2.8 Reactor Systems

2.8.1 Fuel System Design

2.8.1.1 Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. Ginna Nuclear Power Plant, LLC (Ginna) reviewed the fuel system to ensure that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences.
- The fuel system damage is never so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

Ginna's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on:

- 10CFR50.46, insofar as it establishes standards for the calculation of emergency core cooling system performance and acceptance criteria for that calculated performance
- 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
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained
- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any loss-of-coolant accident (LOCA)

Specific review criteria are contained in the Standard Review Plan (SRP), Section 4.2 and other guidance provided in Matrix 8 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 safety related structures, systems and components with respect to fuel system design relative to conformance to:

- 10CFR50.46 is described in Ginna UFSAR section 6.3.3 which provides a design evaluation of the Ginna ECCS against 10CFR50.46 criteria.
- GDC-10 is described in Ginna UFSAR section 3.1.2.2.1 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.
- GDC-27 is described in Ginna UFSAR section 3.1.2.3.8, which states that the reactivity control systems in conjunction with boron addition through the Emergency Core Cooling System (ECCS) has the capability of controlling reactivity changes, including the effects of long-term xenon decay and plant cooldown, under postulated accident conditions with appropriate margins for stuck rods.
- GDC-35 is described in Ginna UFSAR section 3.1.2.4.6, which states that the plant design includes an emergency core cooling system (ECCS) which is capable of providing cooling water to the reactor core in response to various postulated accidents at a rate sufficient to maintain the core in a coolable geometry and to ensure that the clad metal-water reaction is limited. The ECCS is further discussed in UFSAR section 6.3.

A review of fuel system design for impact on license renewal evaluations is not necessary since continued applicability of the EPU safety analysis for the 14x14 nine-grid 422V+ fuel assembly will be evaluated or re-analyzed during the reload safety evaluation process for the reload cycles employing this design.

2.8.1.2 Technical Evaluation

2.8.1.2.1 Fuel System Design Features

The licensing basis for the fuel system design is contained in Section 4.2 of the *Ginna Updated Final Safety Analysis Report* (UFSAR). In order to implement the EPU, the Ginna fuel design will be transitioned from the current 14x14 optimized fuel assembly (OFA) design with 0.400-inch diameter rods to the 14x14 422VANTAGE+ (422V+) fuel assembly design with 0.422-inch diameter rods. With the exception of the tube-in-tube guide thimble, increased dimple contact area, and balanced vane pattern (see LR section 2.8.1.2.2.3), the basic features in this design have already operated prior to their introduction into the core at Ginna Station. At both Point Beach and Kewaunee, the 0.422-inch diameter rod was necessary to comply with departure from nucleate boiling (DNB) acceptance criteria with the desired nuclear steam supply system (NSSS) parameters and core peaking factor at the uprated conditions. However, the standard 14x14 422V+ assembly design, as employed previously at Point Beach and Kewaunee, features seven grids, while the current 14x14 OFA design used at Ginna has nine grids.

The key features of the Ginna 14x14 422V+ fuel assembly are as follows:

- 0.422-inch outside diameter fuel rods
- Annular axial blanket pellets
- 143.25-inch pellet stack length
- Standard height removable top nozzle (RTN)
- Reduced rod bow (RRB) Alloy 718 top grid
- OFA style, ZIRLO™ 422V+ mid-grids (low-corrosion ZIRLO™ thin strap)
- Increased dimple-to-rod contact area
- Balanced mixing vane pattern
- High-force Alloy 718 bottom grid
- ZIRLO™ tube-in-tube guide thimble assembly
- Debris-filter bottom nozzle (DFBN)
- Oxide coated clad for debris mitigation
- ZIRLO™ fuel rod clad and instrumentation tubes

For the purposes of the EPU analysis, fuel-related safety and design parameters have been chosen to bound the current 14x14 nine-grid OFA fuel and the upgraded 14x14 nine-grid 422V+ fuel assembly. These bounding parameters have been used in the safety and design analyses discussed in other sections of this report.

The 14x14 422V+ fuel rod has been sized to accommodate a lead rod burnup of up to 75,000 MWD/MTU. The 14x14 422V+ fuel assembly is currently designed to accommodate a peak fuel assembly average burnup of 62,000 MWD/MTU. VANTAGE+ is currently licensed to 60,000 MWD/MTU by the NRC (Reference 6) with extension to 62,000 MWD/MTU on a cycle-specific basis, as delineated in Reference 1, Appendix R.

Figure 2.8.1-1 provides a comparative illustration of the 14x14 OFA (nine-grid) and 422V+ (nine-grid) designs.

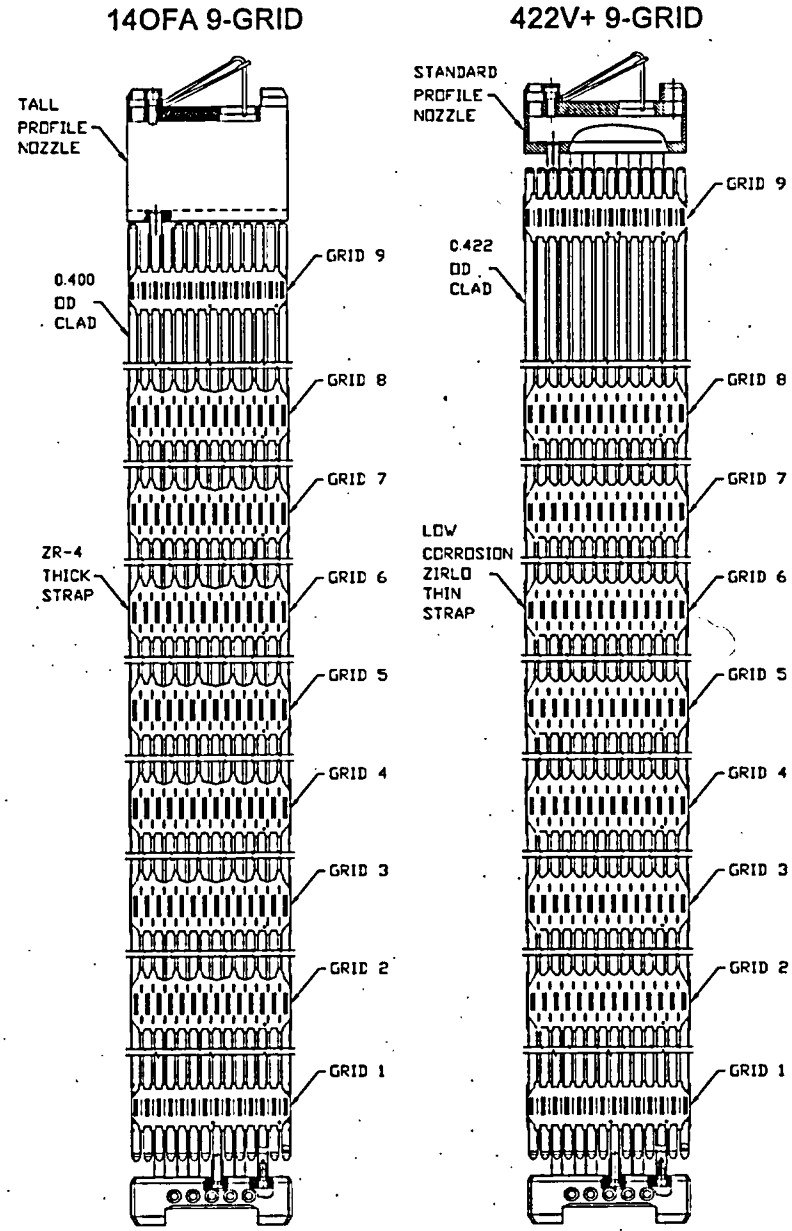

Figure 2.8.1-1

Comparison of the 14x14 OFA (9-grid) and 422V+ (9-grid) designs

2.8.1.2.2 Mechanical Compatibility and Performance

2.8.1.2.2.1 Introduction

The effects of the EPU on the mechanical design are limited to induced changes in the core flow rates and operating temperatures, which have been considered in the supporting calculations. For the mechanical design considerations, the more prominent effects arise from the fuel transition that is coincident with the EPU. The mechanical design evaluation of the 14x14 nine-grid 422V+ fuel assembly is based on the NRC-approved Westinghouse Fuel Criteria Evaluation Process (FCEP) described in WCAP-12488-A (Reference 1). In accordance with the WCAP-12488-A process, the NRC was notified of the design change via Reference 2. Continued applicability of the EPU safety analysis for the 14x14 nine-grid 422V+ fuel assembly will be evaluated or re-analyzed during the reload safety evaluation process for the reload cycles employing this design. While the fuel design changes have been addressed separately via FCEP, Table 2.8.1-1 provides a comparison of the key features of the 14x14 nine-grid OFA and 14x14 nine-grid 422V+ fuel designs for ease of reference.

Table 2.8.1-1
Comparison of Fuel Designs

	OFA (9 grid)	422V+ (9 grid)
Fuel Assembly Overall Length, inch	159.9 (ref)	160.0 (ref)
Fuel Rod Overall Length, inch	149.2 (ref)	152.8 (ref)
Fuel Rod Pitch, inch	0.56	same
Fuel Tube Material	ZIRLO™ (coated bottom)	same
Fuel Tube Clad OD, inch	0.400	0.422
Fuel Rod Clad Thickness, inch	0.024 (ref)	same
Fuel Clad Gap, mil	3.5 (ref) (uncoated pellets)	3.75 (ref) (uncoated pellets)
Enriched Fuel Pellet diameter, inch	0.34 (ref) (uncoated pellets)	0.37 (ref) (uncoated pellets
Enriched Fuel Pellet length, inch	0.41 (ref)	0.44 (ref)
Annular Axial Blanket Pellet diameters	[redacted]	[redacted]
ID, inch	0.17 (ref)	0.18 (ref)
OD, inch	0.34 (ref)	0.37 (ref)
Annular Axial Blanket Pellet length, inch	0.50 (ref)	same
Fuel Stack Height (cold, undensified), inch	141.4 (ref)	143.25 (ref)
Plenum volume (cold, top), inch^3	0.60 (ref)	0.88 (ref)
Guide Thimble Material	ZIRLO™	same
Grid Material, Inner	[redacted]	[redacted]
Mid-grid	Zircaloy-4	ZIRLO™
Nominal (dry) Fuel Assembly weight, lb	1136	1250

Also included in this section is a discussion of the 14x14 422V+ fuel design compatibility with the current 14x14 OFA fuel assembly design during the mixed-core transition. Specifically, <u>LR section 2.8.1.2.2.4</u> discusses the testing performed to substantiate the key parametric values assumed in the safety analyses. The effect of flow redistribution due to hydraulic mismatch of the co-resident fuels is demonstrated for mechanical design (compatibility) considerations via testing. These effects are evaluated separately for other disciplines based on this test data (see LR section 2.8.1.2.2.5 for additional information).

The higher elevation of the top nozzle adapter plate in the 14x14 422V+ fuel assembly raises the rod bottom position for the rod cluster control assemblies. This impacts the nuclear design, the rod position indication system, and reduces the step count to the all rods out position. These issues are addressed in LR section 2.8.2 and LR section 2.8.4.1.

2.8.1.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

In accordance with the WCAP-12488-A process, the various criteria for fuel damage and fuel rod failure, fuel coolability, and nuclear design are screened for impacts based on the known design changes from an established design. Each of the key design changes is then evaluated versus the applicable (screened) criteria. The acceptance criteria evaluated for this design change were:

- Fuel rod clad fretting wear
- LOCA and non-LOCA fuel clad temperatures
- DNB
- Thermal-hydrodynamic stability.

The results of the evaluation were included in the notification letter from Westinghouse to the NRC (Reference 2).

2.8.1.2.2.3 Description of Analyses and Evaluations

For the 14x14 422V+ fuel assembly with nine-grids, a shortened fuel-assembly top nozzle and slightly longer fuel rods will be used. This corresponds to essentially the same fuel assembly used at Point Beach and Kewaunee, but with two additional mid-grids. In 1997, Westinghouse notified the NRC of the introduction of the 14x14 422V+ and the applicability of the WRB-1 DNB correlation to this mid-grid design (Reference 3). The FCEP justified the applicability of the WRB-1 departure from nucleate boiling rate (DNBR) correlation limit of 1.17 for this mid-grid and documented that the design would be used in both Point Beach units and possibly other 14x14 Westinghouse-fueled plants, provided the appropriate analyses were done. This is the same design that is also in use at Kewaunee.

A balanced vane pattern was implemented in the 422V+ Ginna mid-grid design to eliminate a known mechanism for fuel assembly vibration. Balanced vane patterns were developed under the 17x17 Robust Fuel Assembly (RFA) and RFA-2 programs and are also used in the 15x15 Upgrade and Next Generation Fuel (NGF) designs. In addition, a dimple form radii change was also made to the mid-grid for increased formability in manufacturing and the dimple-to-rod contact area was increased for additional fretting margin.

The 17x17 RFA design first introduced a balanced vane pattern, which eliminated abnormal, resonant fuel assembly vibration, while having no impact on the DNB performance of the grid. The objective of the modification was to improve fretting wear, but not significantly affect any other thermal-hydraulic or mechanical performance features of the RFA design.

The revised 14x14 422V+ fuel assembly design will also incorporate the tube-in-tube guide thimble design. The incorporation of the tube-in-tube design will provide additional incomplete rod insertion (IRI) margin to the assembly with negligible design impact.

2.8.1.2.2.4 Testing

While many of the features of the 14x14 422V+ nine-grid design can be appropriately evaluated based on analysis or past experience with similar (or identical) components, some of the features were deemed sufficiently unique to warrant testing. The following is a summary of the testing performed in support of the fuel system evaluation.

The fuel assembly compatibility test system (FACTS) was used to perform hydraulic tests to support the evaluations for the following characteristics of the 14x14 422V+ nine-grid design:

- Guide thimble and instrument tube coolant flow
- Thermal-hydraulic characteristics of grids
- Thermal-hydraulic characteristics for joints and connections
- Top nozzle spring holddown force
- Fuel assembly hydraulic effects
- Fuel assembly hydraulic lift forces
- Fuel assembly thimble bypass flow
- Bulk boiling in thimble
- Fuel assembly crossflow and axial flow profile

A FACTS fuel assembly vibration test was also performed to support the evaluations for the following characteristics of the 14x14 422V+ nine-grid design:

- Requirement for fuel rod support for the grids
- Requirement for fuel rod interfaces for the fuel assembly

A long-term wear test was performed in the VIPER test loop to support the evaluations for the following characteristics of the 14x14 422V+ nine-grid design:

- Grid requirement for fuel rod support
- Fuel assembly axial grid locations
- Fuel assembly requirement for fuel rod interfaces
- Although none of the Westinghouse 14x14 plants have experienced any significant grid-to-rod fretting, the proposed design has been modified to increase the dimple contact area to enhance fretting margin. This is consistent with the Westinghouse effort to enhance grid-to-rod fretting margins for all fuel designs. The 422V+ design, featuring the increased dimple contact area, is described in the FCEP notification (Reference 2). The evaluation of the 422 V+ design features confirms acceptable high frequency vibration for this design.

Several tests of the new mid-grid design were undertaken as follows:

- Mid-grid cell stiffness test
  - Grid requirement for fuel rod support
- Mid-grid thermal relaxation test
  - Grid requirement for fuel rod support
  - Grid thermal relaxation
- Mid-grid dynamic crush test
  - Grid requirement for structural integrity
  - Seismic/LOCA analysis fuel assembly models

In addition, bulge joint loading and strength tests for the tube-in-tube thimble design were performed to address grid requirement for positioning and structural continuity and dimensional stability for joints and connections.

2.8.1.2.2.5 Requests for Additional Information (RAIs)

To facilitate the NRC review, the RAIs applicable to mechanical compatibility and performance that were received in prior power uprating submittals are addressed below for the Ginna EPU specifics.

> "In Section 7.1 of the Application Report, the licensee states the level of fuel rod fretting, oxidation and hydriding of thimbles and grids, fuel rod growth gap, and guide thimble wear was acceptable. Provide a reference to the document which provides the analytical results, and lists the numerical values for these parameters along with their acceptable limit for the SPU conditions. Also, explain how the analysis performed for IP2 SPU conditions met the applicable regulatory criteria and indicate whether the methodology used has been previously approved by the staff."

This RAI discusses several issues as they apply to fuel rods and to fuel assembly structures. All design criteria have been shown to be met and are documented in proprietary calculation notes and test reports that can be made available for audit.

A series of hydraulic tests and analyses were performed by Westinghouse to confirm fuel assembly vibration and fretting performance. Based on these tests and analyses, the 14x14 422V+ nine-grid design has adequate margin.

The fuel assembly structure formerly had a hydriding pickup limit of [ ]a,c. This hydride pickup limit was replaced as indicated in the review and approval by the NRC of WCAP-12488-A, Addendum 1-A, January 2002. The upper bound value is [ ]a,c Maximum grid strap and thimble thinning at the Ginna Station is calculated at EPU conditions to be [ ]a,c, thus the 14x14 422V+ assembly meets this design criterion.

The Westinghouse criteria for fuel rods are [ ]a,c for clad hydriding, and [ ]a,c for clad oxide steady-state interface temperature. All design criteria have been shown to be met and are documented in proprietary calculation notes that are available for audit. These criteria were approved by NRC in WCAP-12610-P-A, which is applicable to the ZIRLO™ cladding used on the 14x14 422V+ design.

The space between the fuel rod end plugs and the fuel assembly nozzles must be sufficient to prevent interference of these members. All aspects of the 14x14 422V+ nine-grid design that affect this requirement are similar to the 14x14 422V+ seven-grid design features currently in the Point Beach and Kewaunee cores and have been shown to be acceptable. These criteria were approved by NRC in WCAP-12610-P-A, which is applicable to the ZIRLO™ cladding used on the 14x14 422V+ nine-grid design.

The Westinghouse design bases and criteria for guide thimble wear are that no localized perforation of the tube wall should occur and the integrity of the guide thimble tube should be maintained throughout the normal life of a fuel assembly. These criteria were approved by NRC in WCAP-12610-P-A, which is applicable to the ZIRLO™ guide thimble tube used on the 14x14 422V+ design. The tube wall thickness, material, initial clearances, and thimble bypass flow do not differ significantly between the 14x14 422V+ nine-grid and 14x14 OFA Ginna fuel assembly designs. Thus, no changes are expected in the guide thimble wear performance for the EPU.

> "In Section 7.1 of the Application Report, the licensee states that analyses verified the fuel assembly holddown spring's capability to maintain contact between the fuel assembly and the lower core plate at normal operating conditions for the SPU. Describe the analyses performed to justify this statement. Additionally, provide the numerical values that show the design criteria are met."

The fuel assembly holddown spring analysis was performed on the 14x14 422V+ nine-grid assembly using the same standard holddown spring methodology approved in WCAP-12488. The analysis that was completed evaluates the net holddown force on the fuel assembly throughout its design lifetime, taking into account fuel assembly growth and spring relaxation on a cycle-by-cycle basis. The analysis accounts for the opposing forces that act on each fuel assembly due to assembly weight, buoyancy, spring forces, and lift force. The analysis ensures that there is a positive net fuel assembly holddown force on the bottom core plate at all times except during a pump over-speed at hot conditions. During a postulated pump over-speed event, the assembly holddown force acceptance criterion allows assemblies to lift off the lower core plate but not enough to plastically deform the holddown spring during the event. This criterion is satisfied for the 14x14 422V+ nine-grid fuel assembly design under the Ginna Station EPU conditions.

The holddown spring for the 14x14 422V+ nine-grid design satisfies all of the standard fuel assembly holddown spring requirements and provides [ ]^{a,c} holddown during normal operation.

2.8.1.2.2.6 Mechanical Compatibility and Performance Results

The changes associated with the revised 14x14 422V+ design were reviewed in accordance with the WCAP-12488-A (Reference 1) process and found to be acceptable with respect to mechanical design.

The results of the testing have confirmed that all applicable criteria for the EPU to 1775 MWt are satisfied for the 14x14 422V+ nine-grid design with the exception of crossflow effects on DNB margins or LOCA margins. These results and conclusions are addressed in the respective sections on thermal-hydraulic analyses (see LR section 2.8.3, Thermal and Hydraulic Design) and LOCA analyses (see LR section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents). Values for other key fuel system parameters for the Ginna EPU have been incorporated into the affected sections of this report.

The results of these evaluations support the EPU in that the effects of the EPU on core flows and operating temperatures have been considered for the Ginna application. With acceptable affirmation for DNB and LOCA crossflow effects (LR section 2.8.3, Thermal and Hydraulic Design and LR section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents), the evaluation of the 14x14 422V+ nine-grid and OFA nine-grid design differences concludes that the two designs are mechanically compatible with each other. The effects of the EPU with regard to the regulatory bases identified in LR section 2.8.1.1 are addressed for the seismic/LOCA LR section 2.8.1.2.3) and fuel performance (LR section 2.8.1.2.4) portions of the fuel system evaluation.

2.8.1.2.3 Seismic/LOCA

2.8.1.2.3.1 Introduction

The current licensing basis for Seismic/LOCA is based on analyses done at the time of steam generator replacement as identified in Section 4.2 of the Ginna UFSAR. An evaluation of 14x14 422V+nine-grid fuel assembly structural integrity has been performed with consideration given to the lateral effects of two LOCA auxiliary line breaks (pressurizer spray and upper plenum injection lines) and a safe shutdown earthquake (SSE) seismic event.

2.8.1.2.3.2 Input Parameters, Assumptions, and Acceptance Criteria

The analysis parameters—the natural frequencies, mode shapes, and span masses of the fuel assembly combined with the structural damping—were used to generate a simplified lumped-mass-spring fuel assembly model. The mid-grid crush strength, stiffness, and damping, the fuel assembly impact stiffness and damping, the number of fuel assemblies, and the gap clearances between fuel-assemblies and at the baffles were used to generate the reactor internal model. The WEGAP computer code was used (Reference 4).

The 422V+ features that are different from the OFA design and impact the Seismic/LOCA analysis are as follows:

- Top grid elevation
- Fuel rod diameter
- Tube-in-tube guide thimbles
- Fuel assembly guide thimble length (driven by overall F/A length increase of 0.04 inches and top nozzle height difference)
- Mid-grid design (strap thickness, cell thickness, and crush strength)

The remaining differences between the OFA and 422V+ fuel assembly designs had insignificant effects on the Seismic/LOCA analysis.

The mid-grid of the 14x14 422V+ Ginna design is the same as the 14x14 422V+ seven-grid design except for a balanced mixing vane pattern, increased dimple-to-rod contact area, and increased dimple form radii. The dynamic impact capability of the revised design has been determined by testing and analysis and has been shown to have sufficient margin under accident conditions for equilibrium and transition cores.

The acceptance criteria for the seismic loading design are that fragmentation of the fuel rod must not occur as a result of the seismic loads, and control rod insertability and coolable geometry must be maintained.

The principal acceptance criteria for a LOCA event are that fragmentation of the fuel rod must not occur as a direct result of the blowdown load, and control rod insertability and coolable geometry must be maintained.

The grid crush strength is established based on analysis of the 95% confidence level on the true mean of the test data at operating temperature.

2.8.1.2.3.3 Description of Analyses and Evaluations

A homogenous core of 14x14 422V+ design and two limiting mixed cores (the 14x14 422V+ and 14x14 OFA 9-grid fuel assembly designs) were evaluated.

The maximum horizontal input motion congruent with the core principal axis was used to determine dynamic fuel responses. The reactor core was analyzed as a de-coupled system with respect to the two lateral directions. The input forcing function was obtained from a separate reactor pressure vessel and reactor internals system analysis.

Based on appropriate modeling, it has been shown that the assumed mode shapes agree well with the predominant fuel assembly vibration frequencies. With the appropriate analysis parameters, the WEGAP reactor core model was used for analyzing transient loadings. The original methodology as defined in Reference 4 has not changed.

The results of the combined LOCA and SSE analysis were obtained using the time-history numerical integration technique. The maximum grid impact forces obtained from both transients were combined using the square root of the sum of squares (SRSS) method. The maximum loads were compared with the allowable grid crush strength.

In the grid load analysis, the time-history motions of the barrel at the upper core plate elevation and the upper and lower core plates were applied simultaneously to the reactor core model.

The time histories representing the SSE motion and the pipe rupture transients were obtained from the time history analyses of the reactor vessel and internals finite element model.

2.8.1.2.3.4 RAIs

To facilitate the NRC review, the RAIs applicable to seismic/LOCA analysis that were received in prior power uprating submittals are addressed below for the Ginna EPU specifics.

> "State whether the core is being treated as a mixed core during the transition cycles. Also, explain how fuel damage was analyzed in a seismic event for the mixed core as it transitions to a homogeneous 15x15 Upgraded fuel loading and describe the worst case scenario analyzed. In addition, provide the technical justification that shows structural integrity at the SPU condition for the mixed core is maintained in a loss-of-coolant accident (LOCA) coincident with a seismic event at IP2."

The licensing basis for fuel structural integrity requires that the loading conditions address seismic loading, LOCA loading, and the combination of LOCA and seismic loading as required by the NRC. The seismic and LOCA analysis of the reactor pressure vessel system was performed for the EPU conditions, including the generation of the core plate seismic motions that were used in the Ginna Station analysis of 14x14 422V+ nine-grid and 14x14 OFA 9-grid fuel assembly designs. The LOCA analysis used LOCA hydraulic forcing functions calculated using the MULTIFLEX computer code and crediting leak-before-break (LBB) for the reactor coolant loop piping.

Detailed site-specific fuel assembly analyses for Ginna Station have been performed under EPU conditions in accordance with approved methodologies. These methodologies were approved by NRC in WCAP 9401-P-A (Reference 5), WCAP-9500-A (Reference 4), WCAP-12610-P-A (Reference 6), and WCAP-12488-A (Reference 1). Results from these analyses demonstrate that for the limiting-loading condition (combined seismic and LOCA loading), the fuel assembly structural integrity is maintained and the grid impact loads and component stresses remain below the allowable limits. Therefore, the requirements to maintain a coolable core geometry are met. These analyses were performed for homogenous cores of 14x14 422V+ nine-grid fuel and transition cores with both 14x14 422V+ nine-grid fuel and 14x14 OFA nine-grid (current resident) fuel. The transition core analyses considered various fuel assembly loading combinations to determine the limiting conditions. The transition core-loading pattern that is limiting for the upgrade fuel occurs when the 14x14 422V+ nine-grid fuel is located at [ ][a,c] and the 14x14 OFA nine-grid fuel is located at [ ][a,c]. The transition core loading pattern that is limiting for the 14x14 OFA nine-grid fuel occurs when the 14x14 OFA nine-grid fuel is located at [ ][a,c] and the 14x14 422V+ nine-grid fuel is located at [ ][a,c]. In both limiting cases, significant margins remain for both the 14x14 422V+ nine-grid and 14x14 OFA nine-grid fuel assemblies, considering combined seismic and LOCA loading.

The maximum calculated load for the combined seismic and LOCA loads was compared to the maximum load that can be applied before plastic deformation occurs in the subject grid (called the allowable limit in the analysis). In all cases the postulated load was well below the allowable limit. The closest ratio of combined seismic and LOCA loading to limit load is [ ]^{a,c}. For thimble tubes and fuel rods, there is no case for which the strength of the thimble tubes and fuel rods is not at least [ ]^{a,c} the calculated loading for the combined seismic- and LOCA-loading condition. Because none of the fuel assembly components will experience loading at or above their strength limit, the fuel assembly geometry is maintained for this limiting loading combination and the coolable geometry conclusions of the LOCA ECCS analyses are not affected. The approval of the methodology is discussed in the following RAI. The mixed core configuration resulted in the limiting loads for all loading conditions and had significant margin.

