In the plant cooldown analysis using water solid steam generator operation for EPU conditions, a standby AFW pump flow rate of 250 gpm per pump is used. A design analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby AFW pumps (250 gpm to each steam generator), and two standby AFW room coolers at EPU conditions.

The EPU does not affect the design flow rate of cooling water to the standby auxiliary feedwater room coolers.

- The EPU does not affect the reactor coolant pump lube oil collection system, and therefore the EPU does not affect the conclusions of the system effectiveness review.

Safe Shutdown Procedures

As addressed above, for the three fire areas which may require water solid steam generator operation (Auxiliary Building basement / mezzanine level, Auxiliary Building operating floor / intermediate level, and Screenhouse), actions required to implement the use of two steam generators in order to meet the Appendix R 72-hour cooldown requirement at EPU conditions include the need to update applicable fire procedures.

Implementation of a contingency activity to use auxiliary spray to reduce RCS pressure for a fire in the Auxiliary Building basement / mezzanine will require development of a new procedure and revision of the alternate shutdown fire procedure for a fire in the Auxiliary Building basement / mezzanine.

In order to accommodate changes in times to perform specific actions at EPU conditions, procedural activities may need to be re-ordered to ensure actions will be performed in the required time. As addressed above, at EPU conditions charging flow will be restored within 23 minutes after reactor shutdown to assure that pressurizer level will not decrease below the indicating range (change from 36 minutes at current conditions). Applicable fire procedures will be updated as required.

Evaluation of Impact of Renewed Plant Operating License Evaluations and License Renewal Programs

The fire protection program attributes and system components that are within the scope of license renewal are addressed in License Renewal SER section 2.3.3.6. SER section 3.3.2.3.2 addresses aging management of the fire water system and associated components. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them.

The License Renewal SER notes that Ginna stated that the Fire Protection Program is consistent with, but includes exceptions to NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." The SER addresses exceptions taken by Ginna to the GALL Report in the following areas:

- Testing frequency of the halon system
- Visual inspection of fire doors
- Periodic flow testing of infrequently used fire water system loops
- Visual inspection frequency of yard fire hydrants
- Frequency of fire hydrant flow tests

The NRC concluded that for those portions of the program which Ginna stated were consistent with the GALL program are consistent with GALL, and that, with regard to exceptions taken to the GALL program, Ginna had demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained.

Based on the uprate evaluation of elements of the fire protection program in this section, the EPU does not affect the evaluation / conclusions in the License Renewal SER regarding the fire protection program, and no new aging effects requiring management are identified.

2.7.1.1.3 Results

Apart from plant modifications, the EPU does not affect the fire protection program. Plant modifications required in support of the EPU are reviewed to ensure any design changes do not adversely impact existing Fire Protection Program requirements.

The EPU does not affect the elements of the fire protection program related to administrative controls and fire protection responsibilities of plant personnel.

The EPU does not affect the minimum set of pre EPU safe shutdown systems / components, including cables, credited with achieving safe and/or alternative shutdown. However, additional equipment is added to the list to account for the effects of increased decay heat.

The EPU does not affect the alternative shutdown methods. The EPU does not modify the function of any mechanical component in the alternative safe shutdown flow paths or introduce any plant equipment failure modes which will impact the ability to achieve any of the alternative shutdown functions. The EPU does not adversely affect any components or circuits that provide power, control, or indication to components required for alternative safe shutdown. Enhancements will be made to selected components or control circuits to improve alternative shut down capability.

The requirement to achieve cold shutdown conditions within 72 hours after reactor shutdown continues to be met for EPU conditions.

To ensure an adequate pressurizer cooldown rate at EPU conditions, a contingency activity will be implemented to use auxiliary spray to reduce RCS pressure if a PORV is not available.

A time-critical task for restoration of charging flow after reactor shutdown will be modified to assure that pressurizer level will not decrease below the indicating range at EPU conditions.

An analysis shows that the city yard fire loop has the capability to simultaneously supply water to one emergency diesel generator, two standby AFW pumps, and two standby AFW room coolers.

The EPU does not affect the reactor coolant pump lube oil collection system.

Actions required to implement the use of two steam generators for three fire areas in order to meet the Appendix R 72-hour cooldown requirement at EPU conditions include the need to update applicable fire procedures.

2.7.1.1.4 Conclusion

The Ginna staff has reviewed the fire-related safe shutdown assessment and concludes that Ginna has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The Ginna staff further concludes that the Ginna Station Fire Protection Program will continue to meet the requirements of 10CFR50.48, Appendix R to 10CFR50, and will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-3 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to fire protection.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

2.7.2.1 Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in post accident environments. These systems at Ginna include the containment recirculation fan cooler (CRFC), the control room emergency air treatment system (CREATS), and the spent fuel pool SFP charcoal filter system. For each ESF atmosphere cleanup system, the Ginna review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits.

The NRC's acceptance criteria for the ESF atmosphere cleanup systems are based on:

- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident;

- GDC 41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents;

- GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and

- GDC-64, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents.

Specific review criteria are contained in NRC SRP Section 6.5.1.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predate those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, the Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional

safety margin. The Ginna current licensing basis for ESF Atmosphere Cleanup Systems is a composite of the SEP review, which was incorporated into the UFSAR, and system modifications as described in references 1 and 2. Specifically, the adequacy of the Ginna Station ESF Atmosphere Cleanup Systems design relative to conformance to:

- GDC-19 is described in UFSAR section 3.1.2.2.10, General Design Criterion 19 – Control Room. As described in this UFSAR section, the control room is designed to be capable of continuous occupancy by the operating personnel under all operating and accident conditions, within specified dose limits. The control room radiological analysis and habitability are described in reference 1. As stated in reference 2, the modification to CREATS, in conjunction with implementation of Alternate Source Term (AST) methodology for dose consequence calculation, is designed to be consistent with GDC-19 requirements for the duration of the postulated accident.

- GDC-41 is described in UFSAR section 3.1.2.4.12, Containment Atmosphere Cleanup. This UFSAR section describes two systems which are designed to provide fission product cleanup of the containment atmosphere following a postulated loss-of-coolant accident (LOCA). The containment spray system (CSS) includes the injection of sodium hydroxide solution into the containment spray to remove elemental iodine. The design and operation of the CSS and its sodium hydroxide injection system are described in UFSAR section 6.5.2. The second containment atmosphere cleanup system consists of charcoal filters that are placed in the air stream flow of two of the four containment recirculation fan coolers (CRFCs) to remove iodine. By letter dated February 25, 2005 (reference 1), the NRC issued a safety evaluation report (SER) to Ginna which, in part, eliminated the requirements for the these charcoal filter units on the basis that the revised radiological analyses for the postulated design basis accident which implements alternate source term (AST) methodology does not take credit for the containment post-accident charcoal filters. However, these charcoal filters remain available to provide post-accident dose mitigation and thus provide additional margin for accident dose reduction.

- GDC 61 is described in UFSAR section 3.1.2.6.2, Fuel Storage and Handling and Radioactivity Control. As described in this UFSAR section, the spent fuel pool (SFP) and cooling system is designed to ensure adequate safety under normal and postulated accident conditions. Control of airborne radioactivity in the spent fuel pool area is further described in UFSAR section 9.4.4 and LR section 2.7.4.

- GDC 64 is described in UFSAR section 3.1.2.6.5, General Design Criterion 64 – Monitoring Radioactive Releases. The containment atmosphere is continually monitored during normal and transient station operations using the containment particulate and gas monitors. Radioactivity levels contained in the facility effluent discharge paths and in the environs are continually monitored during normal and accident

conditions by the station radiation monitoring system and by the Radiation Protection Program for Ginna Station as further described in UFSAR sections 11.5 and 12.5.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's ESF Atmosphere Cleanup Systems were evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

Components of ESF Atmosphere Cleanup Systems that are within the scope of license renewal are described in SER sections 2.3.2.2, Containment Spray, and 2.3.3.10, Essential Ventilation.

2.7.2.2 Technical Evaluation

Control Room

The CREATS is normally in standby and is configured to provide zone isolation, re-circulation and filtration under accident conditions. The system is not designed to pressurize the Control Room Emergency Zone (CREZ) in any mode of operation. It is designed to be consistent with General Design Criteria (GDC) 19, "Control Room" and the 30 day dose acceptance criteria of 5 rem TEDE, provided in 10CFR50.67. Additional information on CREATS is provided in LR section 2.7.3.1. The CREATS is also designed to protect the operators from exposure to toxic gas following an accidental release from sources on or near the Ginna site. The filters are tested in accordance with Ginna Tech Specs Section 5.5.10, "Ventilation Filter Testing Program." The dose analysis for the EPU is presented in LR section 2.9.2. The EPU does not create additional chemical sources, and therefore, the toxic gas analysis is not affected by EPU.

Containment

The CSS and CRFCs are designed to remove fission products from the containment atmosphere following a LOCA. Sodium hydroxide injection into the CSS water serves to scrub fission products from the containment atmosphere. The CRFCs consist of four units, each includes high efficiency particulate air (HEPA) filters, and 2 units include charcoal adsorbers. Two of the four units are required during the post-accident period. Each unit has 30,000 cfm flow capacity. During normal plant operation, the charcoal filters are by-passed.

In the event of a LOCA, the air flow can be directed through the charcoal adsorbers. However, the charcoal adsorbers are not credited for evaluating potential radiological consequences, as stated in reference 1. The HEPA filter portion of the CRFCs is tested in accordance with the Ginna Technical Specification 5.5.10, "Ventilation Filter Testing Program." Two CRFC units recirculate 12,000 cfm within the lower (unsprayed) containment volume, and 48,000 cfm is assumed to mix the sprayed and unsprayed

volumes. The offsite and control room dose analyses, presented in LR section 2.9.2, demonstrate the effectiveness of the CRFC to minimize the release of radioactivity to the environment, following a LOCA.

CRFC Charcoal

UFSAR section 6.5.1.2.3 addresses heat generation in the CRFC charcoal. The EPU dose analysis, based on the AST, does not credit the charcoal, but rather credits the HEPA. Also, since the charcoal and HEPA are in separate locations/housings, high temperature in one has no effect on the other. HEPA fire is not likely (glass/asbestos filter medium, steel frames, typically tested to about 1000°F) and heating does not result in resuspension of fission products.

The EPU increases the core radioactive source by about 20%. The impact of the EPU on the CRFC post-LOCA charcoal filter heat generation, and dissipation, are addressed in terms of offsite and control room dose and potential charcoal ignition. The charcoal system is designed so that the decay heat, produced by the collected fission products, will not cause ignition of the charcoal or overheating to the point of desorption of the collected fission products.

Offsite and Control Room Doses

The radiological consequences of the LOCA, without credit for iodine removal by the CRFC charcoal adsorber, are presented in LR section 2.9.2. The resulting doses are appropriately within the acceptance criteria.

Decay Heat Generation in the Charcoal Adsorbers

The total ($\beta+\gamma$) decay heat generation rate of 6.5E18 MeV/sec, presented in UFSAR section 6.5.1.2.3.1, is based on a TID release (25% of core halogens). The corresponding value for the EPU with AST is estimated to be 5E17 MeV/sec, based on 2% of the core elemental + organic iodine.

Decay Heat Dissipation with Normal Air Flow

The UFSAR concludes that the normal air flow rate is capable of dissipating the decay heat produced by the collected fission products, with a safety margin greater than 2000. The EPU with AST, is estimated to result in about one-tenth (see above section) the collected fission products. Based on this, the EPU will not adversely impact decay heat dissipation with normal air-flow.

Decay Heat Dissipation with Loss of Air Flow

The UFSAR presents cases for air flow loss at 1 and 24 hours post-LOCA. It is concluded that the dousing system is not required to mitigate the consequences of any analyzed accident. The EPU with AST is estimated to result in about one-tenth (see above section) the collected fission products. Based on this, the EPU will not adversely impact decay heat dissipation with loss of air flow.

The containment atmosphere is continually monitored during normal and transient station operations using the containment particulate and gas monitors. Radioactivity levels contained in the facility effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the Radiation Protection Program for Ginna Station. The ability of these monitors to perform their function is not affected by the EPU.

Auxiliary Building and Spent Fuel Pool

The spent fuel pool (SFP) charcoal adsorber system is not an ESF system. However, Ginna Technical Specifications require operation of the system during irradiated fuel movement within the Auxiliary Building when one or more fuel assemblies in the Auxiliary Building have decayed less than 60 days. The radiological analysis of the fuel handling accident (FHA) in the Auxiliary Building takes credit for iodine removal by the SFP charcoal adsorber system. The charcoal filters are tested in accordance with Ginna Technical Specification 5.5.10, "Ventilation Filter Testing Program." The FHA radiological analysis is presented in LR section 2.9.2. The offsite and control room dose analyses, presented in LR section 2.9.2, demonstrate the effectiveness of the SFP charcoal adsorber to minimize the release of radioactivity to the environment, following an FHA in the Auxiliary Building. The Spent Fuel Pool area ventilation is further discussed in LR section 2.7.4.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Components of the ESF Atmosphere Cleanup Systems that are within the scope of License Renewal are described in NUREG-1786, sections 2.3.2.2 and 2.3.3.10. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, sections 3.3.2.4.10 and 3.3.2.5. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the ESF Atmosphere Cleanup Systems at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.2.3 Results

The proposed EPU has no effect on the ability of ESF Atmosphere Cleanup Systems to control the release of radioactivity to the environment within regulatory limits. Control of post-accident radiation in the control room is discussed in LR section 2.7.3.1.

2.7.2.4 References

1. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), "R.E. Ginna Nuclear Power Plant – Modification of the Control Room Emergency Air Treatment System and Change to Dose Calculation Methodology to Alternate Source Term (TAC No. MB9123)," dated February 25, 2005.

2. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), "R.E. Ginna Nuclear Power Plant – Correction to Amendment No. 87 Re: Modification of the Control Room Emergency Air Treatment System (TAC No. MB9123)," dated May 18, 2005.

2.7.2.5 Conclusion

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The Ginna staff concludes that the assessment adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU. The Ginna staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU. Based on this, the Ginna staff concludes that the ESF atmosphere cleanup systems will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-19, GDC-41, GDC-61, and GDC-64. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Ventilation Systems

2.7.3.1 Control Room Area Ventilation System

2.7.3.1.1 Regulatory Evaluation

The function of the Control Room HVAC system (CRHVAC) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The Ginna Nuclear Power Plant, LLC (Ginna) review of the CRHVAC focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRHVAC.

The NRC's acceptance criteria for the CRHVAC are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;

- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific review criteria are contained in SRP Section 9.4.1.

Ginna Current Licensing Basis

By letter dated February 25, 2005 (reference 1), the NRC issued a Safety Evaluation Report which approved certain changes to the Ginna Control Room Emergency Air Treatment System (CREATS) and implementation of Alternate Source Term (AST) methodology for accident dose consequence calculation. This SER was supplemented by letter dated May 18, 2005 (reference 2). The dose consequence of the limiting loss-of-coolant accident (LOCA) was reanalyzed for the control room using AST methodology. Based upon implementation of the changes described in reference 1, the NRC concluded that Ginna will continue to provide reasonable assurance that the radiological consequences in the control room of the postulated LOCA will meet the acceptable radiation dose criteria of 10CFR50.67. In addition, the total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67 replaces the previous dose guidelines of GDC-19.

In conjunction with references 1 and 2, the adequacy of the Ginna CRHVAC system relative to conformance to:

- GDC-4 is described in UFSAR section 3.1.2.1.4, General Design Criterion 4 – Environmental and Missile Design Basis. As described in this UFSAR section, safety-related structures, systems, and components (SSCs) shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accident conditions including loss-of-coolant accident (LOCA). As discussed in UFSAR section 6.4.1, the control room is continuously occupied by the operating personnel under all operating and accident conditions. It is provided with sufficient shielding, ventilation, and habitability provisions to ensure that control room personnel can perform all required safety functions from the control room under all credible postulated accident conditions. The changes to the CRHVAC system approved by the NRC in reference 1, as supplemented in reference 2, do not alter these provisions.

- GDC-19 is described in UFSAR section 3.1.2.2.10, General Design Criterion 19 – Control Room. As described in this UFSAR section, the control room is designed to be capable of continuous occupancy by the operating personnel under all operating and accident conditions, within specified dose limits. The control room radiological analysis and habitability are described in reference 1. As stated in reference 2, the modification to CREATS, in conjunction with implementation of AST methodology for dose consequence calculation, is designed to be consistent with GDC-19 requirements for the duration of the postulated accident.

- GDC-60 is described in UFSAR section 3.1.2.6.1, General Design Criterion 60 – Control of Releases of Radioactive Materials to the Environment. As described in this UFSAR section, GDC-60 is directed at suitably controlling the release of radioactive materials during normal operation and anticipated operational occurrences. Operation of the Control Room Emergency Air Treatment System (CREATS) under postulated accident conditions does not contribute to the generation of radioactive materials. Its operation, however, may serve to assist in the control of release of radioactive material that may bypass post-accident filtration processes via the control room. As described in UFSAR 6.4.2.2.2, in the event of high radioactivity in the control room atmosphere, CREATS will be aligned to provide a closed, filtered ventilation cycle until control room radiation is reduced to a safe level. As stated in reference 1, CREATS is being upgraded to improve its reliability and performance. As stated in reference 2, the modification to CREATS will maintain the levels of radioactive effluent control consistent with regulatory requirements.

In addition to the evaluations described in the Ginna UFSAR, Ginna Station's systems and components have been evaluated for plant license renewal. Plant system and component materials of construction, operating history, and programs to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

Components of the CRHVAC system that are within the scope of license renewal are identified in SER section 2.3.3.10.

2.7.3.1.2 Technical Evaluation

The CRHVAC consists of two separate subsystems

- Normal control room heating, ventilation and air conditioning
- CREATS

The Control Room Emergency Zone (CREZ) is limited to the top floor of the three-story control building. It includes the control room, restroom, kitchen, and Shift Manager's office and all CREATS ductwork bounded by the CREZ isolation dampers. These two systems provide three different modes of operation; Normal, Purge and Emergency.

Normal Ventilation System

The normal HVAC system is located on the bottom floor of the Control Building and is connected to the CREZ by supply and return ducts. In the Normal mode of operation, this system provides the control room with fresh outside air, exhaust, coarse filtration, and temperature control to provide the operators with a safe and comfortable working environment. In the Purge mode of operation, this system provides the maximum amount of fresh air to purge airborne contaminants from the CREZ. The normal HVAC system's outside air intake duct is equipped with redundant trains of radiation, chlorine, and ammonia monitors, any of which will actuate the emergency mode of operation and provide an alarm in the control room. The normal HVAC system is also equipped with a smoke detector, upstream of the normal return air fan, to monitor the return airflow from the CREZ and to provide an alarm in the control room.

CREATS

The Control Room Emergency Air Treatment System (CREATS) is normally in standby and is configured to provide zone isolation, re-circulation and filtration under accident conditions. The system isolates the normal HVAC system from the CREZ and re-circulates the CREZ air through HEPA and charcoal filter banks, but is not designed to pressurize the CREZ. It is designed to satisfy General Design Criteria (GDC) 19, "Control room" and the 30 day dose acceptance criteria of 5 rem TEDE, provided in 10CFR50.67. The CREATS is also designed to protect the operators from exposure to smoke and toxic gas. Detailed dose analyses for EPU conditions are given in LR section 2.9.2, and the results are presented in Table 2.9-1.

As approved by the NRC in references 1 and 2, the CREATS consists of two seismic Category I, 100% capacity trains that are designed to filter, cool, heat, and re-circulate 6000 cfm (+/- 10%) of control room air. The CREATS fans are powered from Class 1E safeguard buses and will start upon a manual, toxic gas, radiation, or safety injection (SI) signal. The heating and cooling coils installed within the CREATS are designed to maintain the CREZ between a minimum temperature of 50°F and a maximum temperature of 104°F to support human habitability and equipment operation.

The major components of the CREATS are located inside the relay room annex, which is east of, and one level below, the control room. The relay room annex is a hardened structure having reinforced concrete outside walls and roof. Ductwork connects the CREATS to the CREZ via penetrations in the roof of the relay room annex and in the east wall of the control room. This ductwork is part of the CREZ isolation boundary, and, because it is located outside, it is designed to survive tornado winds speeds of up to 132 mph. An evaluation of the seismic analysis of the external ductwork is summarized in references 1 and 2.

Appendix R analysis, environmental and dynamic effects, control room habitability requirements, and TS changes are summarized in references 1 and 2.

The CREATS isolation times and iodine removal efficiencies are presented in LR Table 2.9.2-3.