> "In the Fuel Criterion Evaluation Process (FCEP) Notification of the 15x15 Upgrade Designs submitted by Westinghouse Electric Company to the NRC on February 6, 2004, Westinghouse states that evaluations of the 15x15 Upgraded fuel assembly design for seismic and LOCA loading at IP2 have been performed in accordance with the "Reference Core Report 17x17 Optimized Fuel Assembly" methodology. Provide the technical justification showing that the 17x17 design/method referenced is applicable to the 15x15 fuel design."

In Section 3.0, Category B, Item e, "Fuel Assembly Structural response to Seismic/LOCA Loads" of the FCEP, notification to the NRC regarding the 14x14 422V+ nine-grid design, Westinghouse states: "Evaluations of the revised 14x14 422V+ design for seismic and LOCA loading has been performed in accordance with approved methodologies[3]."

The indicated Reference 3 cites:

> Reference 3. Davidson, S. L and Iorii, J. A. (Eds), et al., *Reference Core Report 17x17 Optimized Fuel Assembly*, WCAP-9500-A, May 1982; Beaumont, M. D. and Skaritka, J. (Eds.), et al., *Verification testing and Analysis of the 17x17 Optimized Fuel Assembly*, WCAP-9401-P-A March 1979; and Davidson, S. L, and Iorii, J.A (Eds.), et al., *Supplemental Acceptance Information for NRC Approved Version of WCAP-9401/9402 and WCAP-9500*, February 1983.

The references cited were approved by the NRC for the intended application in WCAP-12488–P-A (Reference 1). On page 5.3 of the SER/TER under 5.4 "Fuel Assembly Structural Damage from External Forces Evaluation," it states: "Generic analysis methods for performing combined LOCA-seismic loading analysis have been described by W in WCAP 9401-P-A (and WCAP-9402-A). These analysis methods not only include the fuel assembly structural response, but also fuel rod cladding loads. These methods have been approved by the NRC and therefore, PNL concludes they remain acceptable for application to W fuel design changes."

In the SER for WCAP-9500-A (Reference 4) and WCAP-9401-P-A (Reference 5), the NRC discusses the generic analysis methodology used to evaluate the 17x17 OFA. The methodology essentially consisted of four mathematical models: a system model, a detailed core model, a lateral fuel assembly model, and an axial fuel assembly model. Details of the methodology are described in WCAP-9401-P-A.

In the NRC's SER approval for WCAP-9500-A, the following statement was made:

> "The methodology described applies not only to three- and four-loop 17x17 plants but generically for plants having other standard arrays (e.g., 14x14, 15x15 and 16x16)."

This methodology was captured in Chapter 18 of WCAP-9500-A (Reference 4), and included seismic and LOCA loads. The methodology was further described in WCAP-9401-P-A (Reference 5). For each fuel transition, the "new" design was compared to the previous design. For the analysis of the combined seismic and LOCA loads, there has been no change that would invalidate the original methodology that was shown and stated to be applicable to all Westinghouse fuel arrays.

WCAP-12488-A (Reference 1), Fuel Criteria Evaluation Process, is not limited to any specific fuel design or geometry, and has been in use since March 1993 based on the NRC approval of this methodology for evaluating Westinghouse fuel changes. Westinghouse has followed the methodology described and approved for seismic and LOCA analysis. While the methodology used is the same as that referenced in WCAP-9500/WCAP-9401, separate calculations and evaluations were conducted for Ginna Station based on EPU conditions.

2.8.1.2.3.5 Seismic/LOCA Results

The maximum SSE and LOCA results for the 14x14 422V+ fuel assembly in both a homogenous core and the mixed transition cores occur in the Z-direction during SSE loading. The maximum structural grid loads for the 14x14 422V+ fuel assemblies occurred in the peripheral assemblies in the three fuel assembly arrays. The maximum fuel assembly deflection occurred in an assembly array consisting of 13 fuel assemblies in the X-direction during a seismic loading.

The maximum grid loads obtained from SSE and LOCA loading analyses were combined using the SRSS method. The results of the combined seismic and LOCA analyses indicate that the maximum impact forces for the 14x14 422V+ nine-grid assembly design using the two-direction grid characteristics are less than the respective allowable grid strengths. The allowable grid strengths are established at the 95% confidence level on the true mean from the distribution of experimentally determined grid crush data at temperature. Based on the results of the combined SSE and LOCA loads, the 14x14 422V+ fuel nine-grid assembly design is structurally acceptable for Ginna. Core coolable geometry requirements are met.

Fuel assembly displacement is limited by the total accumulated gap clearances, plus elastic grid deformations. Fuel assembly stresses were calculated based on the most limiting case. The stresses for the fuel rods and thimble tubes were calculated based on the maximum lateral displacement, the vertical impact load, and operating condition loads. The results indicate that adequate margins for both fuel rods and thimble tubes exist, so that fragmentation of fuel rods will not occur. The reactor can be safely shut down under faulted-condition loading. In conclusion, the 14x14 422V+ assembly design is structurally acceptable under the combined seismic and LOCA loadings for the Ginna Station.

The 14x14 422V+ fuel assembly is structurally comparable to that of the OFA assemblies used in previous cycles. The evaluation of the 14x14 422V+ Ginna fuel assembly in accordance with NRC requirements as given in SRP Section 4.2, Appendix A (28), shows that the 14x14 422V+ nine-grid fuel is structurally acceptable for the Ginna reactor. The grid loads evaluated for the LOCA and seismic events, and combined by the SRSS method identified in SRP Section 4.2 are less than the allowable limit. The same conclusion is true for a transition core composed of both 14x14 422V+ nine-grid and OFA nine-grid fuel assemblies. Therefore, coolable core geometry is maintained. The stresses in the 14x14 422V+ Ginna fuel assembly components resulting from seismic and LOCA-induced deflections are within acceptable limits.

The 14x14 OFA fuel assembly was also evaluated. This evaluation concluded that the stresses of the fuel rod and thimble tube are structurally acceptable under the combined seismic and LOCA loadings for Ginna at EPU conditions. The reactor can be safely shut down under the combined faulted-condition loads.

2.8.1.2.4 Fuel Rod Performance

2.8.1.2.4.1 Introduction

Fuel rod performance for all Ginna fuel is shown to satisfy the NRC SRP fuel rod design bases on a region-by-region basis. These same bases are applicable to all fuel rod designs, including the Westinghouse OFA and 14x14 422V+ fuel designs. The design bases for Westinghouse 14x14 422V+ fuel are discussed in Reference 5. The current licensing basis is described in Section 4.2 of the Ginna UFSAR and is based on the same methods and models (PAD 4.0) used here. This analysis is based on this licensing basis analysis incorporating the transition to 14x14 422V+ fuel design and the bounding high-temperature nuclear design cases representing three cycles at EPU conditions (two transition cycles and one equilibrium cycle) developed for the Nuclear Design (see LR section 2.8.2, Nuclear design). Compliance with the GDC-10 SAFDL criteria for reload cycles is confirmed via the approved reload methodology of WCAP-9273-NP-A (Reference 7).

2.8.1.2.4.2 Input Parameters, Assumptions, and Acceptance Criteria

The fuel rod design analysis is performed on a cycle-specific basis. The reference analysis presented here is based on the bounding high-temperature nuclear design cases representing three cycles at EPU conditions (two transition cycles and one equilibrium cycle) developed for the Nuclear Design (see LR section 2.8.2, Nuclear Design). Both the reference analysis and the cycle-specific analysis consider compliance for all fuel designs in the core. Therefore, there is no impact due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles, since this configuration is explicitly evaluated. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions (for example, the presence of annular pellets in axial blankets or changes in the fuel rod diameter and plenum length). Analysis of integral fuel burnable absorber (IFBA) rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the gas release from the ZrB_2 coating.

Ginna has elected to implement the Relaxed Axial Offset Control (RAOC) methodology coincident with the uprate and the first transition cycle with 14x14 422V+ fuel. The license amendment request for this (Reference 11) was submitted separately. The RAOC methodology provides additional operational margin by reducing analytical margin and is reflected in the reference analysis presented here.

Fuel rod design evaluations for the 14x14 422V+ fuel were performed using NRC-approved models (References 5 and 8) and NRC-approved design criteria methods (References 9 and 10) to demonstrate that all fuel rod design criteria are satisfied.

The fuel rod design criteria given below are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod that has the minimum margin to the design limit. In general, no single rod is limiting with respect to all the design criteria. Generic evaluations alone cannot identify which rods are most likely to be limiting for each criterion, so an exhaustive screening of fuel rod power histories and fuel rods was used to determine the limiting rods. The changes from the current 14x14 OFA design to the 14x14 422V+ design that are important to the Fuel Rod Design analysis reported in this section are:

- Plenum Length
- Fuel stack length
- Pellet diameter
- Spring design
- Clad OD/ID
- IFBA coating length

The NRC-approved PAD 4.0 code, with NRC-approved models (References 5 and 8) for in-reactor behavior, is used to calculate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

PAD 4.0 is a best-estimate fuel rod performance model, and in most cases the design criterion evaluations are based on a best-estimate plus uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties for some critical inputs, such as fuel pellet diameter, can be used in lieu of the fabrication uncertainties.

An evaluation of the clad and structural component oxidation and hydriding was also performed.

The criteria pertinent to the fuel rod design were:

Rod Internal Pressure

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady-state operation, and extensive DNB propagation to occur.

Clad Stress and Strain

The design limit for clad stress is that the volume-averaged effective stress, considering interference due to uniform cylindrical pellet-to-clad contact caused by pellet thermal expansion, pellet swelling, uniform clad creep, and pressure differences between the rod internal pressure and the system coolant pressure, be less than the clad-yield strength for Condition I and II events. While the clad has some capability for accommodating plastic strain, the yield stress has been established as the conservative design limit. The design limit for clad strain during steady-state operation is that the total plastic tensile creep strain due to uniform clad creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1% from the unirradiated condition. The design limit for fuel-rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value. These limits are consistent with proven practice.

Clad Oxidation and Hydriding

The design criteria related to clad corrosion require that the Zircaloy-4/ZIRLO™ clad metal-oxide interface temperature is maintained below specified limits to prevent a condition of accelerated oxidation, which would lead to clad failure.

The best-estimate hydrogen pickup level in Zircaloy-4/ZIRLO™ cladding and structural components is less than or equal to the limit on a volume-averaged basis at EOL.

Fuel Temperature

For Condition I and II events, the reactor protection system is designed to ensure that the fuel centerline temperature does not exceed the fuel melt temperature criterion. The intent of this criterion is to avoid a condition of gross fuel melting that can result in severe duty on the clad. The concern here is based on the large volume increase associated with the phase change in the fuel, and the potential for loss of clad integrity as a result of molten fuel/clad interaction.

Clad Fatigue

The fuel rod design criterion for clad fatigue requires that, for a given strain range, the number of strain fatigue cycles is less than that required for failure, with factors of safety of 2.0 on the stress amplitude and 20.0 on the number of cycles. This criterion addresses concerns about the cumulative effect of short-term cyclic clad stress and strain resulting from daily load follow operation.

Clad Flattening

The clad flattening criterion prevents fuel rod failures due to long-term creep collapse of the fuel rod clad into axial gaps formed within the fuel stack. Current fuel rod designs employing fuel with improved in-pile stability provides adequate assurance that axial gaps large enough to allow clad flattening will not form within the fuel stack.

Fuel Rod Axial Growth

This criterion ensures that there is sufficient axial space to accommodate the maximum expected fuel rod growth. Fuel rods are designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly skeleton to preclude interference of these members.

Plenum Clad Support

This criterion ensures that the fuel clad in the plenum region of the fuel rod will not collapse during normal operating conditions, nor distort so as to degrade fuel rod performance.

Clad Free-Standing

The clad free-standing criterion requires that the clad is short-term, free-standing at beginning of life (BOL), at power, and during hot hydrostatic testing. This criterion precludes the instantaneous collapse of the clad onto the fuel pellet caused by the pressure differential that exists across the clad wall.

These criteria are verified at Ginna EPU-specific operating conditions and fuel rod duties.. The continued validity of the limiting power shapes used in this analysis is confirmed for each reload using Relaxed Axial Offset Control (RAOC) methods.

2.8.1.2.4.3 Description of Analyses and Evaluations

Rod Internal Pressure

The Rod Internal Pressure "no gap reopening" criterion for the Ginna fuel rods has been evaluated at EPU conditions by modeling the gas inventories, gas temperature, and rod internal volumes throughout the life of the limiting rod. The resulting rod internal pressure is compared to the design limit on a case-by-case basis of current operating conditions to EOL. This evaluation showed that the "no gap reopening" criterion is met.

The second part of the rod internal pressure design basis precludes extensive DNB propagation and associated fuel failure. The basis for this criterion is that no significant additional fuel failures, due to DNB propagation, will occur in cores that have fuel rods operating with rod internal pressure in excess of system pressure. The design limit for Condition II events is that DNB propagation is not extensive, that is, the process is shown to be self-limiting and the number of additional rods in DNB due to propagation is relatively small. For Condition III/IV events, it is shown that the total number of rods in DNB, including propagation effects, is consistent with the assumptions used in radiological dose calculations for the event under consideration.

Clad Stress and Strain

Clad temperature and irradiation effects on yield strength were considered in the analysis. The clad stress criterion has been shown to meet the design limits by use of a statistical method which takes into account many uncertainties. Transient clad strain is met based on the clad stress results as described in the previous section. Steady-state clad strain is met by using a Ginna EPU-specific calculation.

Clad Oxidation and Hydriding

The clad surface temperatures were evaluated and satisfied the applicable temperature limits. The base metal wastage of the Zircaloy-4 and ZIRLO™ grids and guide tubes was shown not to exceed the design limit at EOL.

The hydrogen pickup criterion, which limits the loss of ductility due to hydrogen embrittlement which occurs upon the formation of zirconium hydride platelets, has been met with the current approved model for Ginna EPU.

Fuel Temperature

The temperature of the fuel pellets was evaluated by modeling the fuel rod geometry, thermal properties, heat fluxes, and temperature differences in order to calculate fuel surface, average and centerline temperatures of the fuel pellets.

Fuel temperatures have been calculated as a function of local power and burnup. The fuel surface and average temperatures with associated rod internal pressure are provided to transient analysis and LOCA for accident analysis of the 14x14 422V+ nine-grid fuel design. The fuel centerline temperatures are used to show that fuel melt will not occur. For 14x14 422V+ design, the local linear power that precludes fuel centerline melting is 22.70 kW/ft.

Clad Fatigue

Clad fatigue for the 14x14 422V+ nine-grid fuel was evaluated by using a limiting fatigue duty cycle consisting of daily load follow maneuvers. The 14x14 422V+ fuel rod fatigue evaluation, based on a statistical method which takes into account many uncertainties, showed that the cumulative fatigue usage factor is less than the design limit of 1.0.

Clad Flattening

The NRC has approved WCAP-13589-A (Reference 10), which provided data to confirm that significant axial gaps in the fuel column due to densification (and therefore clad flattening) will not occur in current Westinghouse fuel designs. The Ginna fuel meets the criteria for applying the Reference 10 methodology and, therefore, clad flattening will not occur.

Fuel Rod Axial Growth

The Ginna EPU fuel rod growth evaluation, based on similar designs, demonstrates that there is adequate margin to the fuel rod growth design limit for the 14x14 422V+ fuel.

Plenum Clad Support

The helical coil spring used in the 14x14 422V+ fuel design for the Ginna EPU has been shown to provide enough support to prevent potential clad collapse. Therefore, the plenum clad support criterion is met for the 14x14 422 V+ fuel.

Clad-Free Standing

Evaluations of the clad-free standing criteria have shown that instantaneous collapse of the Ginna fuel will be precluded for differential pressures well in excess of the maximum expected differential pressure across the clad under operating conditions. This generic analysis has been shown to be met for all Westinghouse fuel rod geometries.

Fuel rod design evaluations for Ginna Station were performed using the NRC approved models in References 5 and 8 to demonstrate that the SRP fuel rod design criteria are satisfied. For the 14x14 422V+ fuel design, these criteria have been shown to be met.

2.8.1.2.4.4 RAIs

To facilitate the NRC review, the requests for additional information applicable to fuel rod performance that were received in prior power uprating submittals are addressed below for the Ginna EPU specifics.

> "With respect to the impacts of the proposed power uprate on the nuclear, thermal-hydraulic and fuel rod design analyses, please provide a listing of the NRC-approved codes and methodologies used for the design analyses discussed in Section 7.10 of the Attachment III of the submittal and confirm that all parameters and assumptions to be used for analyses described in Sections 7.10 of the Attachment III remain within any code limitations or restrictions."

The fuel rod design code and methodology used for the Ginna EPU analyses was previously approved by the NRC (Reference 8).

2.8.1.2.4.5 Fuel System Design Results

Fuel performance evaluations have been completed for each fuel region to demonstrate that the design criteria can be satisfied for all fuel rod types in the core under the planned operating conditions of a power uprating to 1775 MWt. Based on input from core design, the fuel rod design was analyzed with an $F_{\Delta H}^{N}$ limit of 1.72 for the 14x14 422V+ fuel, while no credit was assumed in this analysis for a reduction in the current $F_{\Delta H}^{N}$ limit of 1.75 for OFA fuel (see LR section 2.8.3, Thermal and Hydraulic Design). Any additional changes from the plant operating conditions originally evaluated for the mechanical design of a fuel region will be addressed for all affected fuel regions as part of the reload safety evaluation process when the plant changes are to be implemented.

As expected, the large increase in power will have a significant impact on the fuel rod design margin, although some of this reduction in design margin is a result of the decision to implement Relaxed Axial Offset Control (RAOC) in order to increase plant operating margins. Rod internal pressure (RIP) criteria, including "no gap reopening" and DNB propagation, clad corrosion, clad stress, steady-state clad strain, clad fatigue, and fuel temperature criteria have all had a significant loss of margin. Although margin is significantly reduced for the aforementioned criteria, all limits were still met in the initial bounding analyses without mechanical changes with the exception of the RIP "no gap reopening" criterion. To counteract the large loss of RIP "no gap reopening" margin, the plenum length will be increased by 0.2 inches, which increases plenum volume and thereby decreases the RIP. With the change in plenum length, analyses confirm that the RIP "no gap reopening" criterion continues to be met. Continued compliance with fuel rod design criteria for both the 14x14 OFA and 14x14 422V+ fuel, including transition core effects, will be confirmed for each reload cycle core design.

Each of these key fuel rod design criteria has been evaluated for application of the Westinghouse 14x14 422V+ fuel assembly design in the Ginna Station. Based on these evaluations, it is concluded that each design criterion can be satisfied through transition cycles to a full core of the 14x14 422V+ design while appropriately accounting for EPU conditions.

2.8.1.3 Fuel System Design References

1. WCAP-12488-A, Davidson, S. L., *Westinghouse Fuel Criteria Evaluation Process*, October 1994, and WCAP-12488-A, Addendum 1-A, Rev. 1, *Revisions to Design Criteria*, January 2002.

2. LTR-NRC-05-34, "Fuel Criterion Evaluation Process (FCEP) Notification of revision to 14x14 422 VANTAGE + Design (Proprietary/Non-proprietary)," June 6, 2005.

3. CAW-97-1166, transmitted via WEPCO (NPL 97-0538) to Document Control Desk (NRC), *14x14, 0.422" OD VANTAGE + (422V+) Fuel Design, Application for Point Beach Units 1 & 2*, September 9, 1997

4. WCAP-9500-A, Davidson, S. L., et al., *Reference Core Report 17x17 Optimized Fuel Assembly*, May 1982.

5. WCAP-9401-P-A, Davidson, S. L,. et al, *Verification and Testing Analyses of the 17x17 Optimized Fuel Assembly*, August 1981

6. WCAP-12610-P-A, Davidson, S. L., et al., *VANTAGE + Fuel Assembly Reference Core Report*, April 1995.

7. *Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.*

8. WCAP-15063-P-A, Rev. 1 with Errata (Proprietary), Foster, Sidener, and Slagle, *Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)*, July 2000.

9. WCAP-10125-P-A (Proprietary), Davidson, S. L., et al., *Extended Burnup Evaluation of Westinghouse Fuel*, December 1985.

10. WCAP-13589-A, Kersting, P. J., et al., *Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel*, March 1995.

11. Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), License Amendment Request Regarding Adoption of Relaxed Axial Offset Control (RAOC), dated April 29, 2005.

2.8.1.4 Conclusion

The Ginna staff has reviewed the analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The Ginna staff concludes that the analyses have adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that

- The fuel system will not be damaged as a result of normal operation and anticipated operational occurrences.
- The fuel system damage will never be so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures will not be underestimated for postulated accidents.
- Coolability will always be maintained.

Based on this, is the Ginna staff concludes that the fuel system and associated analyses will continue to meet the Ginna Station current licensing basis with respect to the requirements of 10CFR50.46, GDC-10, GDC-27, and GDC-35 following implementation of the EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

2.8.2.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core for EPU conditions to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core. The Ginna review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation.

The NRC's acceptance criteria are based on:

- 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
- GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity
- GDC-12, insofar as it requires that the reactor core be designed to ensure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed
- GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges
- GDC-20, insofar as it requires that the protection system be designed to automatically initiate the reactivity control systems to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to automatically initiate operation of systems and components important-to-safety under accident conditions
- GDC-25, insofar as it requires that the protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems

- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained

- GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core

Specific review criteria are contained in SRP, Section 4.3 and other guidance provided in Matrix 8 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 safety related structures, systems and components with respect to nuclear design relative to conformance to:

- GDC-10 is described in Ginna UFSAR section 3.1.2.2.1 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.

- GDC-11 is described in Ginna UFSAR section 3.1.2.2.2 which states that the reactor core and associated coolant systems have been designed so that in the power operating range the net effect of the prompt nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. Specifically, the moderator temperature coefficient is usually, though not always, negative, and the overall power coefficient is negative and so provides a nuclear feedback characteristic to limit a rapid increase in reactivity.

- GDC-12 is described in Ginna UFSAR section 3.1.1.2.3 which states that the reactor core and the associated coolant, control, and protection systems, and operating strategies have been designed to prevent or easily suppress power oscillations that could result in exceeding fuel design limits.

- GDC-13 is described in Ginna UFSAR section 3.1.2.2.4 which states that 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.

- GDC-20 is described in the Ginna UFSAR section 3.1.2.3.1 which states that a protection system 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-25 is described in the Ginna UFSAR section 3.1.2.3.6 which states that the Reactor Trip System (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-26 is described in the Ginna UFSAR section 3.1.2.3.7, which states that two independent reactivity control systems of different design principles are provided. One of the two reactivity control systems employs control rod drive mechanisms to regulate the position of silver-indium-cadmium neutron absorbers within the reactor core. The control rods are designed to shut down the reactor with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The other reactivity control system employs the chemical and volume control system to regulate the concentration of boric acid neutron absorber in the reactor coolant system. The chemical and volume control system is capable of controlling the reactivity change resulting from planned normal power changes.

- GDC-27 is described in Ginna UFSAR section 3.1.2.3.8, which states that the reactivity control systems in conjunction with boron addition through the Emergency Core Cooling System (ECCS) has the capability of controlling reactivity changes, including the effects of long-term xenon decay and plant cooldown, under postulated accident conditions with appropriate margins for stuck rods.

- GDC-28 is described in Ginna UFSAR section 3.1.2.3.9, which states that the maximum reactivity limits of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree which could impair the effectiveness of emergency core cooling.

A review of fuel system design for impact on license renewal evaluations is not necessary since continued applicability of the EPU safety analysis for the 14x14 nine-grid 422V+ fuel assembly will be evaluated or re-analyzed during the reload safety evaluation process for the reload cycles employing this design. The reload design methodology includes the evaluation of the reload core key safety parameters which comprise the nuclear design-dependent input to the UFSAR safety evaluation for each reload cycle.

2.8.2.2 Technical Evaluation

2.8.2.2.1 Introduction

The licensing basis for the reload core nuclear design is defined in Section 4.3 of the Ginna UFSAR. The purpose of the core analysis is to determine prior to the cycle-specific reload design if the previously used values for the key safety parameters remain applicable for the transition to 14x14 422V+ fuel and plant uprating. This will allow the majority of any safety analysis re-evaluations/re-analyses to be completed prior to the cycle specific design analysis. The effects of transitioning to the 14x14 422V+ fuel features and extended power uprate (EPU) conditions on the nuclear design bases and methodologies for Ginna are evaluated in this section.

2.8.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The key features of the Ginna 14X14 422V+ fuel assembly are as follows:

- 0.422 inch outside diameter fuel rods
- Annular axial blanket pellets
- 143.25 inch pellet stack length
- Standard height Removable Top Nozzle (RTN)
- Reduced Rod Bow (RRB) Alloy 718 Top Grid
- OFA style, 422V+ Mid-grids with balanced mixing vane pattern
- High Force Alloy 718 Bottom Grid

- ZIRLO™ Tube-in-Tube Guide Thimble Assembly[1]
- Debris Filter Bottom Nozzle (DFBN)
- Oxide Coated Clad for debris mitigation
- ZIRLO™ fuel rod clad and instrumentation tubes

The 14x14 422V+ fuel design differs from that of 14x14 OFA, with the unique features as described in Section 2.8.1 of this report. Several features in the 14x14 422V+ design that affect nuclear design are:

- A changed fuel stack height within the assembly (increase of 1.85 inches)
- A longer fuel rod (increase of 3.6 inches)
- A longer fuel assembly (increase of 0.040 inches)
- A wider pellet-to-clad gap (increase of 0.25 mil)
- A larger pellet diameter (increase of 0.0215 inches)
- A larger clad diameter (increase of 0.022 inches)

These changes, taken together, result in a noticeably larger rod plenum volume to accommodate fission gas release from the extended burnups of the 14x14 422V+ design and the helium release from IFBA (see Section 2.8.1).

The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worths, and reactivity parameters are loading pattern dependent. The variations in loading pattern dependent safety parameters are expected to be similar to the cycle-to-cycle variations for typical fuel reloads.

The nuclear design is affected by technical specification changes for the uprated power conditions. Relaxed Axial Offset Control (RAOC) has been introduced independently in a prior Ginna submittal (Reference 1), and is reflected in the analyses discussed here.

No changes to the nuclear design philosophy, methods or models are necessary because of the transition to 14x14 422V+ fuel or the EPU. The reload design methodology includes the evaluation of the reload core key safety parameters which comprise the nuclear design-dependent input to the UFSAR safety evaluation for each reload cycle (Reference 2). These key safety parameters will be evaluated for each Ginna reload cycle. If one or more of the parameters fall outside the bounds assumed in the reference safety analysis, the affected

[1] ZIRLO™ is a trademark property of Westinghouse Electric Company, LLC

transients will be re-evaluated/re-analyzed using standard methods and the results documented in the reload evaluation for that cycle.

Table 2.8.2-1 provides the key safety parameter ranges compared to the current limits.

2.8.2.2.3 Description of Analyses and Evaluations

Standard nuclear design analytical models and methods (References 2, 3 and 4) accurately describe the neutronic behavior of the 14x14 422V+ fuel design. The specific design bases and their relation to the General Design Criteria (GDC) in 10CFR50, Appendix A for the 14x14 422V+ design are the same as those of the OFA design (Section 3.1 of Reference 5).