NRC acceptance of the CREATS system design prior to EPU is documented in references 1 and 2.

Comparison to CREATS/Alternate Source Term NRC SER (references 1 and 2)

The design and operation of the control room ventilation system are unaffected by the EPU. Smoke, toxic gas and external event assumptions and conclusions are also unaffected by the EPU. Radiological analysis methods and assumptions for the EPU are consistent with those presented in reference 1 and 2 with the exception of radiological consequences as detailed in LR section 2.9.2. Specific changes include:

- Adsorber testing criteria was revised as part of the Technical Specification amendment associated with references 1 and 2 to reflect the Regulatory Guide 1.52, Rev. 3 requirements, Therefore, increased efficiencies of the CREATS filtration trains were credited as part of the EPU radiological evaluations (LR section 2.9.2).

- Core inventory is consistent with EPU.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Components of the Control Room Ventilation System that are within the scope of License Renewal are described in NUREG-1786, section 2.3.3.10, Essential Ventilation. The new CREATS is also in the scope of License Renewal, per 10CFR50.47(b). Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, sections 3.3.2.4.10 and 3.3.2.5. These are also applicable to CREATS. There are no modifications or additions to system components as the result of EPU that would introduce any new functions or change the functions of existing components that would affect the license renewal system evaluation boundaries. Operation of the CRHVAC system at EPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.3.1.3 Results

The proposed EPU has no effect on the ability of the CRHVAC to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. Ginna has adequately accounted for the increase of radioactive gases that would result from a DBA under the proposed EPU operating conditions, and any associated changes to parameters affecting environmental conditions for control room personnel and equipment. The CRHVAC will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. As stated in references 1 and 2, the design criteria, design bases, and safety classification for the CREATS, and the requirements for system performance continue to provide conformance with the requirements of GDC-4, GDC-19, and GDC-60. Plant operation at the proposed EPU conditions as analyzed utilizing AST methodology would not alter this conclusion. Based on this, the CREATS will continue to meet the requirements of GDC-4, GDC-19, and GDC-60 for EPU operating conditions. Therefore, the proposed EPU is acceptable with respect to the CRHVAC.

2.7.3.1.4 References:

1. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), "R.E. Ginna Nuclear Power Plant – Modification of the Control Room Emergency Air Treatment System and Change to Dose Calculation Methodology to Alternate Source Term (TAC No. MB9123)," dated February 25, 2005.

2. Letter to Mrs. Mary G. Korsnick (Ginna NPP) from Donna M. Skay (NRC), "R.E. Ginna Nuclear Power Plant – Correction to Amendment No. 87 Re: Modification of the Control Room Emergency Air Treatment System (TAC No. MB9123)," dated May 18, 2005.

2.7.3.1.5 Conclusion

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the ability of the CRHVAC to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The Ginna staff concludes that Ginna Station has adequately accounted for the increase of radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, the Ginna staff concludes that the CRHVAC will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The Ginna staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the Ginna staff concludes that the CRHVAC will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-4, GDC-19, and GDC-60. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the CRHVAC.

2.7.4 Spent Fuel Pool Area Ventilation System

2.7.4.1 Regulatory Evaluation

The function of the spent fuel pool area ventilation system is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, anticipated operational occurrences, and following postulated fuel-handling accidents. The Ginna Nuclear Power Plant, LLC's (Ginna) review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system.

The NRC's acceptance criteria for the spent fuel pool area ventilation system are based on:

- GDC-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.
- GDC-61, insofar as it requires that systems containing radioactivity be designed with appropriate confinement and containment.

Specific review criteria are contained in SRP Section 9.4.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Conformance to GDC-60 and 61 is addressed in Ginna UFSAR sections 3.1.2.6.1 and 3.1.2.6.2, respectively. GDC-60 requires that the plant design includes means to control suitably the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

GDC-61, relevant to the spent fuel pool ventilation system and gaseous effluents treatment system, requires that the ventilation system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit

appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, and (3) with appropriate containment, confinement, and filtering systems. Ginna UFSAR section 3.1.2.6.2 states that the gaseous waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

- Components are designed and located such that appropriate periodic inspection and testing may be performed.
- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Ginna UFSAR chapter 12.
- Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

The function of the spent fuel pool area ventilation system portion of the auxiliary building ventilation system is addressed in plant technical specification bases B3.7.10.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's spent fuel pool area ventilation system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

The above SER, discusses the spent fuel pool area ventilation system in section 2.3.3.10, Essential Ventilation. Aging effects, and the programs used to manage the aging effects are described in License Renewal SER section 3.3.2.4.10, Essential Ventilation.

2.7.4.2 Technical Evaluation

2.7.4.2.1 Introduction

The spent fuel pool area ventilation system is part of the auxiliary building ventilation system. The auxiliary building ventilation system is described in Ginna UFSAR section 9.4.2 and the spent fuel pool area ventilation subsystem is described in Ginna UFSAR section 9.4.4. The impact of the proposed EPU on the auxiliary building ventilation system is further evaluated in LR section 2.7.6, Engineered Safety Features Ventilation Systems. As described in the UFSAR, the spent fuel pool ventilation system serves to control airborne radioactivity in the spent fuel pool area during normal operating conditions. This is accomplished by directing air from the auxiliary building supply air unit across both the spent fuel pool and the decontamination pit to exhaust air ducts, which are connected to the suction of the auxiliary building exhaust fan C. Exhaust air from the spent fuel pool water surface is drawn through

roughing filters and, depending on system alignment, charcoal filters. Discharge from the auxiliary building exhaust fan C passes through HEPA filters, a main auxiliary building exhaust fan, and then out the plant vent. During handling of recently irradiated fuel in the auxiliary building, plant technical specifications require that the spent fuel pool charcoal adsorber system, including its associated fans, be in operation.

2.7.4.2.2 Description of Analyses and Evaluations

The spent fuel pool area ventilation system was evaluated to ensure it is capable of performing its intended functions at EPU conditions. The decay heat loads in the spent fuel pool increase due to the EPU conditions. EPU decay heat loads and pool water temperatures have been evaluated to ensure that the system is capable of performing its intended functions under normal EPU and refueling modes. The activities that occur in the decontamination pit are unaffected by EPU, therefore there are no impacts of that portion of the ventilation system due to EPU. Other evaluations are addressed in the following LR section:

- Spent fuel pool cooling system evaluation – LR section 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System
- Offsite dose consequences of a fuel handling accident – LR section 2.9.2, Radiological Consequences of Fuel Handling Accident
- Control of the release of radioactive effluents – LR section 2.10.1, Occupational and Public Radiation Doses

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The spent fuel pool area ventilation system is within the scope of License Renewal. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. There are no changes associated with operation of the spent fuel pool ventilation system at EPU conditions and the EPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.4.3 Results

The air temperature in the spent fuel pool area is affected by heat released from the spent fuel pool. Although the decay heat in the spent fuel is greater at EPU conditions, the spent fuel pool water temperature during normal and abnormal EPU operation does not exceed the current values. Therefore, the spent fuel pool area ventilation system will maintain the required air temperature conditions for personnel and equipment during EPU operation. Refer to LR section 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System. Also, there is no impact to the air temperature in or above the decontamination pit as a result of EPU.