The effect of extended burnup on nuclear design parameters has been previously discussed in detail in Reference 6.[2] That discussion is valid for the anticipated 14x14 422V+ design discharge burnup level. In accordance with the NRC recommendation made in their review of Reference 6, Westinghouse will continue to monitor predicted versus measured physics parameters for extended burnup applications.

The 0.422" OD fuel rod has had extensive nuclear design and operating experience with the original Ginna 14x14 STD fuel assembly. This change has no effect on the ability of standard nuclear design analytical models and methods to accurately describe the neutronic behavior of the 14x14 422V+ fuel (Reference 5).

Core loading patterns for multiple cycles were established to model the transition to a full 14x14 422V+ fueled core. These core loading patterns incorporated assembly dimensional and fuel rod modifications and plant uprating.

Typical loading patterns were developed based on projected energy requirements of approximately 510 EFPDs for Ginna. These models are not intended to represent limiting loading patterns, but were instead developed with the intent to show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. Six core designs were developed and used for the majority of calculations performed here. Existing designs (including current designs) were used for comparison to evaluate the continued adequacy of margins between typical safety parameter values and the corresponding limits.

The first "transition" cycle model was used to capture the initial and predominant transition effects. Appropriate models for the transition (to equilibrium) were developed and used to

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[2] While the 14x14 422V+ product is capable of being extended to a lead rod burnup of up to 75,000 MWD/MTU, VANTAGE + is currently licensed to 60,000 MWD/MTU by the NRC (Reference 4) with extension to 62,000 MWD/MTU on a cycle-specific basis, as delineated in Reference 6, Appendix R).

capture the core characteristics when a full core of the 14x14 422V+ fuel is present at uprated conditions. A key design assumption is the utilization of a range of vessel average temperatures that would bound the best-estimate average temperature. To capture these effects, both "high" and "low" temperature models were generated covering the range of temperatures noted in Table 2.8.2-1.

The increase in elevation of the top nozzle adapter plate in the 14x14 422V+ fuel impacts the rod bottom position. The impact of this on available shutdown margin has been explicitly addressed in the nuclear design. The increased elevation also changes the overlap between rod cluster control assembly banks in 14x14 422V+ and 14x14 OFA fuel and this has also been addressed in the nuclear design.

2.8.2.2.4 Results

Margin to key safety parameter limits (Table 2.8.2-1) is not reduced by the 14x14 422V+ fuel design relative to the 14x14 OFA design in similar applications. Key design characteristics for all models (vessel average temperature range) are summarized in Table 2.8.2-2. Normal operation (Condition I) peaking factors ($F_Q(Z)$) relative to core elevation are summarized in Table 2.8.2-3.

The changes in fuel design and discharge burnup caused only a small impact on the results of the reload transition core analysis relative to the current design. The variations in these parameters are typical of the normal cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle. The implementation of increased peaking factor limits has not resulted in changes to the other key safety parameter ranges.

Changes to the core power distributions and peaking factors are the result of the normal cycle-to-cycle variations in core loading patterns. The discharge burnups and assembly requirements have increased, relative to the current design, due to the increase in core power (Table 2.8.2-2). These will vary cycle-to-cycle based on actual energy requirements. The normal methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control peaking factors. Compliance with the peaking factor Technical Specifications can be assured using these methods.

The key safety parameters evaluated for Ginna as it transitions to an all 14x14 422V+ core and EPU show little change relative to the current design. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change.

Power distributions and peaking factors show slight changes as a result of mechanical design changes, in addition to the normal variations experienced with different loading patterns. The usual practices of enrichment and burnable absorber usage will be employed in the transition and full 14x14 422V+ cores to ensure compliance with the Technical Specifications.

In summary, the changes from the current OFA fuel core to a core containing the upgraded fuel product will not cause changes to the current Ginna UFSAR nuclear design bases. Nuclear design methodology is not affected by the use of upgraded fuel features or the EPU.

2.8.2.2.5 References

1. Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), License Amendment Request Regarding Adoption of Relaxed Axial Offset Control (RAOC), dated April 29, 2005.
2. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.
3. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.
4. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.
5. Davidson, S. L. (Ed.), et al., "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610-P-A, April 1995.
6. Davidson, S. L. (Ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December 1985.
7. Davidson, S. L. (Ed.), et al., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary), WCAP-14204-A (Non-Proprietary), October 1994.

2.8.2.3 Conclusion

The Ginna staff has reviewed the analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The Ginna staff concludes that the analyses have adequately accounted for the effects of the proposed EPU on the nuclear design and have demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core. As incorporated within the Ginna design basis, the nuclear design analyses form the basis for confirmation of acceptable compliance with the requirements of GDC-10 in accordance with the approved reload process of WCAP-9273 (Reference 2). Therefore, based on these analyses, in conjunction with the analyses of the fuel system design, thermal and hydraulic design, and transient and accident analyses, it is concluded that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, -11, -12, -13, -

20, -25, -26, -27, and -28. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the nuclear design.

Table 2.8.2-1

Range of Key Safety Parameters

Safety Parameter	Current Design Values	Analysis Values
Reactor Core Power (MWt)	1520	1811[3]
Vessel Average Coolant Temp. HFP (°F)	561	564.6 to 576.0*
Coolant System Pressure (psia)	2250	2250
Most Positive MTC (pcm/°F)	≤ + 5.0 (Power < 70%) ≤ 0.0 (Power ≥ 70%)	≤ + 5.0 (Power < 70%) ≤ 0.0 (Power ≥ 70%)
Most Positive MDC ($\Delta K/g/cm^3$)	0.43	0.45
Doppler Temperature Coefficient (pcm/°F)	-2.90 to -0.91	-2.90 to -0.91
Doppler Only Power Coefficient (pcm/%Power)	(See below)	(See below)
Least Negative, HFP to HZP	-9.55 to -5.35	-12.0 to -6.6
Most Negative, HFP to HZP	-19.40 to -11.24	-24.0 to -12.0
Beta-Effective	0.0043 to 0.0072	0.0043 to 0.0072
Normal Operation $F^N_{\Delta H}$ (with uncertainties)	1.75	1.72 (422V+) 1.60 (OFA)
Shutdown Margin (%Δρ)	2.45 (N-1 Loop Operation) 1.80 (N Loop Operation)	1.80 (N-1 Loop Operation) 1.30 (N Loop Operation)
Normal Operation $F_Q(Z)$	2.50 (Non-LOCA) 2.45 (LOCA)	2.60

[3]Analyzed core power level assumed in the depletion model. This value bounds best-estimate core power level of 1775 MWt.

*Constant temperature program assumed during nominal depletion.

Table 2.8.2-1
(continued)
Range of Key Safety Parameters

Safety Parameter	Current Design Values		Analysis Values	
Trip Reactivity versus Rod Position	Fraction of Rod Insertion	Rod Worth (%Δk)	Fraction of Rod Insertion	Rod Worth (%Δk)
	0.00	0.0000	0.00	0.0000
	0.02	0.0005	0.10	0.0245
	0.10	0.0070	0.20	0.0525
	0.25	0.0500	0.50	0.1050
	0.40	0.1500	0.60	0.1750
	0.6583	0.6000	0.80	0.5250
	0.80	1.6000	0.90	2.1000
	0.90	3.5000	0.96	3.1500
	1.00	4.0000	1.00	3.5000
Rod Ejection	(See Below) BOL	(See Below) EOL	(See Below) BOL	(See Below) EOL
Maximum Ejected Rod Worth (%Δρ)	0.78 (HZP) 0.30 (HFP)	0.95 (HZP) 0.42 (HFP)	0.75 (HZP) 0.32 (HFP)	0.90 (HZP) 0.40 (HFP)
Maximum Ejected Rod $F_Q(Z)$	7.80 (HZP) 5.00 (HFP)	14.0 (HZP) 5.69 (HFP)	11.0 (HZP) 5.00 (HFP)	14.0 (422V+ at HZP) 12.0 (OFA at HZP) 5.69 (HFP)
Maximum Burnup at Ejected Rod Hot Spot (MWD/MTU)	31034 (HZP) 31034 (HFP)	48276 (HZP) 48276 (HFP)	31034 (HZP) 31034 (HFP)	48276 (HZP) 48276 (HFP)

Table 2.8.2-2

Transition Cycles Core Characteristics for

"High" and "Low" Vessel Average Temperature Models

Vessel Average Temp. (°F)	Transition Cycle	Number of Feed Assemblies	HFP ARO $F_{\Delta H}$ (422V+)	HFP ARO $F_{\Delta H}$ (OFA)	Hot Zero Power MTC (pcm/°F)	Maximum F_Q (Eq. Xenon)
564.6	First	53	1.551	1.386	4.032	1.898
	Second	44	1.552	0.773	0.169	1.872
	Equilibrium	45	1.557	—	0.078	1.867
576.0	First	53	1.536	1.384	4.372	1.895
	Second	44	1.549	0.766	0.939	1.874
	Equilibrium	45	1.543	—	0.688	1.857

Table 2.8.2-3

Normal Operation

(Condition I) Maximum F_Q Relative to Core Elevation

Transition Cycle	Vessel Average Temperature (°F)	Max Transient F_Q	Minimum Power Margin to 2.60 F_Q Limit (%)	Limiting Elevation (ft.)
First	564.6	2.407	7.4	1.8
	576.0	2.362	9.2	1.8
Second	564.6	2.415	7.1	2.0
	576.0	2.432	6.5	1.8
Equilibrium	564.6	2.405	7.5	2.0
	576.0	2.420	6.9	1.8

2.8.3 Thermal and Hydraulic Design

2.8.3.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) has reviewed the core thermal and hydraulic design analyses and the reactor coolant system to confirm that the design:

- Has been accomplished using acceptable analytical methods,
- Is equivalent to or a justified extrapolation from proven designs,
- Provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and
- Is not susceptible to thermal-hydraulic instability.

The Ginna review of the analyses also covered hydraulic loads on the core and reactor coolant system components during normal operation and design basis accident conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events.

The NRC's acceptance criteria are based on:

- 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
- GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to ensure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can reliably and readily be detected and suppressed

Specific review criteria are contained in SRP, Section 4.4 and other guidance provided in Matrix 8 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 safety related structures, systems and components with respect to thermal and hydraulic design relative to conformance to:

- 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.

- GDC-12 is described in Ginna UFSAR section 3.1.1.2.3 which states that the reactor core and the associated coolant, control, and protection systems, and operating strategies have been designed to prevent or easily suppress power oscillations that could result in exceeding fuel design limits.

In addition to the evaluations described in the UFSAR, the Ginna core thermal and hydraulic analyses 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.

2.8.3.2 Technical Evaluation

2.8.3.2.1 Introduction

This section describes the thermal-hydraulic (T/H) analysis supporting the Ginna extended power uprate (EPU) and the transition from the current 14X14 Optimized Fuel Assembly (OFA) design with 0.400 inch diameter rods to the 14X14 422VANTAGE+ (14X14 422V+) fuel assembly design with 0.422 inch diameter rods. The current licensing basis for T/H design for Ginna Station includes the prevention of DNB on the limiting fuel rod with a 95% probability at a 95% confidence level and criteria to ensure fuel cladding integrity, and is documented in Section 4.4 of the Ginna UFSAR. The EPU analysis is based on this licensing basis analysis incorporating the increased core power and the transition to the 14x14 422V+ fuel design. The analysis addresses the departure from nucleate boiling (DNB) performance, including the effects of fuel rod bow and bypass flow. In a mixed core, with assemblies having different hydraulic resistance, the local hydraulic resistance differences are a mechanism for flow redistribution. Consequently, transition core effects on the fuel hydraulic compatibility during the transition from an all 14X14 OFA core through mixed-fuel cores to an all 422V+ core are explicitly evaluated.

Also considered in this section are:

- The effects of hydraulic compatibility associated with higher resistance fuel assemblies (the OFA design) on assembly lift forces,
- The calculation of fuel temperature/pressure data used in various safety analyses, and
- Core stored energy.

2.8.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

VIPRE-01 is the Core Thermal-hydraulic sub-channel analysis code that was used for the EPU analysis. NRC approval of the Westinghouse VIPRE-01 methodology was issued in the SER attached to Reference 1.

For the purposes of the EPU analysis, fuel-related safety and design parameters have been chosen to bound the current 14x14 OFA fuel assembly and the upgraded 14x14 422V+ fuel assembly. These bounding parameters have been used in the safety and design analyses discussed in this section and in other relevant sections of this report.

The changes from the current 14x14 OFA design to the 14x14 422VANTAGE+ (422V+) design that are important to the T/H analysis reported in this section are:

- 0.400 inch outer diameter (OD) fuel rod to 0.422 inch OD fuel rod
- 0.34 inch diameter pellets to 0.37 inch diameter pellets[1]
- mismatch in grid centerline elevations for the top grid
- An increase in the rod internal plenum volume
- 141.4 inch to 143.25 inch fuel rod stack height
- lower pressure drop mixing vane mid-grid design

[1] Pellet diameters are reference dimensions for uncoated pellets.

Table 2.8.3-1 lists the thermal-hydraulic parameters for the current design at 1520 MWt with 14x14 OFA fuel, as well as for the EPU design at 1775 MWt with the 14x14 OFA and 14x14 422V+ fuel designs. Some of the parameters listed in Table 2.8.3-1 are used in the analysis basis as VIPRE-01 input parameters while others are provided since they are listed in the UFSAR. This section identifies those parameters that are used as input parameters to the VIPRE-01 model and also identifies the limiting direction of each parameter. The following parameters from Table 2.8.3-1 are used in the VIPRE-01 model:

- Reactor core heat output (MWt)
- Heat generated in fuel (%)
- Nominal vessel/core inlet temperature (°F)
- $F^N_{\Delta H}$, nuclear enthalpy rise hot-channel factor
- Pressurizer/core pressure (psia)
- Thermal design flow (gpm)

In addition, the average linear power (kW/ft) is used in the PAD analyses for the fuel temperatures and other fuel rod design parameters. The limiting direction for these parameters is shown in Table 2.8.3-2.

The thermal-hydraulic design criteria and methods are the same as those presented in the Ginna Updated Final Safety Analysis Report (UFSAR) (with the addition of the Advanced Setpoints Methodology (Reference 2)). While the methods are the same, the VIPRE-01 code was used instead of the THINC IV code for all core thermal-hydraulic safety analyses.

The thermal-hydraulic analysis of the 14x14 OFA and 422V+ fuel in Ginna is based on the Revised Thermal Design Procedure (RTDP) (Reference 3), the WRB-1 DNB correlation (Reference 4), and the VIPRE-01 code (Reference 1). The DNB analysis of the core containing 14x14 422V+ fuel assemblies has been shown to be valid with the WRB-1 DNB correlation in References 4 and 5. The W-3 correlation and Standard Thermal Design Procedure (STDP) are still used when any one of the conditions is outside the range of the WRB-1 correlation (that is, pressure, local mass velocity, local quality, heated length, grid spacing, equivalent hydraulic diameter, equivalent heated hydraulic diameter, and distance from last grid to critical heat flux (CHF) site) and RTDP (that is, the statistical variance is exceeded on power, T_{IN}, pressure, flow, bypass, $F^N_{\Delta H}$, $F^E_{\Delta H,1}$, and F^E_Q).

The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in DNB performance due to the mixing vane grid effects. NRC acceptance of a 95/95 correlation limit DNB ratio (DNBR) of 1.17 for the 14x14 OFA fuel assemblies is documented in Reference 4. Furthermore, it has been shown that the use of the WRB-1 correlation with a 95/95 correlation limit DNBR of 1.17 is appropriate for the 14x14

422V+ fuel assemblies. The WRB-1 correlation applicability has been accepted by the NRC for the Point Beach Units with 14x14 422V+ fuel (Reference 5).

With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used to define the plant-specific design limit DNBR that satisfies the DNB design criterion. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The uncertainties included in the overall DNB uncertainty factor are:

- The nuclear enthalpy rise hot channel factor, ($F^N_{\Delta H}$)
- The enthalpy rise engineering hot channel factor, ($F^E_{\Delta H}$)
- Uncertainties in the VIPRE-01 and transient codes
- vessel coolant flow
- effective core flow fraction
- core thermal power
- coolant temperature
- system pressure

Because the uncertainties are incorporated in the DNBR limit, nominal values of the peaking and hot channel factors are used as input to the DNB safety analyses. Table 2.8.3-3 provides a listing and description of the peaking factor uncertainties.

Instrumentation uncertainties in core thermal power, RCS flow, pressure and temperature used for the fuel transition and EPU analyses, are listed in Table 2.8.3-4. The instrumentation uncertainties were used in determining the DNBR design limits.

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

A. There is at least a 95% probability that DNB will not occur on the limiting fuel rods during MODES 1 and 2, operational transients, or any condition of moderate frequency at a 95% confidence level.

B. No fuel melting during any anticipated normal operating condition, operational transients, or any conditions of moderate frequency.

The ratio of the heat flux causing DNB at a particular core location, as predicted by a DNB correlation, to the actual heat flux at the same core location is the DNBR. Analytical assurance that DNB will not occur is provided by showing the calculated DNBR to be higher than the 95/95 Limit DNBR for all conditions of normal operation, operational transients and transient conditions of moderate frequency. The Design Limit DNBR is calculated by using the RTDP methodology, which includes appropriate margin to DNB for all operating conditions sufficient to assure compliance with the DNBR criteria above.

For use in the DNB safety analyses, the Design Limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow, transition core, and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. This increase in the design limit DNBR to account for various penalties and operational issues is the plant-specific margin retained between the Design Limit DNBR and the Safety Analysis Limit (SAL) DNBR.

2.8.3.2.3 Description of Analyses and Evaluations

For the fuel transition and EPU analysis, the design limit DNBR values for the 14x14 422V+ fuel is 1.24 for typical and thimble cells. After accounting for the plant-specific margin, the SAL DNBR for the 422V+ fuel is 1.38/1.38 (typical/thimble). These SALs are employed in the DNB analyses.

With the SAL DNBR set, the core limit lines, axial offset limit lines, and dropped rod limit lines are generated. Based on these limit lines, the maximum $F^N_{\Delta H}$ limit that can be supported is 1.72 for the 422V+ fuel. This limit incorporates all applicable uncertainties, including a measurement uncertainty of 4 percent (Reference 6), and is adjusted for power level using the equation:

$$F^N_{\Delta H} = 1.72 \times [1 + 0.3(1-P)]$$

where P is the fraction of full power.

Rod bow can occur between mid-grids, reducing the spacing between adjacent fuel rods and reducing the margin to DNB. Rod bow must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Westinghouse has conducted tests to determine the impact of rod bow on DNB performance; the testing and subsequent analyses were documented in Reference 7.

Currently, the maximum rod bow penalty for the OFA fuel assembly is [ ]^{a,c} at an assembly average burnup of 24,000 MWD/MTU (References 7 and 8). No additional rod bow penalty is required for burnups greater than 24,000 MWD/MTU since credit is taken for the effect of $F^N_{\Delta H}$ burndown due to the decrease in fissionable isotopes and the buildup of fission products (Reference 9). Based on the testing and analyses of various fuel array designs documented in Reference 7, including the 14x14 STANDARD assembly, the 14x14 OFA and the 14x14 422V+ fuel assemblies should have the same rod bow penalty applied to the analysis basis as that used for 14x14 STANDARD fuel assemblies.

Two different bypass flow rates are used in the thermal-hydraulic design analysis. The thermal design bypass flow (TDBF) is the conservatively high core bypass flow used with the thermal design flow (TDF) in power capability analyses that use standard (non-statistical) methods, and is also used to calculate fuel assembly pressure drops. The best estimate bypass flow (BEBF) is the core bypass flow that would be expected using nominal values for dimensions and operating parameters that affect bypass flow without applying uncertainty factors. The BEBF is used in conjunction with the vessel minimum measured flow (MMF) for power capability analyses using the ITDP or RTDP (statistical) design procedures. The BEBF is also used to calculate fuel assembly lift forces.

Flow redistribution occurs between adjacent fuel assemblies with different hydraulic resistances, resulting in a net reduction in flow in the higher-resistance assemblies. Crossflow can also be induced by local hydraulic resistance differences, such as differences in grid elevations and resistances. The flow redistribution affects both mass velocity and enthalpy distribution, which, in turn, affect DNB. The design procedure establishes a transition core DNBR penalty due to flow redistribution, and all further plant-specific analysis proceeds as if it were a full core analysis.

Excessive crossflow is prevented in the EPU transition cores by maintaining grid-to-grid overlap between the 14x14 OFA design and the 14x14 422V+ design, and by ensuring that the difference in grid loss coefficients between the two designs is within previous Westinghouse design experience. The only exception is the variation in the top-grid centerline elevation, which is approximately 3 inches higher in the 422V+ fuel assemblies (see LR section 2.8.1 Fuel System Design), and has been explicitly considered in the axial flow distributions assumed in the thermal-hydraulic analysis. A comparison of fuel assembly crossflow velocities due to grid pressure drop mismatch between the 14x14 Westinghouse 422V+ fuel assembly design and the 14x14 OFA design was performed.

Full-scale hydraulic tests were performed on the 14x14 Westinghouse 422V+ fuel assembly design to confirm the pressure loss compatibility with the 14x14 OFA fuel design. The 14x14 Westinghouse 422V+ fuel assembly design pressure drop is approximately [ ][a,c] lower than the 14x14 OFA design due principally to differences in the grid designs (see Section 2.8.1 Fuel System Design).

Transition cores were analyzed as if they were full cores of one assembly type (full 14x14 422V+ and full 14x14 OFA), applying the applicable transition core penalty as a function of the number of each fuel assembly type in the core using the NRC-approved methodology detailed in Reference 11 and approved in Reference 10.

The thermal-hydraulic analysis was performed using VIPRE-01 with the RTDP methods described above, including fuel rod bow, bypass flow and flow redistribution effects on a full core of 14x14 422V+ fuel. The results of this analysis are presented in LR section 2.8.3.2.4.

The introduction of lower resistance assemblies will influence the lift forces on the remaining assemblies. While the flow redistribution tends to reduce the flow in the higher resistance assemblies, the lower resistance 422V+ fuel assemblies will have a higher average flow than they would in a full core situation. Thus, the lift force on these 422V+ assemblies will be higher during the transition cores, and will be greatest during the first transition cycle since this will have the highest number of co-resident 14x14 OFA fuel assemblies.

Fuel temperatures and associated rod internal pressures have been generated using the NRC-approved PAD code (Reference 12) for the 422V+ fuel. The characteristics of the 14x14 OFA and 14x14 422V+ designs are very similar except for the rod diameter. The 14x14 422V+ design also includes a larger rod plenum for gas accommodation. The performance criteria employed by Westinghouse for the 14x14 OFA and 14x14 422V+ designs are the same throughout life, in terms of fuel temperatures, rod internal pressures, and core stored energy.

The fuel rod average and surface temperatures are needed for the accident analyses. In addition, minimum fuel average and fuel surface temperatures are required by Non-LOCA Analysis. Fuel centerline temperatures were also generated for the 422V+ fuel. These will be used for future verification, during reload design validation, that fuel melt will not occur.

In addition to the fuel temperatures and pressures, the revised core stored energy for the 422V+ fuel has been determined for use in containment analysis (refer to LR section 2.6). Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The local core stored energy is normalized to the local linear power level.

2.8.3.2.3.1 RCCA Drop/Misoperation

This section supplements the methodology discussion of LR section 2.8.5.4.3 for this non-LOCA event. The NRC-approved Westinghouse analysis methods in Reference 13 were used for analyzing the RCCA drop event. The Dropped Rod Limit Lines (DRLL) define DNB-based limits on peaking factors as functions of core inlet temperature, core power and pressure. Based on the DRLL and transient statepoints covering a range of reactivity insertion mechanisms, nuclear design calculations determined pre-drop $F_{\Delta H}$ values corresponding to the post-drop peaking factors at the SAL DNBR. The maximum pre-drop $F_{\Delta H}$ for each reload is specified in the Core Operating Limit Report (COLR). The cycle-specific RCCA drop analysis confirms that all allowed pre-drop $F_{\Delta H}$ values do not violate the COLR limit, and the DNB design basis is met for power uprate. In addition, the maximum linear heat rate from the RCCA drop analysis was 21.0 kw/ft, which is lower than the fuel centerline melt limit. Therefore, the peak fuel centerline melt temperature criterion is also met for this event.

2.8.3.2.3.2 Uncontrolled Rod Cluster Control Assembly Withdrawal from Subcritical

The analysis for the Uncontrolled Rod Cluster Control Assembly Withdrawal from Subcritical is based on the STDP methodology since the event was initiated from Hot Zero power conditions. Results and additional information are contained in LR section 2.8.5.4.1.

2.8.3.2.3.3 Steam Line Break Accident

The event description is provided in LR section 2.8.5.1.2. The prime candidate for the worst stuck rod for the HZP Large Steam Line Break analysis is typically the highest worth stuck rod in the core or the stuck rod that produces the highest $F_{\Delta H}$. A lesser worth rod that produces a lower $F_{\Delta H}$ located in a colder region of the core could be evaluated since it may result in a more limiting DNBR. During a SLB event, regions of the core nearest the vessel inlet from the faulted loop will be coldest. A confirmation of the statepoint reactivities is assessed to ensure a no-return-to-power condition. Limiting power shapes and reactivity data are provided to T/H for confirmation of the DNBR criteria. The continued validity of the limiting power shapes is based on confirmatory calculations for each cycle employing Relaxed Axial Offset Control (RAOC) methods. The DNBR confirmation was performed using the W-3 DNBR correlation and the STDP methodology due to the low pressure condition at the limiting time step.

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

In addition to the evaluations described in the UFSAR, the core thermal and hydraulic analyses considerations were evaluated for plant License Renewal. No systems or components are being added or modified as the result of re-evaluation of the core thermal and hydraulic analyses for EPU conditions. The core thermal and hydraulic 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.

2.8.3.2.4 Results

Table 2.8.3-5 summarizes the available DNBR margin for Ginna power uprate. It should be noted that the DNBR margin summaries are cycle-dependent and may vary from cycle-to-cycle in future reload designs. The continued satisfaction of the DNBR criterion for reload cycles is confirmed via the approved reload methodology of WCAP-9273-NP-A (Reference14).

For the Ginna analyses, the maximum permissible TDBF is [ ]^{a,c} percent and the maximum permissible BEBF is [ ]^{a,c} percent.

[

]a,c

The 14x14 422V+ design allows power uprating at an $F^N_{\Delta H}$ limit of 1.72. All the thermal-hydraulic design criteria are satisfied for the Ginna EPU fuel transition. The anticipated reduction in margin that would result from the increased power level has been offset by the following margin contributors:

- Larger fuel rod diameter (i.e. lower heat flux) for 422V+ fuel design,
- The use of the advanced setpoint methodology (Reference 2) for 422V+ and OFA fuel designs, and
- The use of a lower $F_{\Delta H}$ for the OFA fuel.

The uprate analysis demonstrates that the combined DNBR margin gain is enough to accommodate the extended power uprate to 1775 MWt.

The hydraulic compatibility of the 14x14 422V+ and 14x14 OFA fuel assemblies has been addressed and found to be acceptable. The difference in loss coefficients between the two designs and the respective grid locations of the two designs has been analyzed to demonstrate that crossflow-induced vibration will not result in fretting. The expected fuel assembly crossflow is well within Westinghouse experience with transition cores with intermediate flow mixing (IFM) vane grids. The crossflow velocity profile versus height for the Ginna fuel transition is shown in Figure 2.8.3-1.