The design of the spent fuel pool area ventilation system has not changed following the implementation of the EPU. Airborne radioactivity released from the spent fuel in the pool will continue to be collected and exhausted by the auxiliary building ventilation system. Therefore, the control of airborne radioactivity in the spent fuel pool area is not affected following implementation of the EPU.

The evaluation of the spent fuel pool area ventilation system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This system was evaluated in LR section 2.10.1, Occupational and Public Radiation Doses, and no changes are required as a result of EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, and is described in the offsite dose calculation manual.

The evaluation of the spent fuel pool area ventilation system at EPU conditions demonstrates that the Ginna Station will continue to meet the current licensing basis with respect to GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the EPU.

The evaluation of the ability of the spent fuel pool area ventilation system to maintain the required temperature conditions and to contain radioactivity to permit personnel access during EPU demonstrates that there is no impact on the system design capability following EPU implementation.

2.7.4.4 Conclusions

Ginna has assessed the effects of the proposed EPU on the spent fuel pool area ventilation system. Ginna concludes that the evaluation adequately accounts for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, Ginna concludes that the spent fuel pool ventilation system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-60 and -61. Therefore, the proposed EPU is acceptable with respect to the spent fuel pool and decontamination pit areas ventilation system.

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

2.7.5.1 Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system and the turbine area ventilation system is to maintain ambient temperatures in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during anticipated operational occurrences, and after postulated accidents. The Ginna Nuclear Plant, LLC (Ginna) review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC acceptance criteria for the auxiliary and radwaste area ventilation system and turbine area ventilation system are based on GDC-60, insofar as it requires that the plant design includes means to control the release of airborne radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

The Ginna UFSAR describes the various plant ventilation systems, primarily in section 9.4, Air Conditioning, Heating, Cooling and Ventilation Systems. It is noted that the building in which functionally important equipment is located varies from site to site as do the names used to commonly designate those buildings. In order to minimize confusion and to maintain consistency with previous NRC staff evaluations performed on Ginna, the systems reviewed in this section are designated "Nonessential Ventilation Systems" as used in the Ginna License Renewal safety evaluation report, NUREG-1786. Systems included in this group, and discussed in the following evaluation, are located in the turbine building, service building, and all-volatile secondary water treatment building.

(The ventilation systems important to personnel safety or vital equipment operation are addressed in LR section 2.7.6, Engineered Safety Feature Ventilation System, which is composed primarily of the license renewal grouping called essential ventilation systems. Other systems are addressed elsewhere as called out in RS-001)

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Conformance to GDC-60 is addressed in Ginna UFSAR section 3.1.2.6.1. GDC-60 requires that the plant design include means to control suitably the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's nonessential ventilation system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

The above SER discusses the nonessential ventilation system in section 2.3.3.19, Nonessential Ventilation. The SER concluded that no nonessential ventilation system components are within the scope of license renewal. Therefore, none of the nonessential ventilation system components is subject to aging management review.

2.7.5.2 Technical Evaluation

2.7.5.2.1 Introduction

The nonessential ventilation system provides heating, ventilation and air conditioning to non-vital areas and plant equipment. The principal components of the nonessential ventilation system are filters, fans, dampers, valves, heat exchangers, conditioning/chiller packages, and the ductwork, piping and valves. The nonessential ventilation system serves the turbine, service and all-volatile-treatment buildings.

The turbine building ventilation system is discussed in Ginna UFSAR section 9.4.5. The majority of the building does not require an integrated heating, ventilation and air conditioning system; it uses independent roof vent fans, wall vent fans, windows, and steam unit heaters for ventilation and temperature control. Included in the turbine building is the main feedwater pump room. The main feedwater pump equipment cooling system blends a mixture of outside air and room air to control the room and equipment temperatures. No mechanical means of heating or cooling is provided or necessary.

The service building ventilation system is discussed in Ginna UFSAR section 9.4.6. It consists of air handling units serving the various areas of the service building. Air from uncontaminated areas is exhausted through roof exhaust fans. Air from areas of potential contamination, such as laboratories equipped with hoods and the dress out area are exhausted through the intermediate building controlled access area exhaust fans. Controlled access area fans 1A and 1B can draw air through a common HEPA and charcoal filter, a low-flow alarm and dampers, and discharge to the auxiliary building HEPA filter, which is exhausted by the main auxiliary building exhaust system to the main vent header. The intermediate building and auxiliary

building ventilation systems are described and evaluated in LR section 2.7.6, Engineered Safety Feature Ventilation System.

The all-volatile-treatment building ventilation system is discussed in Ginna UFSAR section 9.4.7. It provides ventilation and heating to maintain required temperatures for the all-volatile-treatment (condensate demineralizer) building and the condensate booster pump area of the turbine building. For the all-volatile-treatment building, including the demineralizer area control room, ventilation and cooling is supplied through outside air intakes by fans and modulating dampers controlled by thermostats. Steam heating coils warm the air for the demineralizer area control room, when necessary. For the condensate booster pump area, ventilation and cooling is supplied by thermostatically controlled outside air intakes, fans, and dampers. No heating is required.

2.7.5.2.2 Description of Analyses and Evaluations

The nonessential ventilation system EPU heat loads were evaluated to ensure that the system is capable of performing its intended functions under normal EPU conditions. The evaluation considered whether heat load changes impacted the maximum ambient temperature for each area.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

The nonessential ventilation systems are not within the scope of License Renewal. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. There are no changes associated with operation of the nonessential ventilation systems at EPU conditions and the EPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.5.3 Results

The nonessential ventilation system's ability to provide required temperature conditions for personnel and equipment during normal operation is unaffected by the changes proposed for EPU. The increased heat loads in these areas are primarily due to changes in the main steam and feedwater system operating conditions, increased BHP for the condensate booster and feedwater pumps, and small increases in electrical loads. For plant areas where temperature is controlled by air conditioning units, the small increase in heat loads is well within the capacity of the units. For plant areas that use outside air exchange to provide cooling, outside air temperature changes dominate any potential temperature changes caused by EPU.

The evaluation of the plant equipment changes for the proposed EPU did not identify any need to modify the nonessential ventilation system. There are no equipment changes as a result of EPU that could create a new potentially unmonitored airborne radioactive release path. Thus,

following the EPU, Ginna Station will continue to meet the current licensing basis with respect to GDC-60. The effects of potential releases to the environment are evaluated in LR section 2.10.1, Occupational and Public Radiation Doses, and remain within current limits following the EPU. The handling, control, and release of radioactive materials are in compliance with 10CFR50, Appendix I, as described in the Offsite Dose Calculation Manual.

The evaluation of the nonessential ventilation system demonstrates that no changes are required to the system. Therefore, the design capability of the system to maintain an acceptable building environment that will permit personnel access following implementation of the EPU is not impacted.

2.7.5.4 Conclusions

Ginna has assessed the effect of the proposed EPU on the nonessential ventilation system. Ginna concludes that the evaluation has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the turbine, service and all-volatile-treatment buildings, permit personnel access, control airborne radioactivity in the area, and control release of gaseous radioactive effluents to the environment. Based on this, Ginna concludes that the nonessential ventilation system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-60. Therefore, Ginna finds that the proposed EPU is acceptable with respect to the nonessential ventilation system.

2.7.6 Engineered Safety Feature Ventilation System

(Note that, the environmental control for engineered safety feature components that are located inside containment is covered in LR section 2.6.5, Containment Heat Removal.)

2.7.6.1 Regulatory Evaluation

The function of the engineered safety feature ventilation system is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents. The Ginna Nuclear Power Plant, LLC's (Ginna) review of the engineered safety feature ventilation system focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the systems. The Ginna review covered:

- The ability of the engineered safety feature equipment in the areas being serviced by the ventilation system to function under degraded engineered safety feature ventilation system performance.
- The capability of the engineered safety feature ventilation system to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel).
- The capability of the engineered safety feature ventilation system to control airborne particulate material (dust) accumulation.

The NRC's acceptance criteria for the engineered safety feature ventilation system are based on:

- GDC-4, insofar as it requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of safety-related structures, systems, and components.
- GDC-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

Specific review criteria are contained in SRP Section 9.4.5.

The Ginna UFSAR describes the various plant ventilation systems, primarily in section 9.4, Air Conditioning, Heating, Cooling and Ventilation Systems. In order to maintain consistency with previous NRC staff evaluations performed on Ginna, the systems reviewed in this section are designated "Essential Ventilation Systems" as used in the Ginna License Renewal safety

evaluation report, NUREG-1786. Systems included in this group, and discussed in the following evaluation, are located in the auxiliary building, intermediate building, standby auxiliary feedwater building, and diesel generator building.

(The control room emergency air treatment system, control building ventilation for areas other than the control room and the technical support center ventilation systems are discussed in LR section 2.7.1, Control Room Habitability System. Other systems are addressed elsewhere as called out in RS-001.)

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50 Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821 Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station essential ventilation designs relative to conformance with:

- GDC 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in that UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC 4 is described in the following:
    - Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
    - Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)
    - Missile Protection (Ginna UFSAR section 3.5)

- GDC 17 is addressed in Ginna UFSAR section 3.1.2.2.8 for electrical power supply for equipment important to safety. GDC 17 requires that the plant safety related systems and components be provided with offsite and onsite power to accommodate operating conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. GDC 60 requires that the plant design include means to control suitably the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's essential ventilation system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

The above SER, discusses the ventilation systems evaluated within this LR section in section 2.3.3.10, Essential Ventilation. Aging effects and the programs used to manage the aging effects are described in License Renewal SER section 3.3.2.4.10, Essential Ventilation.

2.7.6.2 Technical Evaluation

2.7.6.2.1 Introduction

The essential ventilation system functions to maintain temperatures within specified limits in areas containing safety-related equipment. Normal ventilation exhausts from potentially contaminated areas are filtered and the discharge is monitored for radiation. Included in the scope of the essential ventilation system are the following subsystems:

- Auxiliary building ventilation (described in Ginna UFSAR section 9.4.2)

- Intermediate building ventilation (described in Ginna UFSAR section 9.4.2)

- Standby auxiliary feedwater building ventilation (described in Ginna UFSAR section 9.4.9)

- Diesel generator building ventilation (described in Ginna UFSAR section 9.4.9)

The auxiliary building has a nonsafety heating, ventilation, and air conditioning system, which provides clean, filtered, and tempered air to the operating floor of the auxiliary building and to

the surface of the decontamination pit and spent fuel storage pool. The system exhausts air from the equipment rooms and open areas of the auxiliary building and the decontamination pit and the spent fuel pool through a closed exhaust system. The exhaust system includes a 100 percent capacity bank of high-efficiency particulate air filters and redundant 100 percent capacity fans discharging to the atmosphere via the plant vent. This arrangement ensures the proper direction of air flow for removal of airborne radioactivity from the auxiliary building. In addition to the main auxiliary building ventilation system, the residual heat removal, safety injection, containment spray and charging pump motors are provided with additional cooling capability.

The spent fuel pool area ventilation system is a part of the auxiliary building ventilation system. Refer to LR section 2.7.4, Spent Fuel Pool Area Ventilation System for the evaluation of this system.

Air is introduced to the clean side of the intermediate building through two wall dampers mounted in the outside wall, and from the turbine building through a damper mounted in the wall common to both buildings. A supply fan moves air from the intermediate building clean side to its restricted area side. Two exhaust fans, located in the intermediate building restricted area side, draw air from both the clean and restricted area sides of the building and discharge to the auxiliary building discharge header plant vent duct. There are also four roof ventilators on the clean side to provide additional exhaust, and a fan mounted in a floor grating to move basement level air up to higher floor levels in the clean side. Within the intermediate building, control rod drive mechanism control cabinets are served by self-contained air conditioning units.

The standby auxiliary feedwater pump building cooling and heating system provides heating or cooling as necessary to provide an acceptable environment for the safety related equipment housed within the building. Each standby auxiliary feedwater pump building cooling unit uses service water as a cooling medium and is automatically started whenever its corresponding standby auxiliary feedwater pump is started.

The diesel generators are housed in adjacent but separate rooms, each of which is serviced by a safety-related ventilation system having two inlet fans supplying outside air. Excess air is discharged to the outdoors through automatic, pressure-actuated room vents, backdraft dampers, and wall-mounted louvers. No refrigeration or service water air cooling is used.

2.7.6.2.2 Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems in areas served by the essential ventilation system were evaluated to ensure that the ventilation systems are capable of performing their intended functions under EPU conditions including the ability of the system to control airborne particulate material accumulation.

The essential ventilation systems were reviewed for impacts as a result of EPU on any redundancies and diversities provided in the original design to ensure adequate operation with degraded components, and to prevent or dissipate flammable or explosive vapors.

Other evaluations related to the essential ventilation system are addressed in the following LR section:

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and LR section 2.5.1.3, Pipe Failures

- Electrical equipment qualification - LR section 2.3.1, Environmental Qualification of Electrical Equipment

- Onsite and offsite electric power systems, including GDC-17 requirements – LR section 2.3.2, Offsite Power System, and LR section 2.3.3, AC Onsite Power System

- Protection against turbine missiles and internal missiles is discussed in LR section 2.5.1.2, Missile Protection

- The control room ventilation system, including the relay and battery rooms, and the technical support ventilation system are evaluated in LR section 2.7.1, Control Room Habitability System.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the essential ventilation system are within the scope of License Renewal. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Slightly increased heat loads are well within the capability of the current ventilation systems. Because no modifications are necessary for essential ventilation system components, EPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.6.3 Results

The auxiliary and intermediate building, restricted side, area air temperatures are not significantly affected after implementation of the EPU. The increased heat load in the intermediate building, clean side, is primarily due to the changes in the main steam and feedwater system operating conditions. The intermediate building clean side uses outside air exchange to provide cooling, outside air temperature changes dominate any potential temperature changes caused by EPU. The EPU evaluation determined that the effect of EPU on the normal operating temperatures increased by less than 1°F and that the maximum normal operating design temperature of 104°F is not exceeded.