The maximum kW/ft limit for fuel melt is [ ]a,c for the 14x14 422V+ fuel.

Fuel temperatures were generated for the 14x14 422V+ fuel for use in the safety analyses. Figure 2.8.3-2 provides representative data (based on non-IFBA fuel) for the maximum and minimum fuel rod average temperatures. Figure 2.8.3-3 provides the fuel surface temperatures corresponding to the maximum and minimum fuel average temperatures in Figure 2.8.3-2. Figure 2.8.3-4 provides the maximum and minimum fuel centerline temperatures.

Fuel rod internal pressure is important in assessing the degree of burst and blockage which may occur after a loss-of-coolant accident. Pressures are computed with the PAD codes (Reference 12). Fuel parameters for reload fuel are evaluated to confirm that the pressures used in the reference analysis remain applicable to the reload.

The core stored energy for the 14x14 422V+ fuel is 5.25 Full Power Seconds (FPS).

The evaluations demonstrate that the minimum DNBR for the static misaligned rod event is above the SAL DNBR. In addition, the maximum calculated linear heat rate for the static misaligned rod event was less than the fuel centerline melt limit at uprate conditions. Therefore, the peak fuel centerline melt temperature criterion is confirmed to be met.

The SAL DNBR is met for the HZP Large Steam Line Break, including the effects of the worst stuck rod, based on the power distributions from the reference loading plan. A confirmational DNBR calculation will be performed as part of each reload design in accordance with the WCAP-9273-NP-A reload methodology (Reference 14).

The thermal-hydraulic evaluation of the fuel upgrade for Ginna has shown that 14x14 OFA and 14x14 422V+ fuel assemblies are hydraulically compatible and that the DNB margin gained through use of the upgraded fuel is sufficient to allow an increase in the power rating to 1775 MWt. Sufficient DNBR margin in the SAL DNBR exists to cover any rod bow and transition core effects. All current thermal-hydraulic design criteria are satisfied.

Table 2.8.3-1 Ginna Thermal-Hydraulic Design Parameters Comparison

Thermal-Hydraulic Design Parameters	Current Design Value	EPU Analysis Value
Reactor Core Heat Output, MWt	1520	1775(1)
Reactor Core Heat Output, 10^6 BTU/Hr	5186	6057(1)
Heat Generated in Fuel, %	97.4	97.4
Core Pressure, Nominal, psia	2267	2265
Pressurizer Pressure, Nominal, psia	2250	2250
Radial Power Distribution(2)	1.75[1+0.3(1-P)], for OFA fuel	1.72[1+0.3(1-P)], for 422V+ fuel 1.60[1+0.3(1-P)] for OFA fuel
HFP Nominal Coolant Conditions		
Vessel Thermal Design Flow Rate (including bypass)		
10^6 lb_m/hr	64.71	64.71
GPM	170,200	170,200
Core Flow Rate (excluding Bypass,(4) based on TDF)		
10^6 lb_m/hr	60.50	60.50
GPM	159,137	159,137
Core Flow Area, ft^2	29.2 (full-core OFA)	27.1 (full-core 422V+)
Core Inlet Mass Velocity (based on TDF), 10^6 lb_m/hr-ft^2	2.07	2.23
Nominal Vessel/Core Inlet Temperature, °F	543.1	540.2
Vessel Average Temperature, °F	573.5	576.0
Core Average Temperature, °F	576.9	580.3
Vessel Outlet Temperature, °F	604.0	611.8
Core Outlet Temperature, °F	607.8	616.2
Average Temperature Rise in Vessel, °F	60.9	71.6
Average Temperature Rise in Core, °F	64.7	76.0
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	26,673	28,507
Average Heat Flux, BTU/hr-ft^2	189,374	206,950(1)
Average Linear Power, kW/ft	5.81	7.02
Peak Linear Power for Normal Operation,(3) kW/ft	15.11	18.25(1)
Peak Linear Power for Prevention of Centerline Melt, kW/ft	22.45	22.7
Pressure Drop Across Core, psi		
Full core of 14x14 OFA	30	26.6
Full core of 422V+	N/A	24.7

Table 2.8.3-1 Ginna Thermal-Hydraulic Design Parameters Comparison (con't)

Notes:

1. The proposed power level of 1775 MWt has been used for all thermal-hydraulic design analyses.
2. $P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$
3. Based on maximum F_Q of 2.6.
4. Design bypass flow of 6.5% was used for current and uprate conditions (both including 2% due to thimble plug removal).

Table 2.8.3-2 Limiting Parameter Direction

Parameter	Limiting Direction for DNB
$F^N_{\Delta H}$, nuclear enthalpy rise hot-channel factor	maximum
Heat generated in fuel (%)	maximum
Reactor core heat output (MWt)	maximum
Average heat flux (BTU/hr-ft^2)	maximum
Nominal vessel/core inlet temperature (°F)	maximum
Core pressure (psia)	minimum
Pressurizer pressure (psia)	minimum
Thermal design flow for non-RTDP analyses (gpm)	minimum
Minimum measured flow for RTDP analyses (gpm)	minimum

Table 2.8.3-3

Peaking Factor Uncertainties

	$F_{\Delta H} = F^N_{\Delta H} \times F^E_{\Delta H}$	
where: $F^N_{\Delta H}$		Nuclear Enthalpy Rise Hot Channel Factor – The ratio of the relative power of the hot rod, which is one of the rods in the hot channel, to the average rod power. The normal operation value of this is given in the plant Technical Specifications or a Core Operating Limit Report (COLR).
$F^E_{\Delta H}$		Engineering Enthalpy Rise Hot Channel Factor – The nominal enthalpy rise in an isolated hot channel can be calculated by dividing the nominal power into this channel by the core average inlet flow per channel. The engineering enthalpy rise hot channel factor accounts for the effects of flow conditions and fabrication tolerances. It can be written symbolically as:
	$F^E_{\Delta H} = f\,(F^E_{\Delta H,1}, F^E_{\Delta H,2}, F^E_{\Delta H\ \mathrm{inlet\ maldist}}, F^E_{\Delta H\ \mathrm{redist}}, F^E_{\Delta H\ \mathrm{mixing}})$	
where: $F^E_{\Delta H,1}$		accounts for rod-to-rod variations in fuel enrichment and weight
$F^E_{\Delta H,2}$		accounts for variations in fuel rod outer diameter, rod pitch, and bowing
$F^E_{\Delta H\ \mathrm{inlet\ maldist}}$		accounts for the non-uniform flow distribution at the core inlet
$F^E_{\Delta H\ \mathrm{redist}}$		accounts for flow redistribution between adjacent channels due to the different thermal-hydraulic conditions between channels
$F^E_{\Delta H\ \mathrm{mixing}}$		accounts for thermal diffusion energy exchange between adjacent channels caused by both natural turbulence and forced turbulence due to the mixing vane grids
The value of these factors and the way in which they are combined depends upon the design methodology used, that is, STDP or RTDP. Note that no actual combined effect value is calculated for $F^E_{\Delta H}$. These factors are accounted for by using the VIPRE-01 code.		

Table 2.8.3-4

RTDP Uncertainties

Parameter	Uncertainty Used in EPU Safety Analysis
Power	[ ]a,c
Reactor Coolant System Flow	[ ]a,c
Pressure	[ ]a,c
Inlet Temperature	[ ]a,c

Table 2.8.3-5

DNBR Margin Summary[1]

		14x14 OFA	14x14 422V+
DNB Correlation		[ ]a,c	[ ]a,c
DNBR Correlation Limit		[ ]a,c	[ ]a,c
DNBR Design Limit	(TYP)[2]	[ ]a,c	[ ]a,c
	(THM)[3]	[ ]a,c	[ ]a,c
DNBR SAL[6]	(TYP)	[ ]a,c	[ ]a,c
	(THM)	[ ]a,c	[ ]a,c
DNBR Retained Margin[4]	(TYP)	[ ]a,c	[ ]a,c
	(THM)	[ ]a,c	[ ]a,c
Rod Bow DNBR Penalty		[ ]a,c	[ ]a,c
Transition Core DNBR Penalty		[ ]a,c	[ ]a,c
Available DNBR Margin[5]	(TYP)	[ ]a,c	[ ]a,c
	(THM)	[ ]a,c	[ ]a,c

Notes:

1. Steam line break is analyzed using the W-3 correlation with STDP. The correlation limit DNBR is 1.45 in the range of 500 to 1000 psia. Rod withdrawal from subcritical is also analyzed using the W-3 correlation (w/o spacer factor) with STDP below the bottom non-mixing vane grid. The correlation limit DNBR is 1.30 above 1000 psia and the SAL DNBR is 1.447 (422V+) which provides [ ]a,c to cover the rod bow penalty and retain generic margin for operational issues. WRB-1 with RTDP is used for rod withdrawal from subcritical above the bottom non-mixing vane grid.
2. TYP = Typical Cell
3. THM = Thimble Cell
4. DNBR margin is the margin that exists between the SAL and the design limit DNBRs.
5. The margin summary for OFA corresponds to the first transition cycle. For the second transition cycle, the OFA DNBR transition penalty will increase; however, this will be offset by the $F_{\Delta H}$ reduction.
6. The SAL DNBR was changed from the current value of [ ]a,c in order to support the proposed Over Temperature ΔT (OTΔT) trip setpoint revisions. Sufficient DNBR margin has been retained to offset rod bow and transition core effects.

a, c

Crossflow velocity sign convention is as follows:

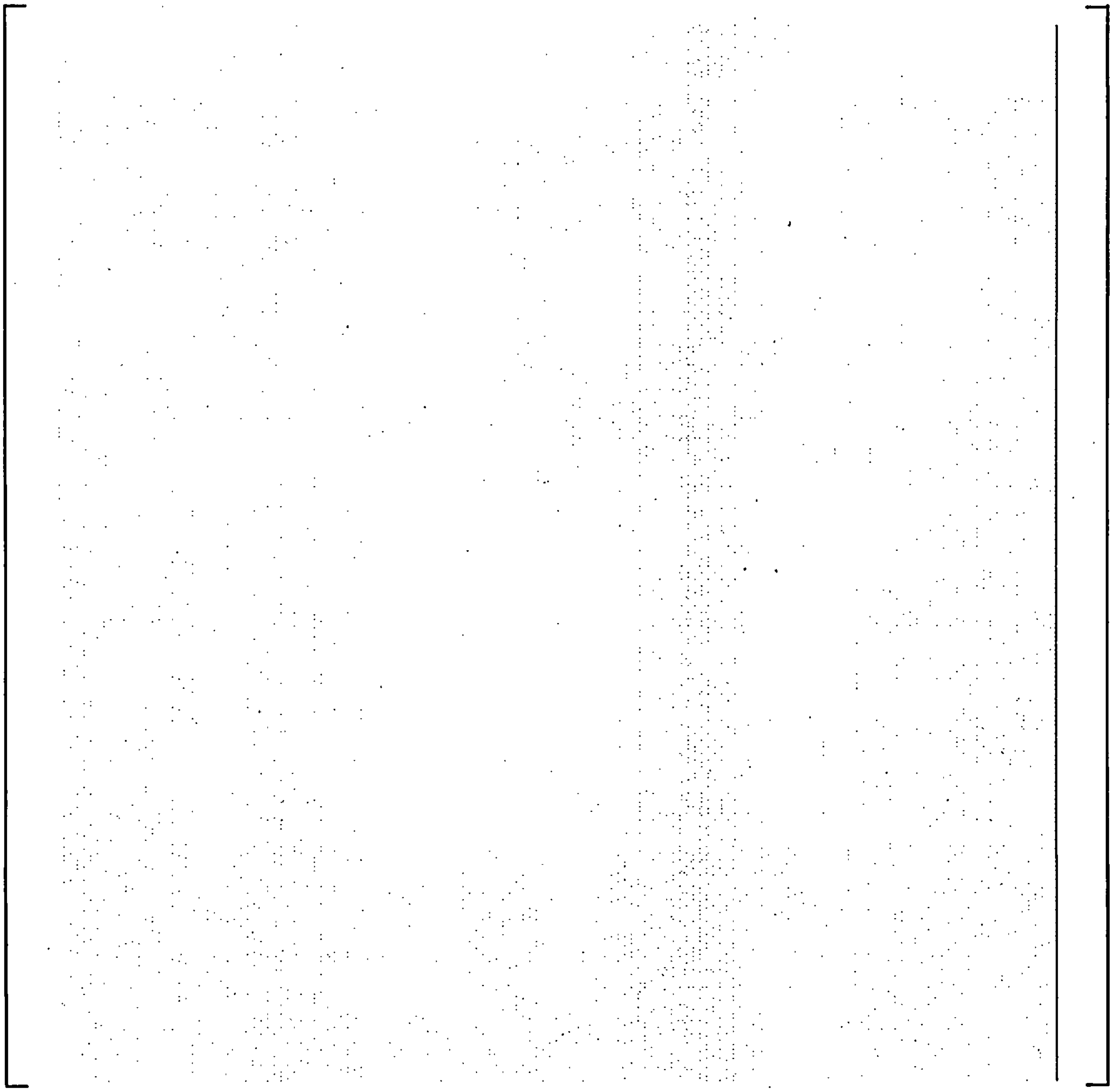
Plus (+) sign indicates flow into the 14x14 OFA fuel assembly and minus (-) sign indicates flow into the 14x14 422V+ fuel assembly[2]

Figure 2.8.3-1. Crossflow Velocity – Westinghouse 14x14 422V+ 14x14 OFA Fuel

[2] See Section 2.8.1 Fuel System Design for discussion of testing and basis for crossflow determination.

a, c

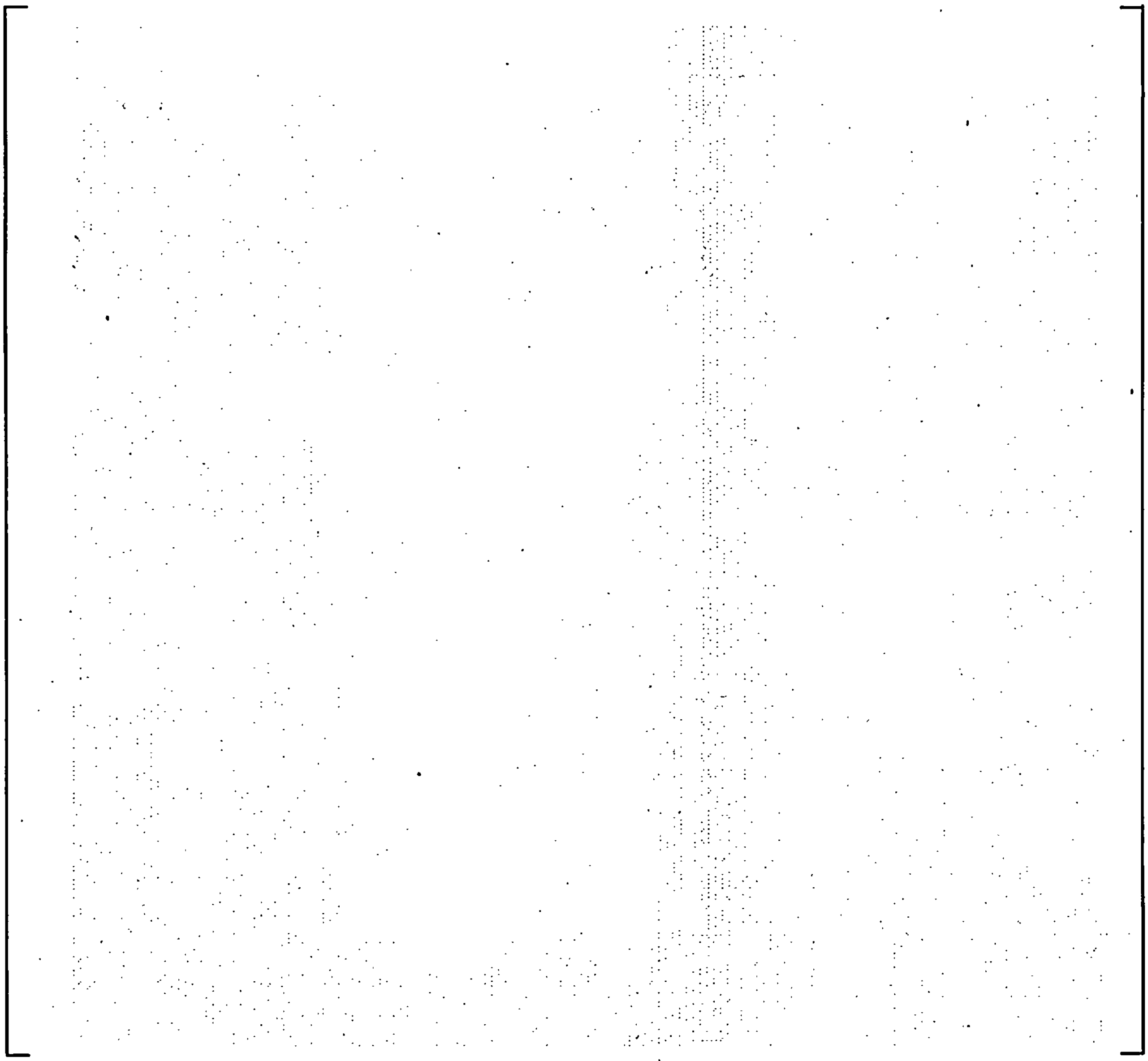
Figure 2.8.3-2. Fuel Average Temperatures



a, c

Figure 2.8.3-3. Fuel Surface Temperatures[3]

[3] The labels for Minimum Fuel Surface Temperature and Maximum Fuel Surface Temperature correspond to the Maximum and Minimum Fuel Average Temperatures in Figure 2.8.3-2.



a, c

Figure 2.8.3-4. Fuel Centerline Temperatures

2.8.3.2.5 References

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13. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.
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2.8.3.3 Conclusion

The Ginna staff has reviewed the analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the reactor coolant system. The Ginna staff concludes that these analyses have adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design:

- Has been accomplished using acceptable analytical methods,
- Is consistent with experience for similar designs,
- Provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and
- Is not susceptible to thermal-hydraulic instability

The Ginna staff further concludes that the thermal and hydraulic design will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10 and GDC-12 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

2.8.4.1.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements.

The NRC's acceptance criteria are based on:

- GDC-4, insofar as it requires 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-23, insofar as it requires that the protection system be designed to fail into a safe state
- GDC-25, insofar as it requires that the protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems
- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained
- GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core
- GDC-29, insofar as it requires that the protection and reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in event of anticipated operational occurrences

Specific review criteria are contained in the SRP, section 4.6.

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 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. During the SEP review, when margins to safety were determined to exist by analysis or review, those margins were incorporated into the Ginna UFSAR. Specifically, with respect to the adequacy of the Control Rod Drive System to the following:

- GDC-4 is described in the Ginna UFSAR section 3.1.2.1.4, General Design Criterion 4 - Environmental and Missile Design Bases. Additional information related to GDC 4 compliance can be found in UFSAR sections 3.5, Missile Protection, 3.6, Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping, and 3.11, Environmental Design Of Mechanical And Electrical Equipment.
- GDC-23 is described in the Ginna UFSAR section 3.1.2.3.4, General Design Criterion 23 – Protection System Failure Modes. GDC-23 requires protection systems be designed to fall 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.

  With respect to the control rod drive system, a 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.
- 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 Reactor Trip System (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 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-26 is described in the Ginna UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. GDC-26 states that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

  One of the two reactivity control systems employs control rod drive mechanisms to regulate the position of silver-indium-cadmium neutron absorbers within the reactor core. The control rods are designed to shut down the reactor with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The other reactivity control system employs the chemical and volume control system to regulate the concentration of boric acid neutron absorber in the reactor coolant system. The chemical and volume control system is capable of controlling the reactivity change resulting from planned normal power changes.

- GDC-27 is described in the Ginna UFSAR section 3.1.2.3.8, General Design Criterion 27 – Combined Reactivity Control System Capability. GDC-27 requires the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

  The reactivity control systems in conjunction with boron addition through the Emergency Core Cooling System (ECCS) has the capability of controlling reactivity changes under postulated accident conditions with appropriate margins for stuck rods.

  Ginna Station is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. Combined use of the rod cluster control system and the chemical shim control system permit the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown, even with the single highest worth control rod stuck out. In a loss-of-coolant accident the safety injection system is actuated and boric acid is injected into the cold legs of the reactor coolant system. This is in addition to the boric acid content of the accumulators which is passively injected on a decrease in system pressure.

- GDC-28 is described in the Ginna UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. GDC-28 states that reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

  The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the coolant pressure boundary or disruptions of the core or vessel internals to a degree which could impair the effectiveness of emergency core cooling. The Core Operating Limits Report (COLR) place appropriate restrictions on the maximum permissible insertion limits and overlap of rod cluster control assembly banks as a function of power.

- 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 Control Rod Drive System include:

- Ginna UFSAR section 3.9.4, Control Rod Drive Systems, which provides a general description of the mechanical design and operation of the control rod drive mechanism.
- Ginna UFSAR section 4.2, Fuel System Design, which provides a description of the design of the control rod cladding and the control rod withdrawal and insertion rate associated with reactor operational load changes.
- Ginna UFSAR section 7.2 Reactor Trip System, which provides a description of the reactor trip system interface with the control rod drive system.
- Ginna UFSAR section 7.7.1.2, Rod Control System, which provides a description of the operation of the control rod drive system and the microprocessor rod position indication (MRPI) and digital (demand) position systems.

- Ginna UFSAR section 9.4.1.2.3, Control Rod Drive Mechanism Cooling System which describes the design of control rod drive mechanism cooling system including the head assembly upgrade package (HAUP) installation.
- Ginna UFSAR Chapter 15.4, Reactivity and Power Distribution Anomalies, describe the transient and accident analyses associated with the malfunctions of the control rod drive and chemical and volume control systems.
- Ginna Technical Specification 3.1 and associated bases which describe the operability requirements for the control rods and control rod position indication system

In addition to the evaluations described in the UFSAR, control rod drive system and the control rod drive mechanism cooling system were evaluated for the Ginna Station License Renewal. With the exception of the cables from the rod control cabinets to the operating coil stacks, the control rod drive system is out of scope. 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.

2.8.4.1.2 Technical Evaluation - Control Rod Drive System

2.8.4.1.2.1 Introduction

The impact of EPU on the control rod drive system results from the transition from the current 14x14 OFA Westinghouse fuel to 14x14 422V+ Westinghouse fuel and temperature effects associated with increasing reactor core thermal power from 1520 MWt to 1811 MWt with an associated increase in reactor coolant system (RCS) average temperature from 561°F to 576°F. The increase in RCS average temperature is expected to increase vessel head temperature from 576.8°F to 599.2°.

As a result of EPU, there are no physical changes required to the control rod drive system, operating coil stacks, power supplies, solid state electronic control cabinets, or the control rod drive cooling system. Minor changes to the microprocessor rod position indication (MPRI) system described in LR section 2.8.4.1.2.2 below to accommodate the differences associated with the fuel assembly nozzle design between the OFA and 422V+ fuel are necessary to assure proper rod position indication to the plant operators. In addition, minor changes such as recalibration, rescaling, and setpoint changes to the reactor protection control system are required to facilitate the fuel design changes and the changes in operating conditions associated with operation at EPU. These changes are discussed in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

2.8.4.1.2.2 Description of the Analyses and Evaluations

Analyses of the differences in the key design features between the Westinghouse 14x14 OFA fuel and the Westinghouse 14x14 422V+ fuel and the effects with regard to the fuel design limits during the transition have been performed and are discussed in LR section 2.8.1, Fuel System Design Features, and LR section 2.8.2, Nuclear Design. These analyses determined the fuel and EPU changes are such that the results of the reload transition core analysis continue to comply with the currently applicable Ginna regulatory and industry design requirements and are therefore acceptable.

The impact of the transition to Westinghouse 422V+ fuel on the control rod drive system is due to the difference in the design of the fuel assembly top nozzles. The 14x14 422V+ fuel nozzle is approximately 3 inches shorter than the 14x14 OFA fuel nozzle. Therefore, the rodlet tips of RCCAs that are inserted in the 14x14 422V+ fuel will be approximately 3 inches higher than the RCCAs inserted in the 14x14 OFA fuel. The impact of this on shutdown margin has been assessed and determined to be minimal. There is no impact to the control rod drive cooling system as a result of the transition to the 422V+ fuel design.

The difference in the 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 between 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+ at 225 steps instead of the current 230 steps. In addition, there would exist the potential 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 individual rod position indications are available to the operator.

The affects to the control rod drive system associated with increasing reactor core power from 1520 MWt to 1811 MWt are:

- Increased thermal stresses associated with the structural integrity of the control rod mechanisms associated with the increased reactor coolant system head temperatures, and the increased hydraulic, cyclic, and seismic forces associated with normal, transient, and accident conditions at EPU conditions.
- Increased heat load to the control rod drive cooling system resulting from the higher head temperatures. The impact to the rod control cooling system is evaluated in LR section 2.8.4.1.3 below.

Transient and accident analyses for the events listed in the Ginna UFSAR chapter 15 were performed for the EPU conditions listed in LR section 1.1, Nuclear Steam Supply System Parameters Table 1-1. These analyses were performed taking into account the actual differences in rod position for those control rods inserted in the Westinghouse 14x14 422V+ fuel. These analyses are described in LR section 2.8.5, Accident and Transient Analyses. Results of the analyses identified that a change to the rod control system power mismatch unit rod speed controller variable gains would be necessary to ensure fuel design values would be maintained following a dropped rod or a continuous rod withdrawal event. All other events associated with the control rod drive system provided acceptable results and maintained DNB, the reactor coolant system pressure, and main steam system pressure within the acceptable limits. The change to the rod control system variable gains is discussed in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

Analyses and evaluations of the impact of EPU on the structural integrity of the control rod drive system during normal, transient, and accident conditions were performed using the EPU conditions listed in LR section 1.1, Nuclear Steam Supply System Parameters Table 1-1. These analyses and evaluations are discussed in LR section 2.2.2.4, Control Rod Drive Mechanisms, and LR section 2.2.6, NSSS Transient Analyses. The results of the analyses and evaluations determined the structural integrity of the control rod drive system remained within acceptable limits at EPU conditions.

2.8.4.1.3 Technical Evaluation – Control Rod Drive Mechanism Cooling System

2.8.4.1.3.1 Introduction

Control rod drive mechanisms (CRDMs) use electro-magnetic coils to position the rod cluster control assemblies (RCCAs) within the reactor core. The insulation and potting materials used in the construction of the coils are subject to thermal aging. In order to reduce the thermal aging, CRDM cooling systems were designed to remove heat supplied by conduction and convection from the reactor head and reactor coolant. These systems are the largest source of containment head load. The Ginna Station recently modified the CRDM cooling system to better facilitate reactor disassembly during refueling. The components of this modification are referred to as a Head Assembly Upgrade Package (HAUP) as discussed in UFSAR section 9.4.1.2.3, Control Rod Drive Mechanism Cooling System.

2.8.4.1.3.2 Input Parameters, Assumptions, and Acceptance Criteria

During cycle 31 start-up, with the HAUP installed, data was obtained for the CRDM cooling system volume flow rate and fan inlet temperature as a function of RCS average temperature.