Heat loads in the standby auxiliary feedwater pump room do not increase after implementation of the EPU (Refer to LR section 2.5.4.5, Auxiliary Feedwater). Therefore, the ventilation

system's ability to provide required temperature conditions for personnel and equipment is not impacted by EPU.

The diesel generator loading is not increased after implementation of the EPU (Refer to LR section 2.3.3, AC Onsite Power System). Therefore, the ventilation system's ability to provide the required temperature conditions for personnel and equipment is not impacted by EPU.

The evaluation of the plant equipment changes for the proposed EPU did not identify any need to modify the essential ventilation system. There are no equipment changes as a result of EPU that could change the existing capability of the essential ventilation system under degraded conditions.

Likewise, there are no equipment changes as a result of EPU that could create a new potentially unmonitored radioactive release path. Thus, following the EPU, Ginna Station will continue to meet the current licensing basis with respect to GDC-60. The effects of potential releases to the environment are evaluated in LR section 2.10.1, Occupational and Public Radiation Doses, and remain within current limits following the EPU.

There are no equipment changes as a result of EPU that could affect the accumulation or dissipation of flammable or explosive vapors. Thus, following EPU, the ventilation systems will continue to circulate sufficient air to prevent flammable or explosive vapors.

The evaluation of the essential ventilation system demonstrates that no changes are required to the system. Therefore, the design capability of the system to maintain an acceptable building environment related to control airborne particulate material accumulation is not impacted.

The evaluations of dynamic effects, electrical equipment qualification, onsite and offsite power, and internal and external missiles, regarding the impact of EPU on essential ventilation systems, all showed acceptable results.

2.7.6.4 Conclusions

The Ginna review of the essential ventilation system has adequately accounted for the effects of the proposed EPU on the ability of the essential ventilation system to provide a suitable and controlled environment for ESF components. The Ginna review further concluded that the essential ventilation system will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, Ginna concludes that the essential ventilation system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC–4, GDC-17, and GDC-60. Therefore, Ginna finds the proposed EPU acceptable with respect to the essential ventilation system.

2.7.7 Other Ventilation Systems (Containment)

2.7.7.1 Regulatory Evaluation

The functions of the containment ventilation system are to provide heat removal from the containment atmosphere, to remove radioactive materials from the containment atmosphere, and to provide containment pressure control under normal and accident conditions. The Ginna Nuclear Power Plant, LLC (Ginna) review of the containment ventilation system focused on the effects that the proposed EPU will have on the functional performance of system.

The acceptance criteria for the containment ventilation system are based on:

- GDC-4, insofar as it requires that safety-related structures, systems, and components be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents
- GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of safety-related structures, systems, and components
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents
- GDC-61, insofar as it requires that systems containing radioactivity be designed with appropriate confinement and containment

Ginna Current Licensing Basis

As noted in Ginna UFSAR section 3.1, the general design criteria used during the licensing of Ginna Station predates those provided today in 10CFR50, Appendix A. The adequacy of the Ginna design relative to the general design criteria is discussed in Ginna UFSAR sections 3.1.1 and 3.1.2. In the late 1970s the Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear power plants to reconfirm and document their safety. The results of the SEP review of the Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), completed in August 1983. The IPSAR describes the methods used by the NRC to assess conformance of the Ginna design to the then current licensing criteria, and identifies cases where bringing the plant into, or closer to, conformance with the newer criteria would provide significant and beneficial additional safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

Specifically, the adequacy of the Ginna Station containment ventilation design relative to conformance to:

- GDC 4 is described in Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. As described in this Ginna UFSAR section, Ginna Station received post-construction review as part of the Systematic Evaluation Program (SEP). The results of this review are documented in NUREG-0821, Integrated Plant Safety Assessment systematic Evaluation Program, R. E. Ginna Nuclear Power Plant. Conformance to the requirements of GDC 4 is described in the following:
    - • Environmental Design Of Mechanical And Electrical Equipment (Ginna UFSAR section 3.11)
    - • Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping (Ginna UFSAR section 3.6)
        - Pipe Breaks Inside Containment (SEP Topic III-5.A)
        - Pipe Breaks Outside Containment (SEP Topic III-5.B)
    - • Missile Protection (Ginna UFSAR section 3.5)
- GDC 17 is addressed in Ginna UFSAR section 3.1.2.2.8 for electrical power supply for safety-related equipment. GDC 17 requires that the plant safety related systems and components be provided with offsite and onsite power to accommodate operating conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC 60 is addressed in Ginna UFSAR section 3.1.2.6.1. GDC 60 requires that the plant design include means to control suitably the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. Ginna UFSAR section 3.1.2.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10CFR50, Appendix I, and is described in the Offsite Dose Calculation Manual.
- GDC 61 is addressed in Ginna UFSAR section 3.1.2.6.2. The requirements of GDC 61 relevant to the containment ventilation system and gaseous effluents treatment system require that the ventilation system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems

Ginna UFSAR section 3.1.2.6.2 states that the gaseous waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

- Components are designed and located such that appropriate periodic inspection and testing may be performed
- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Ginna UFSAR chapter 12

Other Ginna UFSAR sections that address the design features and functions of the containment ventilation system include:

- Ginna UFSAR section 6.2.4, Containment Isolation System, which describes containment isolation features to isolate the containment boundaries in the containment ventilation system post-accident

- Ginna UFSAR section 6.2.2, Containment Heat Removal Systems, which describes means of heat removal from the containment atmosphere with containment recirculation fan coolers and the containment spray system under accident conditions

In addition to the evaluations described in the Ginna UFSAR, the Ginna Station's containment ventilation system was evaluated for License Renewal. Systems and system component materials of construction, operating history and programs used to manage aging effects are documented in:

- License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004

With respect to the above SER, the containment ventilation system is included in section 2.3.3.9, Containment Ventilation. Aging effects and the programs used to manage the aging effects are described in License Renewal SER section 3.3, Auxiliary Systems.

2.7.7.2 Technical Evaluation

2.7.7.2.1 Introduction

The Containment Ventilation Systems are described in Ginna UFSAR section 9.4.1.