The increase in CRDM cooling system heat load associated with the increase in reactor vessel head temperature was determined using Cycle 31 data. The data demonstrated the existence of a linear relationship between reactor vessel head temperature and the CRDM cooling system fan inlet temperature. The air temperature difference between the inlet of the CRDM cooling system and the fan inlet temperature was used to determine the heat rejected by the CRDMs based on knowing the mass flow rate of the system. The volume flow rate of the system was determined prior to plant startup by performing in-situ flow measurements. It was assumed that the CRDM cooling system inlet temperature remained constant during the period that the fan inlet temperatures were recorded. As a result, the change in outlet temperature alone was used to determine the additional heat rejected from the CRDMs as a function of reactor coolant system (RCS) temperature and reactor vessel head fluid temperature. Application of the constant fan inlet temperature assumption results in overestimating the increase in CRDM cooling system heat rejection rate and the associated increase in containment cooling systems heat load.

A maximum electro-magnetic coil temperature of 321°F was calculated following the implementation of the HAUP with an RCS average temperature of 561°F and a corresponding calculated reactor vessel head fluid temperature of 576.8°F. This is less than the maximum design temperature for the CRDM electro-magnetic coil of 392°F. In addition, using the measured data, a best estimate temperature of 178°F at the exit of the coils was calculated with a containment temperature of 120°F which is less than the current maximum evaluated ambient temperature of 213°F for the components of the MRPI system and coils.

The Ginna station reactor vessel head temperature for Cycle 31 with a core thermal power of 1520 MWt and an RCS average temperature of 561°F was calculated to be 576.8°F. From LR section 1.1, Nuclear Steam Supply System Parameters Table 1-1 Case 3 EPU conditions, the highest RCS average temperature of 576°F provides an estimated maximum reactor vessel head fluid temperature of 599.2°F.

The specific acceptance criteria is to demonstrate that the increased temperatures associated with EPU on the components and coils of the MRPI remains acceptable.

2.8.4.1.3.3 Description of Analyses and Evaluations

The impact of EPU on the CRDM Cooling System has been evaluated. It was conservatively assumed the electro-magnetic coil temperature increases by the same amount as the reactor vessel head fluid temperature. As stated previously the maximum calculated reactor vessel head fluid temperature of 599.2°F was added to the reactor vessel head ΔT associated with the EPU and compared to the maximum design temperature for the coil.

As indicated above, the maximum calculated coil temperature from the Cycle 31 operating condition with an RCS average temperature of 561°F and a reactor vessel head fluid temperature of 576.8°F was 321°F. It was conservatively assumed that the electro-magnetic coil experienced a corresponding increase in temperature which yields a maximum coil temperature of 334°F. During withdrawal of the control rods, addition heat load is applied as a result of the increased input to the electro-magnetic coils. Based on a 3.2 minute maximum electro-magnetic coil operating time the maximum expected temperature of the electro-magnetic coil would rise an additional 14°F to 348°F which is well below 392°F design limit.

Additional analyses using the data collected during the plant startup following the implementation of the HAUP was used to estimate the increase in containment heat load and the air temperature to the MRPI coils as a function of the reactor vessel head temperature at EPU.

The analyses determined that for the Table 1-1 Case 3 maximum RCS temperature conditions, the expected increase in CRDM cooling system average temperature at the exit of the operating coils due to EPU is approximately 6.7°F which results in an increase in containment heat load of approximately 211,000 btu/hr.

Analyses to determine the maximum local air temperature to the MRPI coils for single and dual CRDM cooling fan operation was performed. These analyses showed that with a containment temperature of 120°F, dual or single fan operation would maintain the maximum predicted local air temperature to the MRPI coils less than the evaluated temperature for the system and coils of 213°F.

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

The cables from the control rod drive cabinets to the control rod drive operating coil stacks and the control rod drive mechanism cooling system are within the scope of license renewal. and are evaluated in the License Renewal Safety Evaluation Report, NUREG-1786, Section 2.5.1.2. The Control Rod Drive system power supply distribution is evaluated in SER Section 2.5.1.2, Low Voltage Insolated Cables and Connectors, and Section 2.5.1.4, Electrical Phase Bus. The control rod drive mechanism cooling system is described in Section 2.3.3.9, Containment Ventilation.

EPU does not require any new components or introduce any new functions for existing control rod drive system components that would require revision of the license renewal system evaluation boundaries. The operation of the control rod drive and control rod drive mechanism cooling systems at EPU conditions does not result in 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.

2.8.4.1.3.4 Results

The evaluation determined the expected additional heat load associated with expected higher reactor head temperatures associated with the EPU and demonstrated that the design temperature in the electro-magnetic coils, used to move the control rods, was not exceeded. This calculation has determined that the maximum expected electro-magnetic coil temperature is 348°F. This remains well below the electro-magnetic coil design limit of 392°F.

The estimated increase in containment heat load of 211,000 BTU/hr from the CRDM cooling system was evaluated in the evaluation of the containment cooling system described in LR section 2.7.7, Other Ventilation Systems (Containment) and found to be acceptable.

With a containment temperature of 120°F, dual or single CRDM cooling system fan operation will maintain the maximum predicted local air temperature to the MRPI coils less than the evaluated temperature for the system and coils of 213°F. Therefore the increased temperatures associated with EPU on the components and coils of the MRPI remains acceptable.

Ginna has reviewed the functional design of the control rod drive system and the CRDM cooling system for the effects of EPU. Accident and Transient Analyses described in LR section 2.4.2, Plant Operability, and LR section 2.8.5, Accident and Transient Analyses, have demonstrated that at EPU the rod control system will continue to satisfy the design basis for reactivity control and ensure specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems provided minor changes to the rod control system power mismatch variable gains described in LR section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

The impact of the EPU on the structural integrity of the Control Rod Drive Mechanisms (CRDMs) is discussed in LR section 2.2.2.4, Control Rod Drive Mechanism. The impact of EPU NSSS Transients is discussed in LR section 2.2.6, NSSS Design Transients. No modifications have been made to the hardware, logic or operation of the system that affect the system's current ability to fail into a safe state

The impact of EPU on the Control Rod Drive Cooling System was evaluated and determined to be capable of maintaining the operating coils and the MRPI system components and coils within acceptable system limits.

The increased containment heat load from the CRDM cooling system has been evaluated as part of the containment cooling system is discussed in LR section 2.7.7, Other Ventilation Systems (Containment) and determined to be acceptable.

2.8.4.1.4 Conclusions

The Ginna staff has reviewed the analyses related to the effects of the proposed EPU on the functional design of the control rod drive system and the control rod drive mechanism cooling system. The Ginna staff concludes that the evaluation has adequately accounted for the effects of the proposed EPU on the systems and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The Ginna staff further concludes that the evaluation has demonstrated that there is sufficient cooling to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU. Based on this, the Ginna staff concludes that the control rod drive system and the control rod drive mechanism cooling system will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-4, -23, -25, -26, -27, -28, and -29 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the functional design of the control rod drive system.

2.8.4.2 Overpressure Protection During Power Operation

2.8.4.2.1 Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary during power operation is provided by relief and safety valves and the reactor protection system. The Ginna Nuclear Power Plant, LLC (Ginna) review covered pressurizer relief and safety valves and the piping from these valves to the quench tank (pressurizer relief tank).

The NRC's acceptance criteria are based on:

- GDC-15, insofar as it requires that the reactor coolant system and associated auxiliary, control, and protection systems be 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.

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

Specific review criteria are contained in the SRP, Section 5.2.2, and other guidance provided in Matrix 8 of RS-001.

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 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 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 overpressure protection during power operation relative to conformance to:

- GDC-15 is described in 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, control, and protection systems were designed with sufficient margins so that design conditions are not exceeded during MODES 1 and 2 including anticipated operational occurrences. The normal operating pressure is 2235 psig with design pressure being 2485 psig. This provides a reasonable range for maneuvering during operation with allowance for pressure transients without actuation of the safety valves. Overpressurization is prevented by a combination of automatic control and pressure relief devices. The pressurizer safety valves (2485 psig setpoint) and pressurizer power operated relief valves (2335 psig setpoint) prevent overpressuring the reactor coolant system (RCS) during operation at rated power. Cold overpressure protection of the RCS is provided by the pressurizer power operated relief valves (PORV). The PORV lift setting is switched to Low Temperature Overpressure Protection (LTOP) control (lift setting 410 psig) prior to reducing RCS temperature below 330 system in service. Overpressure protection of the reactor coolant system (RCS) during normal operation is further discussed in UFSAR section 5.2.2.1.

- GDC-31 is described in UFSAR section 3.1.2.4.2, General Design Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary. As described in this UFSAR section, the RCPB was fabricated, inspected and tested in accordance with codes (i.e., ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping) that were applicable at the time of fabrication and installation. An evaluation of the Ginna reactor vessel concluded that the Ginna vessel met the ASME, Section III, fracture toughness requirements (see UFSAR Section 5.3.1.2). Overpressure protection of the RCPB is further discussed in UFSAR section 5.1.3.5.

In addition to the evaluations described in the UFSAR, the Ginna reactor coolant pressure boundary overpressure protection 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 for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004.

2.8.4.2.2 Technical Evaluation

2.8.4.2.2.1 Introduction

This section briefly summarizes the analyses documented in two other sections LR section 2.8.5.2.1 and LR section 2.8.5.3.2. These sections are for Loss of Load and Locked Rotor which are the limiting (in terms of overpressurization) Condition II and Condition IV events, respectively. The pre- and post-EPU margin discussion for each of those events is contained in their respective sections.

The limiting Condition II event with respect to primary and secondary system overpressurization is the loss-of-external-electrical-load/turbine-trip (LOL/TT) event. Details of the LOL/TT analysis performed for Ginna Station in support of the EPU are given in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum. The LOL/TT analysis documented in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum demonstrates that the primary and secondary pressures limits are met.

The limiting Condition IV event with respect to primary system overpressurization is the locked-rotor event. See LR section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break, for details of the locked-rotor analysis performed in support of the EPU.

The technical evaluation of the piping from the safety valves to the Pressurizer Relief Tank (PRT) is included in LR section 2.5.2, Pressurizer Relief Tank.

Note that overpressure protection during low temperature operation is discussed in LR section 2.8.4.3, Overpressure Protection During Low Temperature Operation.

2.8.4.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

For Condition II events, primary pressures must remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfy the requirements of GDC-15.

For Condition IV events, primary pressures must remain below 120% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfy the requirements of GDC-31.

2.8.4.2.2.3 Description of Analyses and Evaluations

The limiting Condition II event with respect to primary system overpressurization is the loss-of-external-electrical-load/turbine-trip (LOL/TT) event. Details of the LOL/TT analysis performed for Ginna Station in support of the EPU are given in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum. The LOL/TT analysis documented in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum demonstrates that the primary and secondary pressures limits are met. Specifically, the maximum pressure in the primary system is 2746.8 psia vs. a limit of 2748.5 psia (110% of design).

The limiting Condition IV event with respect to primary system overpressurization is the locked-rotor event. See LR section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break, for details of the locked-rotor analysis performed in support of the EPU. For locked rotor, the peak primary system pressure is 2782 psia compared to a limit of 2997 psia (120% of design).

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 components used to provide overpressure protection. The aging evaluations approved by the NRC in NUREG-1786 for these components remain valid for EPU conditions. See LR section 2.8.5.2.1 and LR section 2.8.5.3.2.

2.8.4.2.3 Results

No changes were needed to the primary or secondary relief or safety valves in order to meet the applicable pressure limits. All field setpoints and flow capacities remain unchanged. (see LR section 2.8.5.2.1 and LR section 2.8.5.3.2). The primary safety valve Technical Specification tolerance was reduced slightly from +2.4% / -3.0% to +2.3% / -3.0%, as discussed in the attached license amendment request to LCO 3.4.10, Pressurizer Safety Valves (see LR section 2.8.5.2.1 and LR section 2.8.5.3.2).

The analyses described in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum and LR section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break, demonstrate that the applicable pressure limits continue to be met for Ginna Station at EPU conditions. No changes to any control or protection setpoints or any valve capacities were necessary to meet these limits.

2.8.4.2.4 Conclusions

The Ginna staff has reviewed the analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The Ginna staff concludes that the analyses have:

- Adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features, and

- Demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded

Based on this, the Ginna staff concludes that the overpressure protection features will continue to provide adequate protection to meet the Ginna station current licensing basis requirements with respect to GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Overpressure Protection During Low Temperature Operation

2.8.4.3.1 Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary (RCPB) during low temperature operation of the plant is provided by pressure-relieving systems that function during the low temperature operation. The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered reactor coolant system (RCS) relief valves with piping to the pressurizer relief tank (quench tank), the charging (make-up) and letdown system, and the residual heat removal (RHR) system which may be operating when the primary system is water solid.

The NRC's acceptance criteria are based on:

- GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences; and,
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Specific review criteria are contained in NRC Standard Review Plan (SRP) section 5.2.2.

Ginna Current Licensing Basis

As noted in Ginna UFSAR 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 August1983. 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 RCPB overpressure protection during low temperature operation (LTOP) relative to conformance to:

- GDC-15 is described in Ginna UFSAR 3.1.2.2.6, General Design Criterion 15 – Reactor Coolant System Design. As described in UFSAR sections 3.2.2.1.1 and 5.1.3.5, low temperature overpressure protection is provided to the RCPB by the LTOP system. Operation of the LTOP system is discussed in UFSAR sections 5.2.2.2. and 7.6.1. The LTOP system is designed to prevent reactor vessel pressure in excess of 10CFR50, Appendix G limits (ASME Code Case N-514 limits).

- GDC-31 is described in Ginna UFSAR 3.1.2.4.2, General Design Criterion 15 – Fracture Prevention of RCPB. The components of the RCPB were fabricated, tested, and inspected in accordance with codes that were applicable at the time of fabrication and installation. The fracture toughness of the reactor vessel over the life of the plant is predicted using heat-up and cool-down curves which use conservative values for nil ductility transition temperature. Operating limitations during plant start-up and shutdown were evaluated in accordance the fracture toughness rules of the ASME Section III Code. Reactor vessel integrity was evaluated by SEP Topic V-6.

The Ginna LTOP System consists of the following two modes of over-pressure protection in Modes 4, 5 and 6 as specified by Ginna Technical Specification 3.4.12:

- Two power operated relief valves (PORVs) with lift settings within the limits specified in the Pressure and Temperature Limits Report (PTLR) and no safety injection (SI) pump capable of injecting into the RCS or,

- RCS depressurized and an RCS vent of 1.1 square inches and a maximum of one SI pump capable of injecting into the RCS.

The Ginna automatic LTOP system that utilizes the two PORVs was originally installed as a result of an NRC request to Westinghouse PWRs in 1976 (reference 1) to prevent over-pressurization events in operating plants. The Ginna LTOP system design was based on a Reference Mitigating System developed by Westinghouse and Westinghouse Owners Group to address the specific NRC concerns. Additionally, as required by Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," (reference 2), Ginna re-evaluated the effect of neutron radiation on reactor vessel material using the methods described in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. Based on the pressure-temperature limits resulting from the implementation of Regulatory Guide 1.99, Revision 2, the Ginna LTOP system setpoints were re-evaluated. As a result of SG replacement in the mid 1990s, the LTOP setpoints were re-evaluated again due to the effects of the larger replacement steam generators. This re-analysis incorporated ASME Code Case N-514 in the determination of the limiting reactor vessel pressure limits in accordance with the requirements of 10CFR50, Appendix G.

For a de-pressurized RCS providing a vent area of 1.1 square inches has also been shown to protect the RCS and the reactor vessel from exceeding the 10CRF50. Appendix G pressure limits as amended by ASME Code Case N-514.

The two redundant PORVs or a depressurized RCS with an open RCS vent area are also sufficient to protect the RHR system during the RHR mode of operation for events which cause an increase in system pressure.

2.8.4.3.2 Technical Evaluation

Existing Design Basis Requirements

Low temperature reactor vessel overpressure protection is provided automatically by the two pressurizer PORVs, as described in UFSAR Section 5.4.10, with a low-pressure setpoint as specified in the PTLR. As described in UFSAR section 7.6.1, the LTOP circuitry for low pressure PORV actuation circuitry uses multiple pressure sensors, power supplies and logic trains to improve system reliability.

Whenever the RCS cold leg temperature is below the temperature setpoint specified for LTOP in the PTLR or the RHR system is in operation, the low-pressure PORV setpoint is manually enabled from the control room. Pressure transients caused by mass addition or heat addition are terminated below the limits of 10CFR50, Appendix G, as amended by ASME Code Case N-514, by automatic operation of the pressurizer PORVs. The system is designed to protect the RCS pressure boundary from the effects of operating errors during MODES 4, 5, and 6 (as applicable in the Technical Specifications) when the RCS is in a water-solid condition. The system also supplies protection for the RHR system from over-pressurization.

The basic functional design requirements of the Ginna LTOP system as required by the NRC are summarized in UFSAR section 5.2.2.2.1. These requirements are:

- i) Automatic operation
- ii) Single failure proof
- iii) Be capable of periodic testing
- iv) Use equipment/components that comply with Seismic Category I requirements

A detailed description of the LTOP design and functional requirements is provided in UFSAR section 5.2.2.2.2.

Pre-EPU Analyses

The existing automatic LTOP system is designed to mitigate pressure transients which cause a rapid increase in RCS pressure when the RCS is in a water solid condition in Modes 4, 5 and 6. The types of transients evaluated for Ginna (reference 3) are divided into the following two categories:

- Mass input transients from injection sources such as charging pumps, safety injection pumps, or safety injection accumulators.
- Heat input transients from sources such as steam generators, decay heat or pressurizer heaters

For the automatic LTOP system the limiting mass addition transient for Ginna at the pre-EPU conditions is due to isolation of the letdown system with continued operation of all three charging pumps as described in UFSAR section 5.2.2.2.3.2. Since all Safety Injection (SI) pumps are required to be de-energized prior to enabling the PORVs, mass addition due to the start of an SI pump is not a credible event for the Ginna LTOP system. The limiting mass addition case was initialized at a primary temperature of 60°F and a primary pressure of 315 psig. Two reactor coolant pumps (RCPs) were assumed running and the pressurizer was water solid. Since this scenario assumes that the RHR system was removing decay heat, the RHR system was not explicitly modeled. The analysis assumed 180 gpm of charging flow due to three pump operation. The analysis was run for ten minutes at which time operator action to terminate the transient was assumed.

As discussed in UFSAR section 5.2.2.2.3.2, the peak reactor vessel pressure for this case was 587.4 psia. The allowable pressure, based on ASME Code Case N-514, at 60°F was 608.7 psia. Therefore, there is 21.3 psi margin to the Appendix G acceptance criterion for the reactor vessel. The peak RHR pump discharge pressure for this case was 663.5 psia. Since the allowable RHR system pressure is 674.7 psia (110 % of design pressure), there is 11.2 psi margin to the acceptance criterion for the RHR system.

For the automatic LTOP system, the most limiting heat addition transient for Ginna at the pre-EPU conditions is the RCS heat input transient associated with a restart of one RCP with the steam generator (SG) secondary side and primary side water temperature 50°F hotter than the rest of the RCS. Based upon analyses performed for the Westinghouse Owners Group (WOG), this type of transient is a significantly more rapid transient than that obtained due to either inadvertent actuation of pressurizer heaters or to loss of all RHR cooling. Since these two types of transients are slow when compared to the RCP start transient, they are not considered significant transients for the design of the LTOP system.

Therefore, as discussed in UFSAR section 5.2.2.2.3.3, the most limiting heat addition transient for Ginna at the pre-EPU conditions is due to an RCP start with the RCS at 60°F and 315 psig and the secondary system and primary side of the steam generator at 110°F. This case provides the least margin to the Appendix G acceptance criteria for the reactor vessel. Initially, the pressurizer was water solid, the RCPs were not running and RCS cooling was being provided by the RHR system. The event was initiated by starting the

RCP in the loop that contained the pressurizer.

The peak pressure in the reactor vessel for this case was 551.3 psia. The allowable pressure limit according to ASME Code Case N-514 at 60°F is 608.7 psia. Therefore, there is 57.4 psi margin to the Appendix G acceptance criterion for the reactor vessel. The peak RHR pump discharge pressure for this case was 650.0 psia. Since the allowable RHR system pressure is 674.7 psia (110 % of design pressure), there is 24.7 psi margin to the acceptance criterion for the RHR system.

The start of a RCP with the RCS at 320°F and the secondary system and primary side of the steam generator at 370°F was also analyzed in UFSAR section 5.2.2.2.3.4. For this case the peak pressure in the reactor vessel was 563.8 psia which is higher than the peak pressure obtained for the 60°F case. However, due to the higher RCS temperature the reactor vessel Appendix G pressure limit is 1529.4 psia. Therefore, appreciable margin to the Appendix G limit exists at this higher RCS temperature. The peak RHR system pressure for this case is 655.7 psia which is slightly higher than the peak RHR pressure calculated for the 60°F case. Since the allowable RHR system pressure is 674.7 psia (110 % of design pressure), there is 19.0 psi margin to the acceptance criterion for the RHR system for the 320°F RCS temperature case.

Based upon the results of the mass addition and heat addition transients performed for Ginna with the pre-EPU conditions, the mass addition transient associated with letdown isolation and operation of three charging pumps represents the limiting condition for the automatic LTOP system operation for both the reactor vessel and the RHR system.

For times when the RCS is de-pressurized and the automatic LTOP system is not available, low pressure protection is provided by a passive system that requires a minimum RCS vent area of 1.1 in^2. As discussed in UFSAR section 5.4.5.3.2.2, this minimum vent area has been determined to provide over-pressure protection for both the RCS and the RHR system from the most limiting mass addition scenario associated with the inadvertent operation of an SI pump. This mass addition case bounds the mass addition that is capable from the three operating charging pump transient analyzed for the automatic LTOP system.

Mass addition analyses for the start-up of one SI pump with a de-pressurized RCS and a 1.1 in^2 vent on the pressurizer have been performed (reference 3). Transients for RCS temperatures of both 60°F and 212°F were evaluated to determine the maximum RCS and RHR system pressures. The results from both analyses determined that the peak RCS and RHR pressures for both cases were well below the results obtained for the automatic LTOP mass addition case associated with three charging pump operation.

The maximum RCS pressure calculated for an LTOP event is 587.4 psig for the limiting mass addition case. Since the design pressure for the piping from the RCS relief valves to the pressurizer relief tank (quench tank) is 600 psig, this piping is unaffected by LTOP events.

Impact of EPU on LTOP Analyses

Since the limiting mass addition case involves a pump start event, increased decay heat has no impact on the RCS transient response. The mass addition case assumes that RHR is operating to remove decay heat prior to the limiting mass addition scenario resulting from the operation of three charging pumps delivering 180 gpm to the RCS. Therefore, the resulting RCS pressure transient response for the mass addition case is unaffected by the EPU.

Although decay heat during low RCS temperature operation in Modes 4 and 5 will increase due to EPU, the limiting LTOP heat addition case associated with a RCP start is not affected. The analyses performed for the Westinghouse Owners Group (WOG) demonstrated that a RCP start with a 50°F temperature difference between the RCS and the secondary side of a SG is the most limiting heat addition event for Westinghouse plants. The RCS pressurization associated with a loss of RHR cooling results in an appreciably slower RCS pressurization rate than that experienced from a RCP pump start. Therefore, although the EPU will cause a slight increase in the RCS pressurization rate due to a loss of RHR cooling heat addition event, the RCS pressurization rate from a RCP start still remains the limiting heat addition case. Since the RHR is operating to remove decay heat prior to the RCP start, the transient analysis for the RCP start limiting heat addition case is unaffected by the EPU.

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

In addition to the evaluations described in the UFSAR, the Ginna Station's LTOP 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 components for which the LTOP system provides a protective function are evaluated within the system that contains them which is as described in section 2.3.1, "Reactor Systems." Aging effects and programs used to manage the aging effects are discussed in section 3.1, "Reactor Coolant Systems." Since the existing limiting mass addition and heat addition transients are unaffected by the EPU, the peak reactor vessel and RHR system transients presently reported in UFSAR remain valid for the period of extended operation of the plant. Based on the EPU evaluation, no new aging effects requiring management are identified for the period of extended operation of the plant.

2.8.4.3.3 Results

Based on the foregoing, neither the existing limiting mass addition nor the existing limiting heat addition transients for the Ginna LTOP system are affected by changes in decay heat. Both transients assume that RHR is operating to remove decay heat, and the initiator for the RCS pressure transient is not impacted by decay heat. Since the loss of RHR cooling heat addition transient is a slow developing transient, the impact of increased decay heat due to the EPU can be accommodated by the existing LTOP system without any changes to the existing plant LTOP setpoints. Additionally, the existing requirement for a 1.1 square inch RCS passive vent area when LTOP is not in service with a depressurized RCS is also unchanged by EPU. Its size is based on the bounding mass addition case for a depressurized RCS associated with a start of one SI Pump which is unaffected by the EPU.

Since the existing limiting mass addition and heat addition transients are unaffected by the EPU, the peak reactor vessel and RHR system transients presently reported in UFSAR sections 5.2.2.2.3 are still valid. Therefore, sufficient margin exists for the reactor vessel, the RHR system and the piping from the RCS relief valves to the pressurizier relief tank to comply with the requirements of GDC-15 and GDC-31. Additionally, no changes to existing LTOP setpoints, administrative controls, testing or inspections requirements are required due to the EPU. Finally, the ability of the LTOP System to provide pressure relief for 10 minutes without any operator actions is still maintained.

2.8.4.3.4 References

1. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," July 12, 1988.

2. NRC Letter to Westinghouse PWR Utilities, "Summary of Meeting Held on November 4, 1976 Concerning Proposed Measures to Prevent Reactor Vessel Overpressurization in Operating Westinghouse (PWR) Facilities", November 17, 1976.

3. B&W Nuclear Technologies, LTOP Report for Ginna, 86-1234820-03, approved September 19, 1997

2.8.4.3.5 Conclusion

The Ginna staff has reviewed the analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during low temperature operation. The Ginna staff concludes that:

(1) The analyses adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and

(2) The plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.

Based on this, the Ginna staff concludes that the low temperature overpressure protection features will continue to provide adequate protection to meet the Ginna Station current licensing basis requirements with respect to GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU is acceptable with respect to overpressure protection during low temperature operation.

2.8.4.4 Residual Heat Removal System

2.8.4.4.1 Regulatory Evaluation

The residual heat removal (RHR) system cools down the reactor coolant system following shutdown. The residual heat removal system is typically a low-pressure system that takes over the shutdown cooling function when the reactor coolant system temperature is reduced. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the reactor coolant system following shutdown and provide decay heat removal.

The NRC's acceptance criteria are based on:

- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be protected against dynamic effects
- GDC-5, insofar as it requires that important-to-safety structures, systems, and components 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, which specifies requirements for a residual heat removal system.

Specific review criteria are contained in SRP, Section 5.4.7 and other guidance provided in Matrix 8 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 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 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 Ginna Station Residual Heat Removal System 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 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)
- GDC–5 is described in UFSAR section 3.1.2.1.5, General Design Criterion 5 – Sharing of Structures, Systems, and Components. As described in this UFSAR section, the Ginna Nuclear Power Plant is a single unit installation.
- GDC–34 is described in UFSAR section 3.1.2.4.5, General Design Criterion 34 – Residual Heat Removal. As described in this UFSAR section, the Ginna RHR system, in conjunction with steam power conversion system, is designed to transfer the fission product decay heat and other residual 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 with the two residual heat removal pumps and two heat exchangers. The RHR system is able to operate on either onsite or offsite power systems. Details of the system design are given in UFSAR section 5.4.5. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System" is addressed in UFSAR section 5.4.5.3.

In addition to the evaluations described in the UFSAR, the Ginna Station's RHR 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.

2.8.4.4.2 Technical Evaluation

2.8.4.4.2.1 Introduction

The RHR System is described in Ginna UFSAR section 5.4.5. The system is designed to remove residual and sensible heat from the core and reduce the temperature of the reactor coolant system during the second phase of plant cool down. During the first phase of cool down, the temperature of the reactor coolant system is reduced by transferring heat from the reactor coolant system to the steam and power conversion system.

2.8.4.4.2.2 Description of Analyses and Evaluations

The EPU increases the residual heat generated in the core during normal cooldown, refueling operations and accident conditions. This provides a higher heat load on the RHR Heat Exchangers (HXs) during cooldown and also during refueling outages. The removal of core decay heat for accident conditions is addressed in LR section 2.6.5, Containment Heat Removal. The increased heat loads will be transferred to the Component Cooling Water System (CCWS) and ultimately to the Service Water System (SWS). Evaluation of the EPU performance of the RHRS in conjunction with the CCWS and SWS with the increased heat loads is addressed in this subsection, LR section 2.5.4.3, Reactor Auxiliary Cooling Water Systems, and LR section 2.5.4.2, Service Water System.

The EPU affects the plant cooldown time(s) since core power, and therefore the decay heat, increases. The plant cool down calculation was performed at a core power of 1811 MWt to support the EPU (LR section 1.1, Nuclear Steam Supply System Parameters, Table 1-1). The RCS heat capacity and the other CCWS heat loads were explicitly considered in these analyses. The analysis was performed to demonstrate that the RHR and CCW systems continue to comply with their design basis functional requirements and performance criteria for plant cooldown under the EPU conditions. The two-train system alignment was considered to address the design capability in the Ginna UFSAR. In addition, a cooldown analysis was performed to support the worst-case scenario for the 10CFR50, Appendix R fire hazards and safe shutdown analysis. Also, analysis was performed to demonstrate that existing technical specification cooldown time limits will be achieved at EPU conditions.

The following considerations were applied to these cooldown analyses:

- The CCW and RHR HX data assumes design tube plugging of 9-percent and 7.5-percent respectively. This results in slightly degraded cooldown performance.
- The CCW and RHR HX data assumes design fouling factors.
- The design service water temperature of 85°F was assumed.
- The CCWS supply temperature is limited to 120°F during cooldown.
- Decay heat curves were based on 24-month fuel cycles.

The normal plant cooldown time to 140°F for refueling (Mode 6) or cold shutdown maintenance (Mode 5) with both trains of CCW and RHR available (i.e. two RHR pumps and Heat exchangers & two CCW pumps and heat exchangers) increased from 64-hours for the current power rating to 110-hours for the EPU assuming a normal cooldown start time of 4 hrs after reactor shutdown. The normal plant cooldown time to cold shutdown (Mode 5 - ≤200°F) with both trains of CCW and RHR available increased from 11 hours for the current power rating to 15 hours for the EPU. Since there is no design criterion for normal plant cooldown time, these increases in calculated values, based on design conditions, are acceptable .

The Appendix R/safe shutdown requires that cold shutdown (Mode 5 - ≤200°F) be achieved in 72-hours after reactor shutdown. Continued compliance with this time limit was demonstrated at the EPU conditions. The worst case cooldown scenario assumes loss-of-offsite power and only one train of RHR and CCW equipment available. At EPU conditions, one train of RHR and CCW equipment can match the cooldown heat load 11-hours after reactor shutdown and achieve cold shutdown 48-hours after reactor shutdown. This is the minimum time required to achieve cold shutdown. The maximum time limit of 72-hours would require that RHR cooldown be initiated no later than 60-hours after reactor shutdown. The first phase of plant cooldown must be accomplished with the steam system atmospheric relief valves. For the worst case only one main steam atmospheric relief valve is assumed to be available and under natural circulation conditions cooldown to the RHRS cut-in conditions can be achieved in 40-hours. This provides a 20-hour margin in terms of the time required to initiate RHRS cooldown and achieve the 72-hours cooldown time limit. Therefore, continued compliance with the Appendix R cold shutdown requirement within the 72-hour time was demonstrated at EPU conditions with no plant changes required.

Analysis was also performed to demonstrate that continued compliance with all technical specification cooldown time limits will be achieved at EPU conditions. The plant technical specifications require that the plant be in hot shutdown (Mode 3) within 6-hrs and cold shutdown (Mode 5) within 36-hrs with required equipment for power operation out of service. With both trains of RHR and CCW equipment, cold shutdown can be achieved in 15-hours at EPU conditions if RHRS operation is initiated 6-hours after reactor shutdown. For the worst case scenario, that is, loss of RCPs coupled with the loss of one RHR pump and one CCW pump cold shutdown will be achieved in 28-hours after reactor shutdown if RHR operation is initiated 6-hours after reactor shutdown. Therefore, continued compliance with the Technical Specification cooldown time requirements was demonstrated at the EPU conditions.

The EPU does not impact the design temperature and pressure of the RHRS piping and associated components. Refer to LR section 2.2.2.1, NSSS – Piping and Supports (Class 1), LR section 2.2.2.2, Balance of Plant Piping and Supports (Non Class 1), and LR section 2.5.1.3, Pipe Failures for the RHRS piping evaluation and the environmental and dynamic effects evaluation relative to meeting the Ginna Station current licensing basis requirements with respect to GDC 4.

The EPU has no affect on the ability of the RHRS to remove residual heat at reduced reactor coolant system inventory, and therefore the Ginna Station will continue to meet the current licensing basis requirements with respect to NRC Generic letter 88-17. Additional discussion of NRC Generic letter 88-17 is provided in LR section 2.8.7.3, Loss of Residual Heat Removal at Midloop.

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

The Ginna Station's RHRS 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 RHRS is described in the License Renewal SER, Section 2.3.2.3. The programs used to manage the aging effects associated with Engineered Safety Features Systems are discussed in Section 3.2 of the Ginna Licensing Renewal submittal.

EPU activities do not add any new components nor do they introduce any new functions for existing components of the RHR system that would change the license renewal system evaluation boundaries. The changes associated with operating the RHR 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. Thus no new aging effects requiring management are identified.

2.8.4.4.3 Results

Continued compliance with the RHRS cooldown performance requirements was demonstrated at the EPU conditions with no plant changes being necessary. The EPU cooldown analyses results are as follows:

- The Normal plant cooldown time to cold shutdown (Mode 5 - ≤200°F) with both trains of RHR and CCW equipment in service will increase from 11-hours to 15-hours. The normal plant cooldown time to 140°F for refueling (Mode 6) or cold shutdown maintenance will increase from 64-hours to 110-hours. Since there are no design criteria for normal plant cooldown times, these increases in calculated values, based on design conditions, are acceptable.

- For the Appendix R/safe shutdown cooldown scenario, cold shutdown (Mode 5 - ≤200°F) will continue to be achieved within the 72-hour time limit. The worst case fire scenario assumes loss of offsite power, one atmospheric dump valve and one train of RHR and CCW equipment. For this worst case scenario, the RHRS can be placed in service 40-hours after reactor shutdown which is 20-hours earlier than the required initiation time to achieve cold shutdown in 72-hours.

- Continued compliance with the technical specification time limits with respect to achieving hot shutdown (Mode 3 within 6-hours) and cold shutdown (Mode 5 within 36-hours) was demonstrated at the EPU conditions. With all RHR and CCW equipment available cold shutdown will be achieved approximately15-hours after reactor shutdown and with the loss of one RHR pump and one CCW pump cold shutdown will be achieved approximately 28-hours after reactor shutdown.

Evaluations described in LR section 2.2.2.1, NSSS – Piping and Supports (Class 1), LR section 2.2.2.2, Balance of Plant Piping and Supports (Non Class 1), and LR section 2.5.1.3, Pipe Failures show the response of the RHRS piping to the EPU environmental and dynamics effects remain acceptable relative to meeting the Ginna Station current licensing basis requirements with respect to GDC 4.

The EPU has no affect on the ability of the RHRS to comply with GDC 34 and the associated NRC Branch Technical Position RSB 5-1. Although RSB 5-1 was issued after the design of Ginna Station, UFSAR Section 5.4.5.3 provides a comparison of the Ginna design to these Guidelines. Basically, the EPU operating conditions have no adverse affect on the following:

- The design and operating characteristics of the RHRS with respect to its shutdown and long-term cooling function (Refer to the above evaluation relative to the RHRS cooldown performance at EPU conditions).

- The isolation provisions provided between the high pressure RCS and the lower pressure RHRS and the RHRS overpressure protection features. The design and operating pressures of the RCS and RHRS are not affected by the EPU.

2.8.4.4.4 Conclusion

The Ginna staff has reviewed the effects of the proposed EPU on the RHR system. The Ginna staff concludes that the effects of the proposed EPU on the system are adequately accounted for and it has been demonstrated that the RHR system will maintain its ability to cool the reactor coolant system following shutdown and provide decay heat removal. Based on this, the Ginna staff concludes that the RHR system will continue to meet Ginna Station current licensing requirements with respect to GDC-4, GDC-5, and GDC-34 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.5 Accident and Transient Analyses

2.8.5.0 Non-LOCA Analyses Introduction

This section summarizes the non-loss-of-coolant accident (LOCA) transient analyses and evaluations performed to support the EPU program at the Ginna Station.

2.8.5.0.1 Fuel Design Mechanical Features

The fuel currently in use at Ginna is the Westinghouse 14×14 optimized fuel assembly (OFA) with integral fuel burnable absorber (IFBA). In support of the EPU, Ginna will transition to Westinghouse VANTAGE+ 14×14 fuel with PERFORMANCE+ (422V+) fuel assemblies with IFBA. Detailed information on the 422V+ fuel design is provided in LR section 2.8.1. With respect to the non-LOCA transient analyses, the effects of fuel design mechanical features were accounted for in fuel-related input assumptions, such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

2.8.5.0.2 Peaking Factors

For the 422V+ fuel, the power distribution is characterized by a nuclear enthalpy rise hot channel factor (radial peaking, $F^N_{\Delta H}$) of 1.654 for analyses employing the *Revised Thermal Design Procedure* (RTDP)(Reference 1), and 1.720 for non-RTDP analyses, and a full-power heat flux hot channel factor (total peaking, F_Q) of 2.60. $F^N_{\Delta H}$ is important for transients that are analyzed for departure from nucleate boiling (DNB) concerns (Table 2.8.5.0-1 identifies which events are analyzed for DNB concerns, as well as the DNB methodology used, RTDP or non-RTDP). As $F^N_{\Delta H}$ increases with decreasing power level due to rod insertion, all transients analyzed for DNB concerns are assumed to begin with an $F^N_{\Delta H}$ consistent with the $F^N_{\Delta H}$ defined in the *Technical Specifications Core Operating Limits Report* (COLR) for the assumed nominal power level. The F_Q, for which the limits are specified in the COLR, is important for transients that are analyzed for overpower concerns, e.g., rod cluster control assembly (RCCA) ejection.

The minimum shutdown margin at hot zero power (HZP) conditions, with the most reactive RCCA fully withdrawn, is assumed to be 1.3% Δk/k. This was assumed in the HZP steam line break analysis.

2.8.5.0.3 EPU Program Features

Key EPU Program features that were considered in the non-LOCA transient analyses were as follows:

- A nuclear steam supply system (NSSS) power level of 1817 MWt (includes a net reactor coolant pump [RCP] heat of 6 MWt)
- 14×14 422V+ fuel with an increased fuel rod outer diameter of 0.422 inches
- A nominal, full-power reactor coolant vessel average temperature (T_{avg}) window between 564.6° and 576.0°F
- A reactor coolant system (RCS) thermal design flow (TDF) of 170,200 gpm (85,100 gpm/loop)
- Babcock & Wilcox (BWI) replacement steam generators (RSGs) with a maximum steam generator tube plugging (SGTP) of 10%
- A nominal operating pressurizer pressure of 2250 psia
- A design core bypass flow of 6.5%, which accounts for all thimble plugs removed
- A nominal, full-power main feedwater temperature window between 390°F and 435°F

For most transients that were analyzed for departure from nucleate boiling (DNB) concerns, the RTDP methodology (Reference 1) was employed. With this methodology, nominal values were assumed for the initial conditions of power, temperature, pressure, and flow, and the corresponding uncertainty allowances were accounted for statistically in defining the departure from nucleate boiling ratio (DNBR) safety analysis limit. The nominal RCS flow assumed in RTDP transient analyses was the minimum measured flow (MMF) of 177,300 gpm, and the difference between TDF and MMF was the applicable flow uncertainty.

As discussed in LR section 2.8.3, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions were combined statistically to obtain the overall DNB uncertainty factor, which was used to define the design-limit DNBR (1.22 for both typical and thimble cells). In other words, the design limit DNBR was a DNBR value that is greater than the WRB-1 DNB correlation limit (1.17) by an amount that accounted for the RTDP uncertainties. To provide DNBR margin to offset various penalties, such as those due to rod bow and instrument bias, and to provide flexibility in design and operation of the plant, the design limit DNBR was conservatively increased to a value designated as the safety analysis limit DNBR, to which transient-specific DNBR values were compared. The DNBR safety analysis limit selected for Ginna was 1.38 for both typical and thimble cells.

For transient analyses that were not DNB-limited, or for which RTDP is not employed, the initial conditions were obtained by applying the maximum, steady-state uncertainties to the nominal values in the most conservative direction; this is known as Standard Thermal Design Procedure (STDP), or non-RTDP. In these analyses, the RCS flow was assumed to be equal to the TDF, and the following steady-state initial condition uncertainties were applied:

- The maximum NSSS power is 1817 MWt with zero power uncertainty allowance.
- The T_{avg} allowance for deadband and system measurement uncertainties was ± 4°F.
- The pressurizer pressure allowance for steady-state fluctuations and measurement uncertainties was ± 60 psi.
- The RCS flow allowance is represented by the difference between TDF and MMF. The flow uncertainty is equivalent to 4%.

2.8.5.0.4 Other Major Assumptions

Table 2.8.5.0-2 lists the non-LOCA initial condition assumptions used. Other major assumptions considered in the non-LOCA transient analyses are discussed below:

- At least ± 1.4% setpoint tolerance was considered in modeling of the main steam safety valves (MSSVs). Staggered lift setpoints were modeled for the MSSVs using plant-specific Technical Specification setpoints, as shown in Table 2.8.5.0-3.
- The pressurizer safety valves (PSVs) were modeled assuming a setpoint tolerance range of at least +2.3% to -3.0%. Additionally, when it was conservative to do so (that is, for peak RCS pressure concerns), the effects of the PSV loop seals were explicitly modeled, as discussed in Reference 2. See Table 2.8.5.0-3 for more information.
- Consistent with the Ginna Technical Specifications (COLR), for minimum reactivity feedback a maximum moderator temperature coefficient (MTC) of +5 pcm/°F was applicable for power levels less than 70%, and a 0 pcm/°F MTC was applicable for power levels greater than or equal to 70%. For maximum

reactivity feedback, a maximum moderator density coefficient (MDC) of at least 0.45 Δk/g/cc was assumed.

- The fission product contribution to decay heat assumed in the non-LOCA analyses was consistent with the standard ANSI/ANS-5.1-1979 for decay heat power in light water reactors (Reference 3), including two standard deviations of uncertainty.

- The assumed core bypass flow percentages were 5.6% for RTDP analyses, and 6.5% for STDP analyses.

2.8.5.0.5 Overtemperature and Overpower ΔT Reactor Trip Setpoints

The overtemperature and overpower ΔT (OTΔT/OPΔT) reactor trip setpoints were recalculated using the methodology described in WCAP-8745-P-A (Reference 4). Conservative core thermal limits developed using the RTDP methodology (as described in LR section 2.8.3) were used to calculate the OTΔT and OPΔT reactor trip setpoints. Although the core thermal limits are currently based on a nominal core power of 1775 MWt, it was assumed that future DNBR margin gained with a full core of 422V+ fuel will allow the same core thermal limits to be applicable to a nominal core power of 1811 MWt. The assumed core thermal limits are presented in Figure 2.8.5.0-1. The OTΔT and OPΔT trip setpoints are illustrated in Figure 2.8.5.0-2 and presented in Table 2.8.5.0-4.

The adequacy of these setpoints was confirmed by showing that the DNB design basis is met in the analyses of those events that credit these functions for accident mitigation. The revised safety analysis setpoints were based upon the assumption that the reference average temperature (T′) used in the OTΔT and OPΔT setpoint equations was equal to the nominal full power T_{avg}.

The boundaries of operation defined by the OTΔT and OPΔT trips are represented as "protection lines" in Figure 2.8.5.0-2. The protection lines were drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. These protection lines are based upon the safety analysis limit OTΔT and OPΔT setpoint values, which are essentially the Technical Specification nominal values with allowances for instrumentation errors and acceptable drift between instrument calibrations. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line (ΔT versus T_{avg}). The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.38 for both typical and thimble cells). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit DNBR value.

The area of permissible operation (power, temperature, and pressure) was bounded by the combination of the high neutron flux (fixed setpoint), high- and low-pressurizer pressure (fixed setpoints), and OTΔT and OPΔT (variable setpoints) reactor trips, and the opening of the MSSVs, which limited the maximum RCS average temperature. The adequacy of the OTΔT and OPΔT setpoints was confirmed by demonstrating that the DNB design basis was met for those transients analyzed for DNB concerns.

As a result of the revised OTΔT and OPΔT setpoint equations, the temperature ranges provided below will be used for the resistance temperature detector (RTD) instrumentation. Changes will be reflected in the setpoints document for the plant.

- T_{cold}: 510°F – 590°F
- T_{hot}: 540°F – 650°F
- T_{avg}: 540°F – 620°F
- ΔT: 0°F – 85°F
- f(ΔI): -49% – 54%

2.8.5.0.6 RPS and ESFAS Functions Assumed in Analyses

Table 2.8.5.0-5 contains a list of the different reactor protection system (RPS) and engineered safety features actuation system (ESFAS) functions credited in the non-LOCA transient analyses. The safety analysis setpoints, as well as the time delays associated with each of these functions, are also presented in Table 2.8.5.0-5.

2.8.5.0.7 RCCA Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the non-LOCA transient analyses, the critical parameter was the time from beginning of RCCA insertion to dashpot entry, or approximately 85% of the RCCA travel, although negative reactivity addition continued to be modeled until rods were completely inserted. For the non-LOCA analyses, the assumed insertion time from fully withdrawn to dashpot entry was 1.8 seconds (based on full core flow), which is consistent with Technical Specification Surveillance Requirement 3.1.4.4.

Three figures relating to RCCA drop time and reactivity worth are presented in this report. The RCCA position (fraction of full insertion) versus the time from release is presented in Figure 2.8.5.0-3. The normalized reactivity worth assumed in the safety analyses is shown in Figure 2.8.5.0-4 as a function of rod insertion fraction and in Figure 2.8.5.0-5 as a function of time.

2.8.5.0.8 Reactivity Coefficients

The transient response of the reactor core is dependent on reactivity feedback effects, in particular the MTC and the Doppler power coefficient (DPC). Depending upon event-specific characteristics, conservatism dictates the use of either maximum or minimum reactivity coefficient values. Justification for the use of the reactivity coefficient values was treated on an event-specific basis. Table 2.8.5.0-6 presents the core kinetics parameters and reactivity feedback coefficients assumed in the non-LOCA analyses.

The maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 2.8.5.0-6. Note that the HZP steam line break core response analysis used a different DPC, which was based on an RCCA being stuck out of the core (not shown in Figure 2.8.5.0-6).

2.8.5.0.9 Computer Codes Utilized

Summary descriptions of the principal computer codes used in the non-LOCA transient analyses are provided below. Table 2.8.5.0-7 lists the computer codes used in each of the non-LOCA analyses.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod, and the transient heat flux at the surface of the cladding, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature, and density. The code uses a fuel model that simultaneously contains the following features:

- A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accident
- Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel

The FACTRAN licensing topical report, WCAP-7908-A (Reference 5), was approved by the NRC via a Safety Evaluation Report (SER) from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse), dated September 30, 1986. The FACTRAN SER identifies seven conditions of acceptance, which are summarized and discussed in LR Appendix A.

RETRAN

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot- and cold-leg piping, RCPs, steam generators (tube and shell sides), main steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves can also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The RPS simulated in the code includes reactor trips on high neutron flux, high neutron flux rate, OTΔT and OPΔT, low RCS flow, high- and low-pressurizer pressure, high pressurizer level, and low-low steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, can also be modeled. RETRAN conservatively approximates the transient value of DNBR based on input from the core thermal safety limits.

The RETRAN licensing topical report, WCAP-14882-P-A (Reference 6), was approved by the NRC via an SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse), dated February 11, 1999. The RETRAN SER identifies three conditions of acceptance, which are summarized and discussed in LR Appendix A.

During licensing planning meetings to discuss the Ginna EPU licensing submittal, the NRC staff verbally communicated two items of concern regarding the use of RETRAN. The first item was whether or not two-loop RETRAN results were compared to operational data and/or LOFTRAN code results. The second item was in regard to the ability of RETRAN to address water-solid pressurizer conditions. As it turns out, these two items were previously addressed by Westinghouse in responses to NRC requests for additional information (RAIs) in support of WCAP-14882-P-A. The original responses to these RAIs are included in Westinghouse letters to the NRC attached in Appendix B of WCAP-14882-P-A (see Question 2 in letter NSD-NRC-98-5765 and Question 5 part o in letter NSD-NRC-98-5809). Key points of interest from the two responses along with additional supporting information are provided as follows.

Item 1 – Benchmarking

- WCAP-14882-P-A (Reference 6) documents RETRAN benchmarks for a Westinghouse four-loop plant against operational data and LOFTRAN.

- The only significant effect of the number of loops is the mixing in the reactor vessel upper and lower plenums.

- The results presented in WCAP-14882-P-A for all the events, in particular the asymmetric events, sufficiently demonstrated the capability of the Westinghouse model to handle the independent behavior of the loops and the thermal-hydraulic interactions between loops. This was also true for the two-loop model.

- RETRAN results compared favorably to LOFTRAN results, and LOFTRAN results have been compared to a wide variety of plant operational and test transients for two-, three-, and four-loop plants.
- Westinghouse performed RETRAN analyses for the following plants in support of a variety of plant licensing amendments that have been approved by the NRC: South Texas Project (four-loop), J. M. Farley (three-loop), Kewaunee (two-loop), Prairie Island (two-loop).
- For the J. M. Farley plant, Westinghouse performed equivalent LOFTRAN analyses to provide additional verification of the consistency of behavior of the results of the codes.
- No additional two-loop benchmarking is planned to support the Ginna EPU.

Item 2 – Pressurizer Filling

- It was shown in WCAP-14882-P-A (Reference 6), and it continues to be shown in more recent analyses performed by Westinghouse, that the RETRAN pressurizer model demonstrates adequate stability for filling conditions.
- Studies performed by the RETRAN code developer, Computer Simulation & Analysis, Inc. (CSA), in support of plant-specific RETRAN analysis submittals, have not shown any instabilities or discontinuities when transitioning from a two-phase condition to a single phase condition, which occurs when the pressurizer becomes water solid.
- In 1984, the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute (EPRI) contracted CSA to perform RETRAN studies of anticipated transient without scram (ATWS) tests based on experiments performed at the loss of fluid test (LOFT) facility at the Idaho National Engineering Laboratory. The ATWS events resulted in the filling of the pressurizer, and as has been demonstrated in the analysis of plant safety analyses, the RETRAN results were stable and showed no abnormalities or discontinuities that would lead one to conclude that it was not capable of handling such conditions. These studies were documented in EPRI report NSAC-78 (Reference 7).
- The general RETRAN-02 code SER "limitation" related to pressurizer filling does not exist for RETRAN-3D, although no code changes related to the

pressurizer were made. This supports the conclusion that the issue of filling the pressurizer (that is, the RETRAN code's ability to handle changes from a two-phase to a single-phase, or vice-versa) is not a concern or code limitation.

LOFTRAN

The LOFTRAN computer code is used only for anticipated transients without scram (ATWS) because RETRAN is not approved for use in ATWS events. This code simulates a multi-loop system by a model containing the reactor vessel, hot- and cold-leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the RPS, which includes reactor trips on high neutron flux, OTΔT and OPΔT, high- and low-pressurizer pressure, low RCS flow, low-low steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The SIS, including the accumulators, is also modeled. LOFTRAN can also approximate the transient value of DNBR based on input from the core thermal safety limits.

The LOFTRAN licensing topical report, WCAP-7907-P-A (Reference 8), was approved by the NRC via an SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), dated July 29, 1983. The LOFTRAN SER identifies one condition of acceptance, which is summarized and discussed in LR Appendix A.

TWINKLE

TWINKLE is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8000 spatial points and performs steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various outputs, such as channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

The TWINKLE licensing topical report, WCAP-7979-P-A (Reference 9), was approved by the U.S. Atomic Energy Commission (AEC) via an SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse), dated July 29, 1974. The TWINKLE SER does not identify any conditions, restrictions, or limitations that need to be addressed for application to Ginna.

Advanced Nodal Code (ANC)

ANC is an advanced nodal code capable of two-dimensional (2-D) and three-dimensional (3-D) neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, 3-D ANC validates 1-D and 2-D results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

The ANC licensing topical report, WCAP-10965-P-A (Reference 10), was approved by the NRC via an SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse), dated June 23, 1986. The ANC SER does not identify any conditions, restrictions, or limitations that need to be addressed for application to Ginna.

VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core.

The VIPRE licensing topical report, WCAP-14565-P-A (Reference 11), was approved by the NRC via an SER from T. H. Essig (NRC) to H. Sepp (Westinghouse), dated January 19, 1999. The VIPRE SER identifies four conditions of acceptance, which are summarized and discussed in LR Appendix A.

2.8.5.0.10 Classification of Events

Each of the non-LOCA events listed in Table 2.8.5.0-8 is presented in Section 15 of the *Ginna Updated Final Safety Analysis Report* (UFSAR). Each non-LOCA event is categorized with respect to its potential consequences. Since 1970, the classification of plant conditions in American Nuclear Society Standard ANSI N18.2-1973 (Reference 13) has often been used to facilitate the evaluation of nuclear plant safety and the functional requirements for structures, systems, and components. The plant conditions are divided into four categories in accordance with the anticipated frequencies of occurrence and potential radiological consequences. The four categories (or conditions) are:

- Condition I – Normal Operation
- Condition II – Faults of Moderate Frequency
- Condition III – Infrequent Faults
- Condition IV – Limiting Faults

The basic principle applied in relating requirements to each of the conditions is that the more probable occurrences must result in little or no risk to the public, and those extreme situations having the potential for greater risk should be those situations least likely to occur. Where applicable, the reactor trip system and/or engineered safety features are assumed in fulfilling this principle. Each condition is described in more detail as follows.

Condition I – Normal Operation

Condition I occurrences are those that are expected frequently or regularly during power operation, refueling, maintenance, or maneuvering of the plant. Condition I occurrences are accommodated with margin between any plant parameter and the value of the parameter that would require either automatic or manual protective action. In this regard, analysis of the fault condition is typically based on a conservative set of initial conditions corresponding to the most adverse set of conditions occurring during Condition I operation.