The containment ventilation system is designed to accomplish the following:

- Remove the normal heat loss from the equipment and piping in the containment during plant operation and maintain a normal ambient temperature below about 120°F and 50% relative humidity

- Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the reactor containment within 2 hours after reactor shutdown

- Provide for positive circulation of air across the refueling water surface to ensure personnel access and safety during shutdown

- Provide a minimum containment ambient temperature of 50°F during reactor shutdown

- Provide for purging of the containment to the plant vent for dispersion to the environment as allowed by applicable regulations

Included within the scope of the containment ventilation systems are the following subsystems:

- Containment recirculating cooling and filtration system

- Control rod drive mechanism cooling system

- Reactor compartment cooling system

- Refueling water surface and purge system

- Containment auxiliary charcoal filter system

- Containment post-accident charcoal filter system

- Containment shutdown purge system

- Containment mini-purge system

- Penetration cooling subsystem

The principal components of the containment ventilation system include filters, fans, dampers, valves, heat exchangers, essential ductwork, containment isolation valves, and piping.

The containment recirculation fans, control rod drive mechanism fans, and reactor compartment fans are direct-driven units, each with standby units for redundancy. The fans and motors of these units are provided with vibration detecting devices to detect abnormal operating conditions in the early stages of the disturbance. Each of the associated systems is provided with flow switches to verify existence of air flow in the associated duct system. Dampers in the following systems and ducts are provided with air by dual supply air mains, including primary compartment ducts, dome ducts, containment auxiliary charcoal filter systems, butterfly valves which isolate the post-accident charcoal filters, and containment purge supply and exhaust ducts. Two of the four fans and coolers plus one containment spray pump (i.e., one train of each system) are required to provide sufficient capacity to maintain the containment pressure within

design limits after a LOCA or steam line break accident. The containment recirculation fan cooler electrical connections and other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a LOCA.

The control rod drive mechanism cooling system consists of fans and ductwork that draw air through the control rod drive mechanism shroud and eject it to the main containment volume. The reactor compartment cooling system consists of a plenum, cooling coils, fans; and ductwork arranged to supply cool air to the annulus between the reactor vessel and the primary shield and to the nuclear instrumentation external to the reactor.

The refueling water surface and purge system supplies air to the surface of the refueling cavity and exhausts from the area above the refueling manipulator crane to protect the operators during refueling operations. The containment auxiliary charcoal filter system's purpose is to absorb radioactive iodine vapor and radioactive particles that may occur as a result of normal primary system leakage inside the containment.

The containment shutdown purge system is independent of the main auxiliary building exhaust system and includes provisions for both supply and exhaust air. The supply system includes an outside air connection to roughing filters, heating coils, fans, duct system, and supply penetration with a butterfly isolation valve outside containment and a blind flange inside containment. The exhaust system includes an exhaust penetration with a butterfly isolation valve and a blind flange identical to those above, a duct system, a filter bank with high-efficiency particulate air and charcoal filters, fans, and a building exhaust vent. The shutdown purge supply and exhaust duct blind flanges inside the containment are closed during modes 1, 2, 3, and 4.

The containment mini-purge system is capable of purging containment during modes 1 and 2 at a relatively low flow rate (approximately 1500 cfm). The exhaust is through a 6-inch line to the auxiliary building charcoal filters arranged with automatic air-operated butterfly isolation valve inside and outside containment. The isolation valves are capable of closing fully against 60 psig in a maximum of 2 sec after receiving an isolation signal. The mini-purge system is connected to the plant vent and is automatically isolated on high radiation signal.

The containment penetration cooling system is designed to prevent the bulk concrete temperature surrounding the containment penetrations from exceeding 150°F.

2.7.7.2.2 Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems in the containment were evaluated to ensure that the ventilation systems are capable of performing their intended functions under normal EPU modes.

Other evaluations related to the containment ventilation system are addressed in the following LR section:

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR section 2.5.1.3, Pipe Failures
- Evaluation of the control rod drive mechanism cooling system – LR section 2.8.4.1, Functional Design of the Control Rod Drive System
- Electrical equipment qualification - LR section 2.3.1, Environmental Qualification of Electrical Equipment.
- Onsite and offsite electric power systems, including GDC-17 requirements – LR section 2.3.2, Offsite Power System and LR section 2.3.3, AC Onsite Power System
- Protection against turbine missiles and internal missiles is discussed in LR section 2.5.1.2, Missile Protection
- Containment post accident heat removal – LR section 2.6.1, Primary Containment Functional Design
- Radiological consequences analysis – LR section 2.9.7, Radiological Consequences of a Design Basis Loss-of-Coolant Accident
- Impact of containment purge related to normal operational radwaste effluents and associated doses – LR section 2.10.1, Occupational and Public Radiation Doses

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Portions of the containment ventilation system are within the scope of License Renewal. EPU activities do not add any new components nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. Operating the containment ventilation system at EPU conditions does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.7.7.3 Results

The containment ventilation system's ability to provide the required temperature conditions for personnel and equipment in the containment during normal operation was evaluated. The results of the evaluation determined that an increase in the containment bulk air temperature of approximately 1°F from the current observed level will occur at EPU conditions. This increase in the normal operating containment bulk air temperature will not exceed the maximum normal operating bulk temperature limit of 120°F.

The following hot mechanical containment penetrations are cooled by forced air drawn through a common suction damper via the roof by the penetration cooling system:

- Main steam lines
- Feedwater lines
- Steam generator blowdown lines
- Pressurizer steam and liquid space sample lines

As a result of EPU there will be only minor temperature changes in the process fluids contained in these systems. The minor increase in heat loads can be adequately compensated for by the existing automatic temperature controller within the penetration cooling system. Thus, no changes are required for the penetration cooling system as a result of EPU. Refer to LR section 2.5.5.1, Main Steam, LR section 2.5.5.4, Condensate and Feedwater, and LR section 2.1.10, Steam Generator Blowdown System for additional information related to the process systems served.

As a result of EPU there will be only minor temperature changes of less than 1°F in the reactor compartment cooling system air leaving the annulus space around the reactor vessel (based on expected 1600 Btu/h heat gain and 21,175 cfm air flow). The minor increase in heat loads can be adequately compensated for by the existing reactor compartment cooling system. Thus, no changes are required for the reactor compartment cooling system as a result of EPU.

The EPU did not require changes to the containment ventilation isolation system because the containment peak pressure remains below the containment design pressure of 60 psig as a result of EPU; refer to LR section 2.6.1, Primary Containment Functional Design. Therefore, the ability of the containment ventilation isolation system to provide containment isolation is not impacted by EPU.

The containment shutdown purge system is required to provide sufficient air to purge containment for access and discharge the air via the plant vent. The EPU did not impose any changes on the purge system requirements. Therefore, the containment purge system ability to purge for access is not impacted by EPU. Radiation monitors are provided to monitor the releases. Refer to LR section 2.10.1, Occupational and Public Radiation Doses, for the evaluation of the impact on normal releases and the impact on the radiation monitors set-points.

Related to post accident operation, refer to LR section 2.6.1, Primary Containment Functional Design for the system evaluation following an accident.

2.7.7.4 Conclusions

The Ginna review of the containment ventilation system has adequately accounted for the effects of the proposed EPU on the ability of the containment ventilation system to provide a suitable and controlled environment for the containment components. Based on this, Ginna concludes that the containment ventilation system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4, GDC-17, GDC-60 and GDC-61. Therefore, Ginna finds the proposed EPU acceptable with respect to the containment ventilation system.