Condition II – Faults of Moderate Frequency

These faults occur with moderate frequency during the life of the plant, any one of which may occur during a calendar year (i.e., between 1/year and 1×10^{-1}/year). These faults, at worst, result in a reactor trip with the plant being capable of returning to operation after corrective action. Any release of radioactive materials in effluents to unrestricted areas should be in conformance with 10CFR20 (Reference 14). A Condition II fault (or event), by itself, does not propagate to a more serious incident of the Condition III or Condition IV type without the occurrence of other independent incidents. A single Condition II incident should not cause the loss of any barrier to the escape of radioactive products.

Condition III – Infrequent Faults

Condition III faults occur very infrequently during the life of the plant, any one of which may occur during the plant's lifetime (i.e., between 1×10^{-1}/year and 1×10^{-2}/year). Condition III faults can be accommodated with the failure of only a small fraction of the fuel rods, although sufficient fuel damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactivity due to Condition III faults may exceed the guidelines of 10CFR20, but is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary. A Condition III fault does not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers.

Condition IV – Limiting Faults

Condition IV occurrences are faults that are not expected to occur, but are postulated because their consequences have the potential for the release of significant amounts of radioactive material (i.e., $< 1\times10^{-2}$/year). Condition IV faults are the most drastic occurrences that must be designed against, and represent the limiting design cases. Condition IV faults should not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of the guideline values in 10CFR100 (Reference 15). A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the RCS and the reactor containment.

2.8.5.0.11 Events Evaluated or Analyzed

Each of the UFSAR transients listed in Table 2.8.5.0-1 were evaluated or analyzed as shown in Table 2.8.5.0-8 in support of the Ginna EPU Program. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria are satisfied for Ginna. Table 2.8.5.0-1 summarizes the results obtained for each of the non-LOCA transient analyses.

The non-LOCA analyses did not identify the need to add any new system components to achieve accident mitigation. (Some analyses credit a new feedwater isolation configuration that was submitted for NRC review under a previous license amendment request (Reference: letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), *License Amendment Request Regarding Main Feedwater Isolation Valves*, dated April 29, 2005). Additional details of this change are provided in the appropriate accident analysis). Because the non-LOCA analyses did not require the addition of new components, or require existing components to function in a way not previously evaluated, the license renewal evaluation boundaries are not changed. The results of the non-LOCA analyses show that components credited for the accident analysis will be exposed to the same internal and external environments as previously evaluated. Because there is no change to components, component materials of construction, and internal or external operating environments required to mitigate non-LOCA accidents, no new aging effects requiring management have been identified. Accordingly, there is no impact on plant license renewal as a result of the non-LOCA analyses performed in support of the Ginna EPU.

2.8.5.0.12 Analysis Methodology

The transient-specific analysis methodologies that were applied to Ginna have been reviewed and approved by the NRC via transient-specific topical reports (WCAPs) and/or through the review and approval of plant-specific safety analysis reports. There are only two non-LOCA transients analyzed for Ginna that have a transient-specific topical report applicable to Ginna: RCCA ejection (Ginna UFSAR, Section 15.4.5) and dropped rod (Ginna UFSAR, Section 15.4.6).

The dropped rod licensing topical report, WCAP-11394-P-A (Reference 16), was approved by the NRC via an SER from A. C. Thadani (NRC) to R. A. Newton (Westinghouse Owner's Group), dated October 23, 1989. The dropped rod SER identifies one condition of acceptance, which is summarized below along with justification for application to Ginna.

> *"The Westinghouse analysis, results and comparisons are reactor and cycle specific. No credit is taken for any direct reactor trip due to dropped RCCA(s). Also, the analysis assumes no automatic power reduction features are actuated*

by the dropped RCCA(s). A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."

Justification

For the reference cycle assumed in the Ginna EPU Program, it is affirmed that the methodology described in WCAP-11394-P-A was performed and the required comparisons have been made with acceptable results (DNB limits are not exceeded). Cycle-specific confirmation will be performed as part of the normal reload evaluation (Reference 17).

The RCCA ejection licensing topical report, WCAP-7588 Rev. 1-A (Reference 18), was approved by the AEC via an SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse), dated August 28, 1973. The RCCA ejection SER identifies two conditions of acceptance, which are summarized below along with justification for application to Ginna.

1. *"The staff position, as well as that of the reactor vendors over the last several years, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection accident to 280 cal/gm. This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO_2 have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Westinghouse has decreased their limiting fuel failure criterion from 280 cal/gm (somewhat less than the threshold of significant conversion of the fuel thermal energy to mechanical energy) to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision on the side of safety, the staff concludes that it is an acceptable fuel failure criterion."*

   Justification

   The maximum fuel pellet enthalpy at the hot spot calculated for each Ginna-specific RCCA ejection case was less than 200 cal/gm. These results satisfy the fuel failure criterion accepted by the staff.

2. *"Westinghouse proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation imposed in the AEC ECCS Interim Acceptance Criteria, this is felt to be adequate in view of the relatively short time at temperature and the highly localized effect of a reactivity transient."*

Justification

As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (Reference 19), the 2700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long-term core coolability and compliance with applicable offsite dose requirements are identified in LR section 2.8.5.4.6.

2.8.5.0.13 Operator Actions

The feedwater system pipe break event is the only event for which an operator action is credited in the analysis. LR section 2.8.5.2.4 discusses the details of the feedwater system pipe break analysis.

2.8.5.0.14 References

1. WCAP-11397-P-A, *Revised Thermal Design Procedure*, A. J. Friedland and S. Ray, April 1989.

2. WCAP-12910 Rev. 1-A, *Pressurizer Safety Valve Set Pressure Shift*, G. O. Barrett, et al., May 1993.

3. ANSI/ANS-5.1-1979, *American National Standard for Decay Heat Power In Light Water Reactors*, August 29, 1979.

4. WCAP-8745-P-A, *Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions*, S. L. Ellenberger, et al., September 1986.

5. WCAP-7908-A, *FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod*, H. G. Hargrove, December 1989.

6. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, D. S. Huegel, et al., April 1999.

7. NSAC-78, *Retran-02 Analysis of LOFT ATWS Experiments*, EPRI, October 1984.

8. WCAP-7907-P-A, *LOFTRAN Code Description*, T. W. T. Burnett, et al., April 1984.

9. WCAP-7979-P-A, *TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code*, D. H. Risher, Jr. and R. F. Barry, January 1975.

10. WCAP-10965-P-A, *ANC: A Westinghouse Advanced Nodal Computer Code*, Y. S. Liu, et al., September 1986.

11. WCAP-14565-P-A, *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Y. X. Sung, et al., October 1999.

12. NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, *ATWS Submittal.*

13. ANS N18.2-1973, *Nuclear Safety Criteria for the Design of Stationary PWRs*, American Nuclear Society.

14. 10CFR20, *Standards for Protection Against Radiation.*

15. 10CFR100, *Reactor Site Criteria.*

16. WCAP-11394-P-A, *Methodology for the Analysis of the Dropped Rod Event*, R. L. Haessler, et al., January 1990.

17. WCAP-9272-P-A, *Westinghouse Reload Safety Evaluation Methodology*, S. L. Davidson (Ed.), July 1985.

18. WCAP-7588 Rev. 1-A, *An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods*, D. H. Risher, January 1975.

19. W. J. Johnson (Westinghouse) to R. C. Jones (NRC), NS-NRC-89-3466, *Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents*,. October 23, 1989,

Table 2.8.5.0-1
Non-LOCA Analysis Limits and Analysis Results

UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.1.1	Decrease in Feedwater Temperature	(1)	N/A	N/A
15.1.2	Increase in Feedwater Flow	Minimum DNBR (RTDP, WRB-1) (HFP) Minimum DNBR (STDP, W-3) (HZP)	1.38 (HFP) 1.613 (HZP)	1.60 (HFP) (2) (HZP)
15.1.3	Excessive Load Increase	Minimum DNBR (RTDP, WRB-1)	1.38	> 1.38
15.1.4	Inadvertent Opening of a Steam Generator Relief/Safety Valve	Bounded by Steam Line Break (UFSAR, section 15.1.5)	N/A	N/A
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	Minimum DNBR (non-RTDP, W-3)	1.566	2.58
	Steam System Piping Failure – Full Power (Core response only)	Minimum DNBR (RTDP, WRB-1 correlation) (typical/thimble)	1.38/1.38 (422V+)	1.392/1.395 (422V+)
		Peak Linear Heat Generation (kW/ft)	22.7[3]	22.67
15.1.6	Combined Steam Generator ARV and Feedwater Control Valve Failures	Minimum DNBR (RTDP, WRB-1)	1.38	1.52
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	Bounded by Loss-of-External-Electrical Load (UFSAR, section 15.2.2)	N/A	N/A
15.2.2	Loss-of-External-Electrical Load	Minimum DNBR (RTDP, WRB-1)	1.38	1.61
		Peak RCS Pressure, psia	2748.5	2746.8
		Peak MSS Pressure, psia	1208.5	1208.0

Table 2.8.5.0-1 (cont.) Non-LOCA Analysis Limits and Analysis Results				
UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.2.3	Turbine Trip	Bounded by Loss-of-External-Electrical Load (UFSAR, section 15.2.2)	N/A	N/A
15.2.4	Loss-of-Condenser Vacuum	Bounded by Loss-of-External-Electrical Load (UFSAR, section 15.2.2)	N/A	N/A
15.2.5	Loss-of-Offsite-AC-Power to the Station Auxiliaries	Maximum pressurizer mixture volume, ft^3	800	635
15.2.6	Loss-of-Normal Feedwater	Maximum Pressurizer Mixture Volume, ft^3	800	537
15.2.7	Feedwater System Pipe Breaks	Margin to Hot Leg Saturation, °F	0.0	2
15.3.1	Flow Coastdown Accident – PLOF[4]	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.601/1.597 (422V+)
	Flow Coastdown Accident – CLOF[5]	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.489/1.491 (422V+)
	Flow Coastdown Accident – UF[6]	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.385/1.392 (422V+)
15.3.2	Locked Rotor Accident	Peak RCS Pressure, psia	2997	2782
		Peak Cladding Temperature, °F	2700	1924.6 (422V+)
		Maximum Zirc-Water Reaction, %	16	0.53 (422V+)

Table 2.8.5.0-1 (cont.)
Non-LOCA Analysis Limits and Analysis Results

UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Minimum DNBR Below First Mixing Vane Grid (non-RTDP, W-3 correlation) (typical/thimble)	1.447/1.447 (422V+)	1.987/2.238 (422V+)
		Minimum DNBR Above First Mixing Vane Grid (non-RTDP, WRB-1 correlation) (typical/thimble)	1.302/1.302 (422V+)	1.957/1.951 (422V+)
		Maximum Fuel Centerline Temperature, °F	4800(7)	2108 (422V+)
15.4.2	Uncontrolled RCCA Withdrawal at Power	Minimum DNBR (RTDP, WRB-1)	1.38	1.384
		Peak RCS Pressure, psia	2748.5	2748.1
		Peak MSS Pressure, psia	1208.5	1207.7
15.4.3	Startup of an Inactive Reactor Coolant Loop, (RCL)	No Analysis Performed (See Section Licensing Report 2.8.5.4.4)	N/A	N/A
15.4.4	Chemical and Volume Control System (CVCS) Malfunction (Boron Dilution)	Minimum Time to Loss of Shutdown Margin, Minutes	15	30.3 (Mode 1 manual)
			15	33.3 (Mode 1 auto)
			15	25.1 (Mode 2)
			30	32.0 (Mode 6)

Table 2.8.5.0-1 (cont.) Non-LOCA Analysis Limits and Analysis Results				
UFSAR Section	**Event Description**	**Result Parameter**	**Analysis Result**	
			Analysis Limit	**Limiting Case**
15.4.5	Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)	Maximum Fuel Pellet Average Enthalpy, cal/g	200	151.8 (BOC-HZP) 177.9 (BOC-HFP) 155.1 (EOC-HZP) 177.2 (EOC-HFP)
		Maximum Fuel Melt, %	10	0.00 (BOC-HZP)(8) 6.62 (BOC-HFP)(8) 0.00 (EOC-HZP)(9) 9.00 (EOC-HFP)(9)
		Peak RCS Pressure, psia	Generically addressed in Reference 15	
15.4.6	RCCA Drop	Minimum DNBR (RTDP, WRB-1)	1.38	> 1.38
		Peak Linear Heat Generation (kW/ft)	22.7(3)	< 22.7
		Peak Uniform Cladding Strain (%)	1.0	< 1.0
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Minimum DNBR (WRB-1)	1.38	1.49
15.8	ATWS	Peak RCS Pressure, psig	3200	3,193

Table 2.8.5.0-1 (cont.) Non-LOCA Analysis Limits and Analysis Results				
UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case

Notes:

1. Event bounded by the steam system piping failure at full power event. See LR section 2.8.5.1.1.
2. Bounded by zero power steam line break.
3. Corresponds to a UO_2 fuel melting temperature of 4700°F.
4. PLOF ≡ partial loss of flow (one-loop flow coastdown).
5. CLOF ≡ complete loss of flow (two-loop flow coastdown).
6. UF ≡ underfrequency (frequency decay of RCP power supply)
7. UO_2 fuel melting temperature corresponding to a burnup of ~48,276 MWd/MTU.
8. Fuel melting temperature = 4900°F
9. Fuel melting temperature = 4800°F

Table 2.8.5.0-2 Non-LOCA Plant Initial Condition Assumptions			
Parameter	**RTDP**	**Non-RTDP**	**Notes**
NSSS Power (MWt)	1817.0	1817.0	1
Nominal Total Net RCP Heat (MWt)	6.0	6.0	1, 2, 3
Maximum Full-Power Vessel T_{avg} (°F)	576.0	576.0 ± 4.0	1, 4
Minimum Full-Power Vessel T_{avg} (°F)	564.6	564.6 ± 4.0	1, 4
No-Load RCS Temperature (°F)	547.0	547.0	1, 4
Pressurizer Pressure (psia)	2250	2250 ± 60	1
Steam Flow (lbm/hr)	see Note 5	see Note 5	5
Steam Pressure (psia)	see Note 5	see Note 5	5
Feedwater Temperature (°F)	390 to 435	390 to 435	1
Pressurizer Water Level (% span)	see Note 6	see Note 6	6
Steam Generator Water Level (% NRS)	see Note 7	see Note 7	7

Notes:

1. See Table 1-1 in LR section 1.1.
2. Total RCP heat input minus RCS thermal losses.
3. A maximum net RCP heat of 10 MWt was conservatively assumed in some non-RTDP analyses, e.g., loss-of-normal feedwater.
4. All analyses assumed a programmed no-load T_{avg} of 547°F. For the events initiated from a no-load condition (rod withdrawal from subcritical, steam line break, rod ejection, boron dilution), the use of the no-load temperature as the initial temperature bounded the case of startup operations at Ginna with a temperature less than 547°F.
5. The nominal steam flow rate and steam pressure depended on other nominal conditions. See Table 1-1 in LR section 1.1.
6. The nominal/programmed pressurizer water level varied linearly from 20% of span at the no-load T_{avg} of 547°F to either 44.3% of span at the minimum full-power T_{avg} of 564.6°F or 60% of span at the maximum full power T_{avg} of 576°F. The programmed level remained constant at the full-power T_{avg} level for T_{avg} values greater than the full-power T_{avg}. An uncertainty of ±5% of span was applied when conservative.
7. The programmed steam generator water level modeled in the analyses was a constant 52% narrow range span (NRS) for all power levels. An uncertainty of -4% NRS/+8% NRS was applied when conservative.

Table 2.8.5.0-3 Pressurizer and Main Steam System (MSS) Pressure Relief Assumptions			
UFSAR	Event Description	Pressure Relief Model(1)	
		Pressurizer	MSS
15.1.1	Decrease in Feedwater Temperature	5	5
15.1.2	Increase in Feedwater Flow	1	3A
15.1.3	Excessive Load Increase	5	5
15.1.4	Inadvertent Opening of a Steam Generator Relief/Safety Valve	(2)	
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	4	4
	Steam System Piping Failure – Full Power (Core response only)	4	4
15.1.6	Combined Steam Generator ARV and Feedwater Control Valve Failures	1	3A
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	(3)	
15.2.2	Loss-of-External-Electrical Load – DNB Case	1	3B
	Loss-of-External-Electrical Load – Peak RCS Pressure Case	2B	3B
	Loss-of-External-Electrical Load – Peak MSS Pressure Case	1	3B
15.2.3	Turbine Trip	(3)	
15.2.4	Loss-of-Condenser Vacuum	(3)	
15.2.5	Loss-of-Offsite-ac-Power to the Station Auxiliaries	1	3A
15.2.6	Loss-of-Normal Feedwater (LONF)	1	3A
15.2.7	Feedwater System Pipe Breaks	1	3A
15.3.1	Flow Coastdown Accidents	2A	7
15.3.2	Locked Rotor Accident	2B	3A
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	5	5
15.4.2	Uncontrolled RCCA Withdrawal at Power – DNB Case	1	3B
	Uncontrolled RCCA Withdrawal at Power – Peak RCS Pressure Case	2B	3B
15.4.3	Startup of an Inactive RCL	Analysis not required	
15.4.4	CVCS Malfunction (Boron Dilution)	5	5
15.4.5	RCCA Ejection	5	5

Table 2.8.5.0-3 (cont.) Pressurizer and Main Steam System (MSS) Pressure Relief Assumptions			
		Pressure Relief Model[1]	
UFSAR	Event Description	Pressurizer	MSS
15.4.6	RCCA Drop	6	6
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	4	4
15.8	ATWS	(4)	

Notes:
1. The pressure relief models are described below.
2. Transient bounded by steam system piping failure (UFSAR, Section 15.1.5).
3. Transient bounded by loss-of-external-electrical load (UFSAR, Section 15.2.2).
4. Generic. (See Reference 12)

Model 1 (Maximum Pressurizer Pressure Relief)

The setpoint for each of the two pressurizer power-operated relief valves (PORVs) was either 100 psi above the initial pressure or 2350 psia, whichever was lower. Each PORV had a relief rate of 179,000 lbm/hr. The pressurizer spray system was actuated when the indicated pressurizer pressure exceeded the initial value by 25 psi. The pressurizer spray valves were full-open when the indicated pressurizer pressure exceeded the initial value by 75 psi. A linear increase in the pressurizer spray valve flow area was assumed between these points. The full-open spray valve flow area was 0.0376 ft^2.

The PSV setpoint was 3% below the nominal setpoint of 2485 psig. Once the PSVs came open, they did not reseat until the pressure dropped 5% below the opening setpoint. No time delay penalty was applied to account for purging the water in the PSV loop seals. The PSV design relief rate was 288,000 lbm/hr per valve (2 valves total). Note that for the LONF, loss-of-offsite-ac power to the station auxiliaries, and feedwater system pipe break transients, the PSV model was irrelevant because the PORVs and sprays were sufficient to control pressure.

Model 2A (Minimum Pressurizer Pressure Relief)

The pressurizer PORVs and pressurizer sprays were assumed to be unavailable. Although the PSVs were modeled, they do not actuate during the transient.

Model 2B (Minimum Pressurizer Pressure Relief)

The pressurizer PORVs and pressurizer sprays were assumed to be unavailable. The PSVs setpoint was increased at least 2.3% above the nominal set pressure of 2485 psig to account for set pressure tolerance, plus an additional 1% to address the set pressure shift phenomenon associated with PSVs that had water-filled loop seals (see WCAP-12910 [Reference 2]). A maximum time delay of 0.8 seconds was applied to account for purging the water in the PSV loop seals. The PSV design relief rate was 288,000 lbm/hr per valve (2 valves total).

Table 2.8.5.0-3 (cont.)
Pressurizer and Main Steam System (MSS) Pressure Relief Assumptions

Model 3A (Staggered MSSV Setpoints)

There were 4 MSSVs on each loop with a total relief capacity of ~1861 lbm/sec (total of 8 valves). The assumed setpoints are listed below.

Valve Bank	Nominal Setpoint	Initial Open Pressure of the MSSVs*
1	1085 psig	1134.20 psia
2	1140 psig	1190.00 psia
3	1140 psig	1190.00 psia
4	1140 psig	1190.00 psia

* Pressure includes +1.5% for the setpoint tolerance, +18.2 psi for the pressure drop from the inlet connection of the 30-inch main steamline pipe to the MSSV, and +14.7 psi to convert to atmospheric pressure. The full-open pressure for each MSSV was 5 psi above the initial open pressure.

Model 3B (Staggered MSSV Setpoints)

Same as Model 3A, except that a less conservative setpoint tolerance of +1.4% (instead of +1.5%) and/or a slightly less conservative pressure drop from the inlet of the 30-inch main steamline pipe to the MSSVs of 18.07 psid (instead of 18.2 psid) were/was assumed.

Model 4

No specific pressurizer pressure or MSS relief inputs were modeled. The pressurizer pressure and steam pressure both decrease during this event. Thus, the pressurizer spray, relief valves, and safety valves, and the MSSVs were irrelevant.

Model 5

Pressurizer and MSS relief was not modeled because either the computer code(s) used for this analysis did not include pressurizer or steam generator models, or the analysis was a hand calculation that did not involve these plant components. Refer to the accident-specific analyses for additional information.

Model 6

The generic (that is, not plant-specific) analysis performed to address this event assumed that the pressurizer PORVs actuated at 2350 psia with a total maximum relief capacity of 16.65 ft^3/sec. The pressurizer spray valve setpoints assumed were the same as those specified for Model 1, but the total spray capacity was 52.2 lbm/sec. The PSVs and MSSVs were modeled and assumed to be available, but did not actuate.

Model 7

No specific MSS relief inputs were modeled because the secondary side pressure transient during the event was non-limiting.

Table 2.8.5.0-4 Overtemperature and Overpower ΔT Setpoints	
Allowable Full-Power T_{avg} Range	564.6° to 576.0°F
K_1 (safety analysis value)	1.30
K_2	0.00093/psi
K_3	0.0185/°F
K_4 (safety analysis value)	1.15
K_5	0.0014/°F (1)
K_6	0.00/°F
T′	564.6° to 576.0°F (2)
P′	2250 psia
f(ΔI) Deadband (3)	-14% ΔI (4) to +6% ΔI
f(ΔI) Negative Gain (3)	-3.08%/%ΔI (4)
f(ΔI) Positive Gain (3)	+2.27%/%ΔI
High-Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	2425 psia
Low-Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	1775 psia

Notes:

1. K_5 = 0.0014/°F is valid for increasing T_{avg}. For decreasing T_{avg}, K_5 = 0.0/°F.
2. Value to be set equal to or less than the full power operating T_{avg} chosen.
3. The f(ΔI) penalty is implicitly assumed in the non-LOCA safety analyses.
4. Value supported by non-LOCA transient analysis. Value will change based on fuel rod design analysis.

Table 2.8.5.0-5 Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.1.1	Decrease in Feedwater Temperature	N/A	N/A	N/A
15.1.2	Increase in Feedwater Flow	High-High Steam Generator Water Level Feedwater Regulator Valve Closure	100% NRS	22.0
15.1.3	Excessive Load Increase	N/A	N/A	N/A
15.1.4	Inadvertent Opening of a Steam Generator Relief/Safety Valve	(1)		
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	High-High Steam Flow Setpoint	~155% of nominal	2.0
		High Steam Flow Setpoint	1.5E6 lbm/hr	2.0
		Low Steam Pressure Safety Injection (SI) Setpoint	327.7 psia (lead/lag = 12/2)	2.0
		Steam Line Isolation Delay from SI Coincident with High-High Steam Flow	N/A	7.0
		Feedwater Isolation Delay from SI	N/A	32.0
		SI Pumps at Full Flow Following SI Signal (with/without offsite power)	N/A	12.0/22.75
	Steam System Piping Failure – Full Power (Core response only)	OPΔT reactor trip	Table 2.8.5.0-4	10.0(2)
15.1.6	Combined Steam Generator ARV and Feedwater Control Valve Failures	High-High Steam Generator Water Level Feedwater Regulator Valve Closure	100% NRS	22.0
		OPΔT Reactor Trip	Table 2.8.5.0-4	10.0(2)
		Low-Pressurizer Pressure Safety Injection	1715.0 psia	32.0

Table 2.8.5.0-5 (cont.) Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	**Event Description**	**RPS or ESFAS Signal(s) Actuated**	**Analysis Setpoint**	**Delay (sec)**
15.2.1	Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	(3)		
15.2.2	Loss-of-External-Electrical Load	High-Pressurizer Pressure Reactor Trip	2425 psia	2.0
		OTΔT Reactor Trip	Table 2.8.5.0-4	7.0(2)
15.2.3	Turbine Trip	(3)		
15.2.4	Loss-of-Condenser Vacuum	(3)		
15.2.5	Loss-of-Offsite-AC Power to the Station Auxiliaries	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level Auxiliary Feedwater (AFW) Pump Start	0% NRS	60.0
15.2.6	LONF	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level AFW Pump Start	0% NRS	60.0
15.2.7	Feedwater System Pipe Breaks	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level AFW Pump Start	0% NRS	60 & 870
15.3.1	Flow Coastdown Accidents	Low RCL Flow Reactor Trip	87%	1.0
		RCP Undervoltage Reactor Trip	N/A	1.5
		RCP Underfrequency Reactor Trip	57 Hz	1.4
15.3.2	Locked Rotor Accident	Low RCL Flow Reactor Trip	87%	1.0

Table 2.8.5.0-5 (cont.) Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	**Event Description**	**RPS or ESFAS Signal(s) Actuated**	**Analysis Setpoint**	**Delay (sec)**
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Power-Range High Neutron Flux Reactor Trip (Low Setting)	35%	0.5
15.4.2	Uncontrolled RCCA Withdrawal at Power	Power-Range High Neutron Flux Reactor Trip (High Setting)	115%	0.5
		OTΔT Reactor Trip	Table 2.8.5.0-4	7.0(2)
		High Pressurizer Pressure Reactor Trip	2425 psia	2.0
15.4.3	Startup of an Inactive RCL	N/A	N/A	N/A
15.4.4	Chemical and Volume Control System Malfunction (Boron Dilution)	OTΔT Reactor Trip	Table 2.8.5.0-4	7.0(2)
15.4.5	RCCA Ejection	Power-Range High Neutron Flux Reactor Trip (Low and High Settings)	35% (low setting)	0.5
			118% (high setting)	0.5
15.4.6	RCCA Drop	Low-Pressurizer Pressure Reactor Trip	Note 4	2.0
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	OTΔT Reactor Trip	Table 2.8.5.0-4	10.0(2)
15.8	ATWS	ATWS Mitigation System Actuation Circuitry (AMSAC) – Turbine Trip (TT), AFW Pump Start (AFW)	N/A	30 (TT) 60 (AFW)

Table 2.8.5.0-5 (cont.) Summary of RPS and ESFAS Functions Actuated
Notes: 1. Transient bounded by steam system piping failure (UFSAR, Section 15.1.5). 2. Modeling the OTΔT and OPΔT reactor trips included a time constant (first order lag) of 2.0 seconds for the RTDs and a filter (lag) of 3.5 (or 6.0) seconds on the hot-leg temperature measurement. The RTD lag accounted for the response of the RTDs and the RTD electronic filter (if any). In addition, after the overtemperature or overpower setpoint was reached, a delay of 1.5 (or 2.0) seconds was assumed to account for electronic delays, reactor trip breakers opening, and RCCA gripper release. 3. Transient bounded by loss-of-external-electrical load (UFSAR, Section 15.2.2). 4. The generic two-loop dropped RCCA analysis, applicable to Ginna, modeled the low-pressurizer pressure reactor trip setpoint as a "convenience trip." The cases that actuated this function assumed dropped rod and control bank worth combinations that were non-limiting with respect to DNB. The fact that the plant-specific low-pressurizer pressure setpoint (1775 psia) was lower than the value assumed in the generic analysis (1860 psia) did not invalidate the applicability of the generic two-loop statepoints to Ginna. Therefore, the low-pressurizer pressure reactor trip setpoint value that was used in the generic two-loop dropped RCCA analysis (1860 psia) did not represent an analytical limit for this function for Ginna

Table 2.8.5.0-6
Core Kinetics Parameters and Reactivity Feedback Coefficients

Parameter	Beginning of Cycle (Minimum Feedback)	End of Cycle (Maximum Feedback)
MTC, pcm/°F	5.0 (< 70% RTP)[1] 0.0 (≥ 70% RTP)	N/A
Moderator Density Coefficient, Δk/(g/cc)	N/A	0.45
Doppler Temperature Coefficient, pcm/°F	-0.91	-2.90
Doppler-Only Power Coefficient, pcm/%power (Q = power in %)	-12.0 + 0.045Q	-24.0 + 0.100Q
Delayed Neutron Fraction	0.0072 (maximum)	0.0043 (minimum)
Minimum Doppler Power Defect, pcm		
– RCCA Ejection	1000	950
– RCCA Withdrawal from Subcritical	1100	N/A

Note:
1. RTP ≡ Rated Thermal Power

Table 2.8.5.0-7
Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial NSSS Power	RCS Flow (gpm)	RCS Temp (°F)	RCS Pressure (psia)
Decrease in Feedwater Temperature	Event bounded by the excessive-load-increase event						
Increase in Feedwater Flow	RETRAN VIPRE	WRB-1 (HFP) W-3 (HZP)	Yes (HFP) No (HZP)	1817 MWt 0 MWt	177,300 (HFP) 170,200 (HZP)	576.0 (HFP) 547.0 (HZP)	2250
Excessive Load Increase	N/A	WRB-1	Yes	1817 MWt	177,300	576.0	2250
Inadvertent Opening of a Steam Generator Relief/Safety Valve	Event bounded by the steam system piping failure event.						
Rupture of a Steam Pipe – Zero Power Core Response	RETRAN VIPRE	W-3	No	0 MWt	170,200	547.0	2250
Rupture of a Steam Pipe – Full Power Core Response	RETRAN VIPRE	WRB-1	Yes	1817 MWt	177,300	576.0	2250
Combined Steam Generator ARV and Feedwater Control Valve Failures	RETRAN	WRB-1	Yes	1817 MWt	177,300	576.0	2250
Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	Event bounded by the loss-of-external-electrical-load event.						

Table 2.8.5.0-7 (cont.) Summary of Initial Conditions and Computer Codes Used							
Accident	Computer Codes Used	DNB Correlation	RTDP	Initial NSSS Power	RCS Flow (gpm)	RCS Temp (°F)	RCS Pressure (psia)
Loss-of-External-Electrical Load	RETRAN	WRB-1	N/A (pressure) Yes (DNB)	1817 MWt (pressure) 1817 MWt (DNB)	170,200 (pressure) 177,300 (DNB)	580.0 (pressure) 576.0 (DNB)	2190 (pressure) 2250 (DNB)
Turbine Trip	Event bounded by the loss-of-external-electrical-load event.						
Loss-of-Condenser Vacuum	Event bounded by the loss-of-external-electrical load event.						
Loss-of-Offsite-ac-Power to the Station Auxiliaries	RETRAN	N/A	N/A	1817 MWt	170,200	572.0	2310
Feedwater System Pipe Breaks	RETRAN	N/A	N/A	1817 MWt	170,200	580.0	2190
Flow Coastdown Accident	RETRAN VIPRE	WRB-1	Yes	1817 MWt	177,300	576.0	2250
Locked Rotor Accident	RETRAN VIPRE	N/A	N/A	1817 MWt	170,200	580.0	2310
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	W-3(1) WRB-1(2)	No	0 MWt (core power)	76,420(3)	547	2190
Uncontrolled RCCA Withdrawal at Power	RETRAN	WRB-1	Yes (DNB) N/A (Pressure)	1817 MWt (100%) (DNB/MSS Press.) 1090.2 MWt (60%) (DNB/MSS press.) 181.7 MWt (10%) (DNB/MSS Press.) 145.4 MWt (8%) (RCS Press.)	177,300 (DNB/MSS) 170,200 (RCS Press.)	576.0 (100%) 564.4 (60%) 549.9 (10%) 553.9 (8%)	2250 (DNB/MSS) 2190 (RCS Press.)
Startup of an Inactive RCL	See LR section 2.8.5.4.4						

Table 2.8.5.0-7 (cont.) Summary of Initial Conditions and Computer Codes Used							
Accident	Computer Codes Used	DNB Correlation	RTDP	Initial NSSS Power	RCS Flow (gpm)	RCS Temp (°F)	RCS Pressure (psia)
CVCS System Malfunction	N/A	N/A	N/A	1817 MWt (100%) (Mode 1) 90.9 MWt (5%) (Mode 2) 0 MWt (0%) (Mode 6)	N/A	580 (Mode 1) 547 (Mode 2) 140 (Mode 6)	2,250 (Modes 1 and 2) 14.7 (Mode 6)
RCCA Ejection	TWINKLE FACTRAN	N/A	N/A	1811 MWt (core power) (HFP) 0 MWt (core power) (HZP)	170,200 (HFP) 76,420(3) (HZP)	580.0 (HFP) 547.0 (HZP)	2190
RCCA Drop	LOFTRAN(4) ANC VIPRE	WRB-1	Yes	1817 MWt	177,300	576.0	2250
Inadvertent Opening of a Pressurizer Safety or Relief Valve	RETRAN	WRB-1	Yes	1817 MWt	177,300	576.0	2250
ATWS	LOFTRAN	N/A	N/A	1817 MWt	170,200	574.5	2250

Notes:

1. Below the first mixing vane grid.
2. Above the first mixing vane grid.
3. Single-loop flow = 0.449 * TDF.
4. The LOFTRAN portion of the analysis was generic; the DNB evaluation performed with VIPRE utilized the plant-specific values presented.

Table 2.8.5.0-8 Non-LOCA Transients Evaluated or Analyzed			
Transient	**Report Section**	**UFSAR Section**	**Notes**
Decrease in Feedwater Temperature	2.8.5.1.1	15.1.1	2
Increase in Feedwater Flow	2.8.5.1.1	15.1.2	1
Excessive Load Increase	2.8.5.1.1	15.1.3	2
Inadvertent Opening of a Steam Generator Relief/Safety Valve	2.8.5.1.1	15.1.4	2
Rupture of a Steam Pipe – Zero Power Core Response	2.8.5.1.2	15.1.5	1
Rupture of a Steam Pipe – Full Power Core Response	2.8.5.1.2	15.1.5	1
Combined Steam Generator ARV and Feedwater Control Valve Failures	2.8.5.1.1	15.1.6	1
Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	2.8.5.2.1	15.2.1	2
Loss-of-External-Electrical Load	2.8.5.2.1	15.2.2	1
Turbine Trip	2.8.5.2.1	15.2.3	2
Loss of Condenser Vacuum	2.8.5.2.1	15.2.4	2
Loss-of-Offsite-ac-Power to the Station Auxiliaries	2.8.5.2.2	15.2.5	1
LONF	2.8.5.2.3	15.2.6	1
Feedwater System Pipe Breaks	2.8.5.2.4	15.2.7	1
Flow Coastdown Accident	2.8.5.3.1	15.3.1	1
Locked Rotor Accident	2.8.5.3.2	15.3.2	1
Uncontrolled RCCA Withdrawal from a Subcritical Condition	2.8.5.4.1	15.4.1	1
Uncontrolled RCCA Withdrawal at Power	2.8.5.4.2	15.4.2	1
Startup of an Inactive RCL	2.8.5.4.4	15.4.3	2
CVCS Malfunction	2.8.5.4.5	15.4.4	1
RCCA Ejection	2.8.5.4.6	15.4.5	1
RCCA Drop	2.8.5.4.3	15.4.6	1
Inadvertent Opening of a Pressurizer Safety or Relief Valve	2.8.5.6.1	15.6.1	1
ATWS	2.8.5.7	15.8	1
Notes: 1. Complete analysis 2. Evaluation			

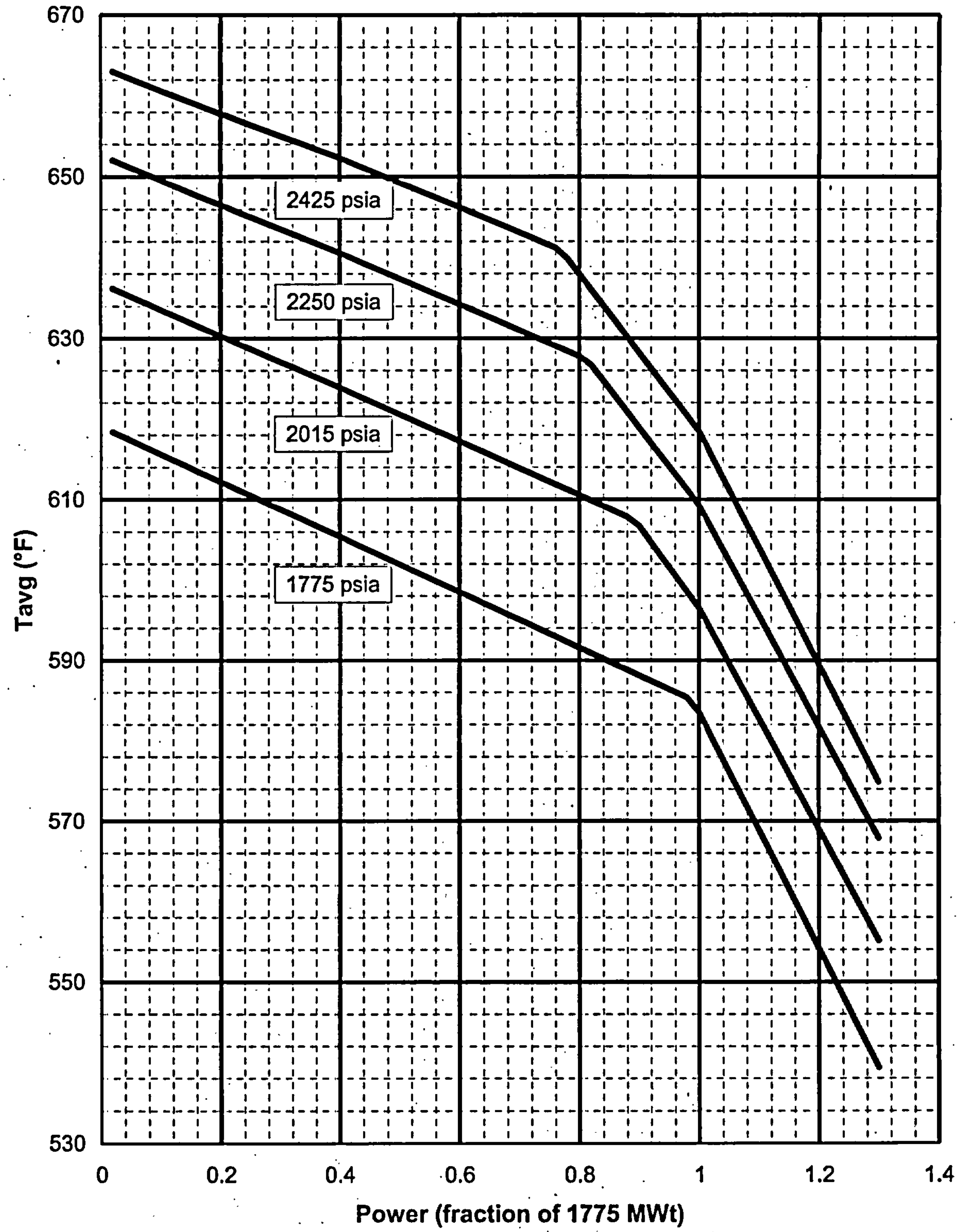


Figure 2.8.5.0-1
Reactor Core Safety Limits

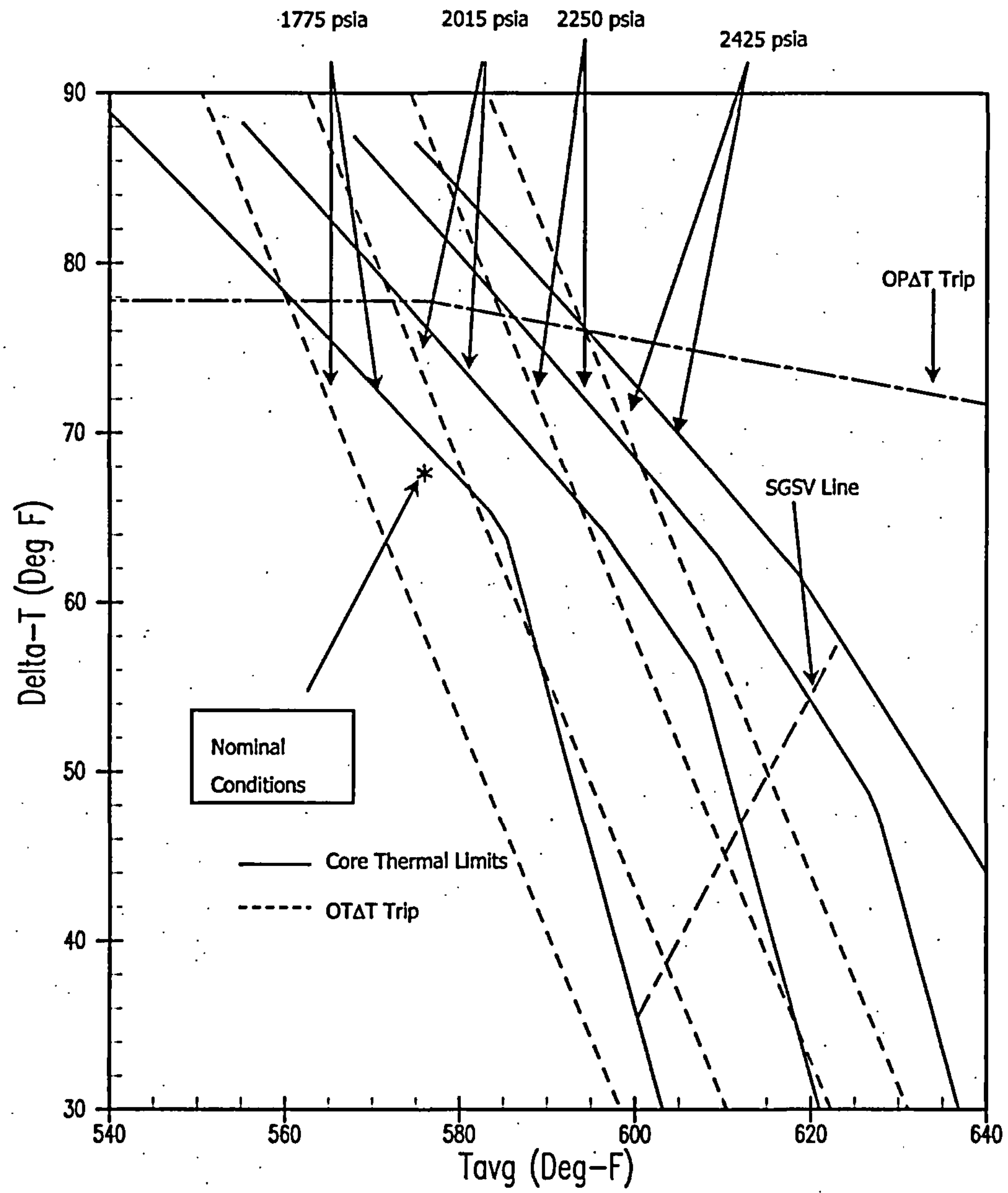


Figure 2.8.5.0-2
Illustration of OTΔT and OPΔT Protection

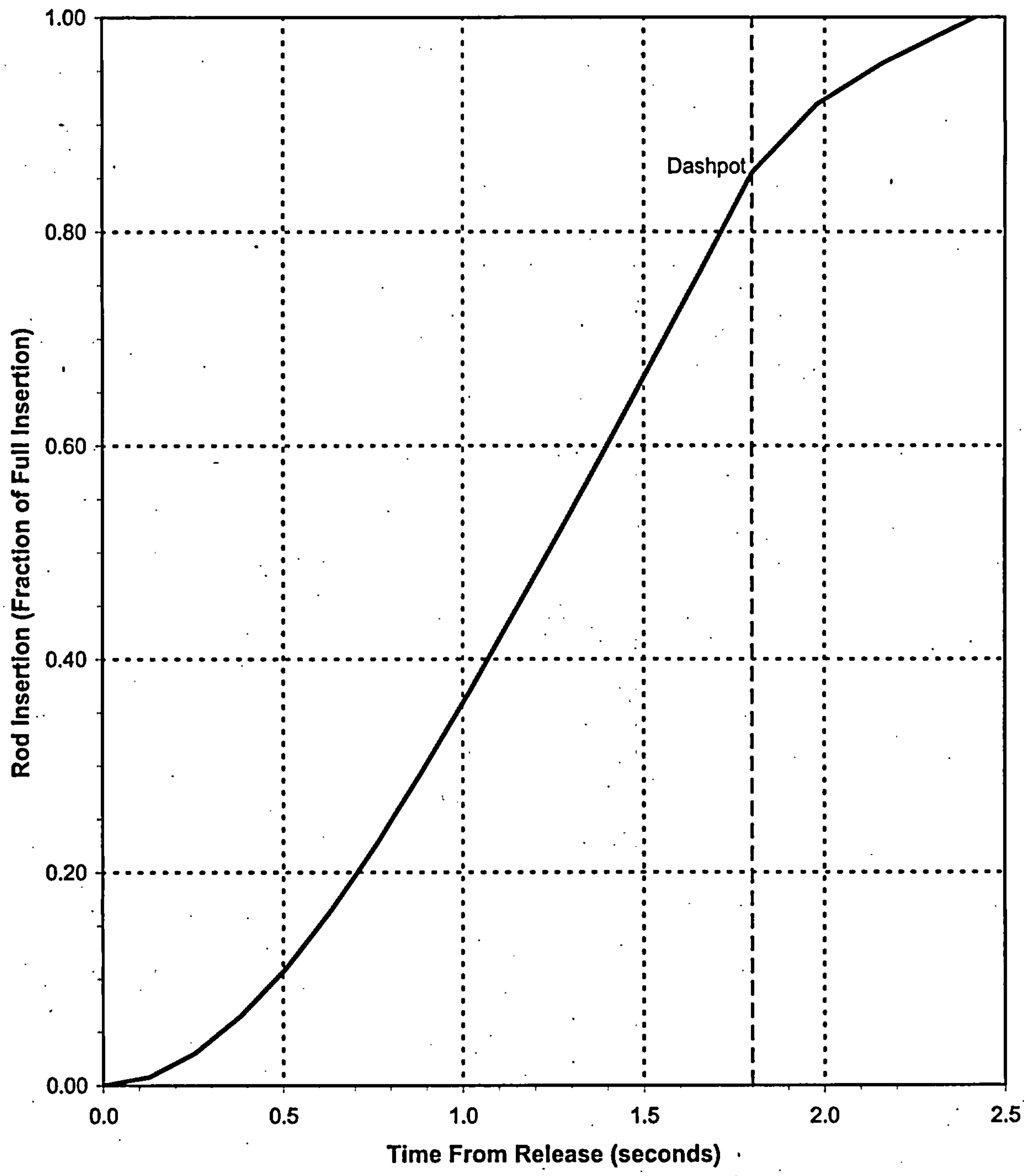


Figure 2.8.5.0-3
Fractional Rod Insertion vs. Time from Release

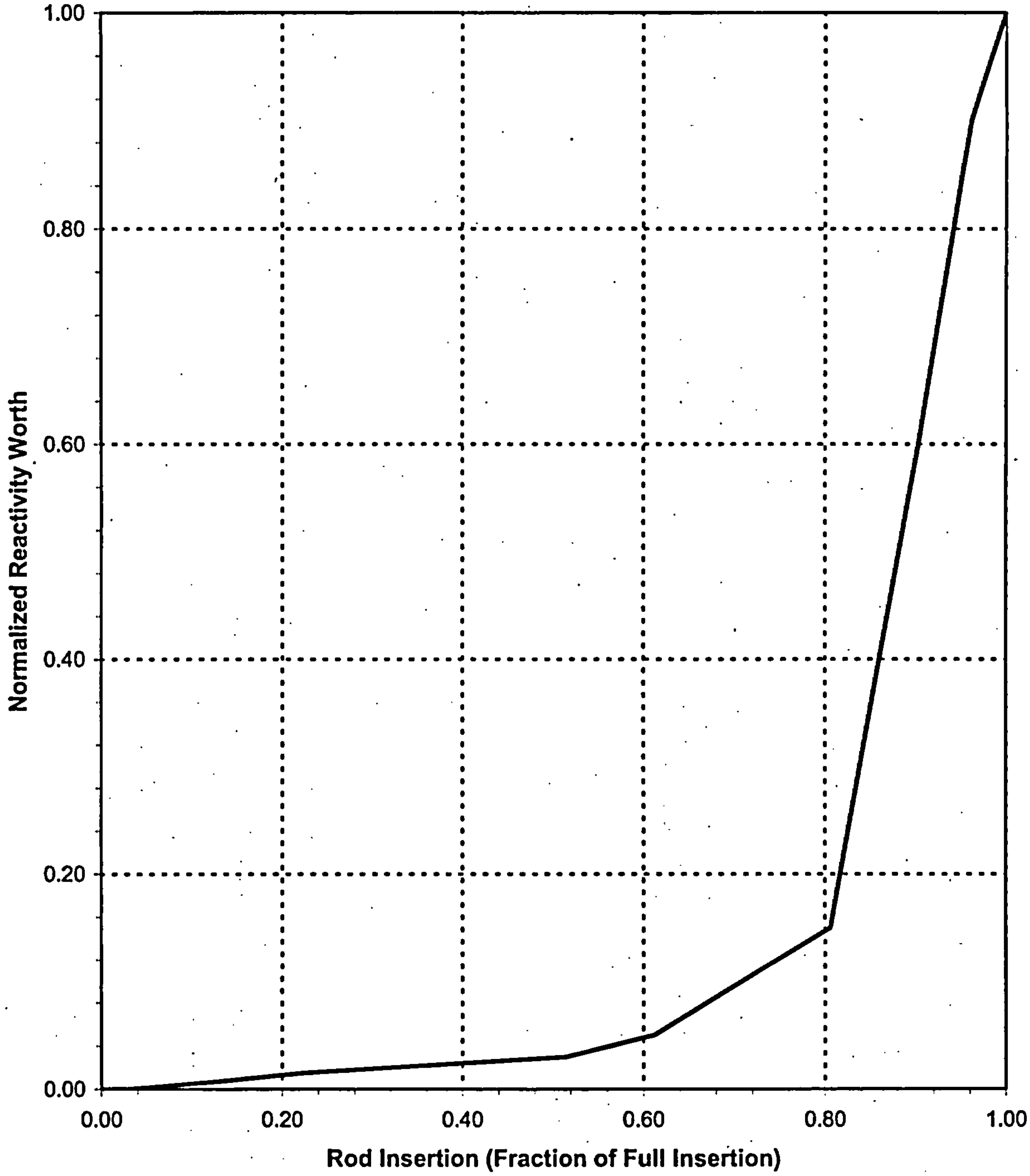


Figure 2.8.5.0-4
Normalized RCCA Reactivity Worth vs. Fractional Rod Insertion

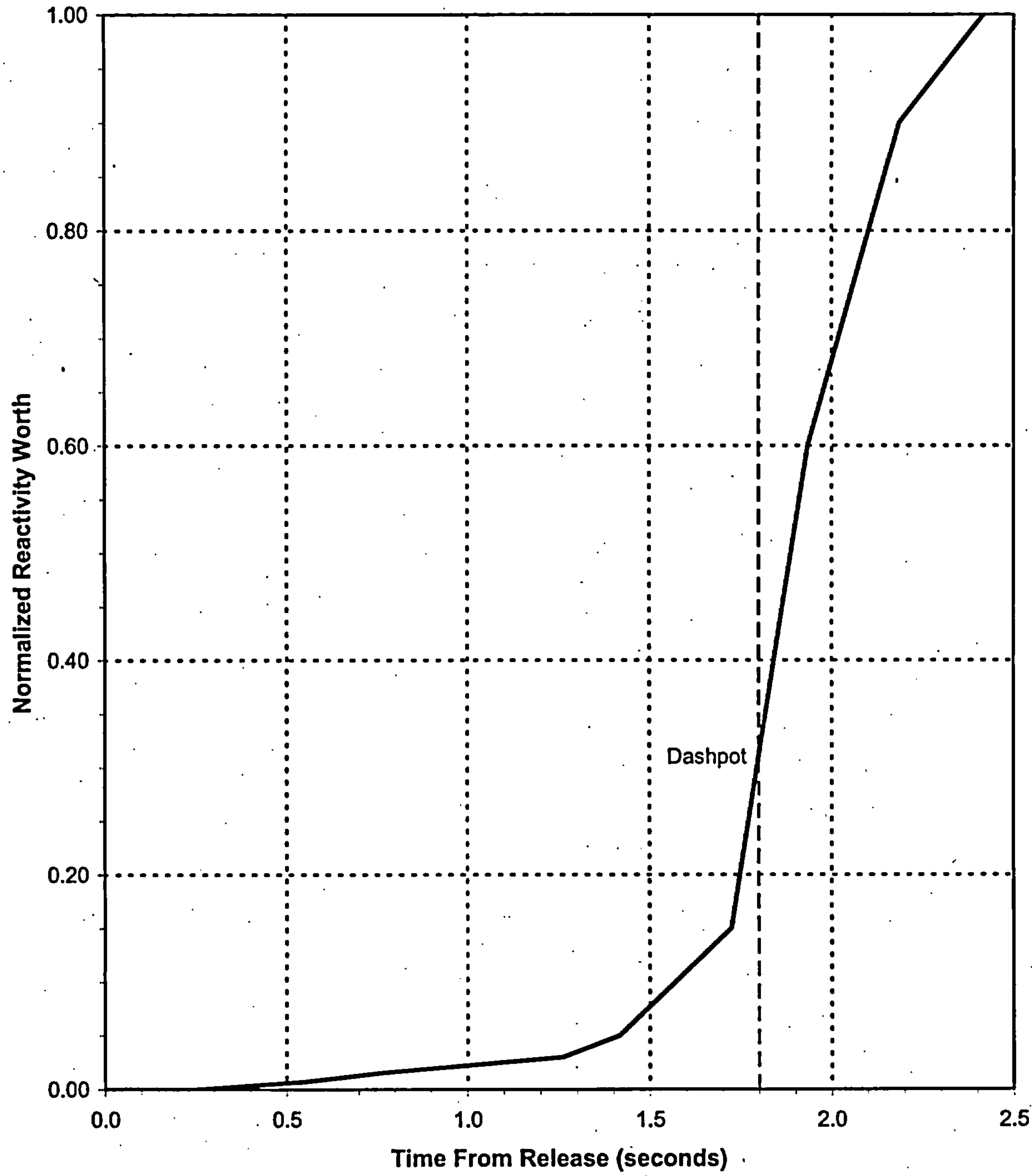


Figure 2.8.5.0-5
Normalized RCCA Reactivity Worth vs. Time from Release

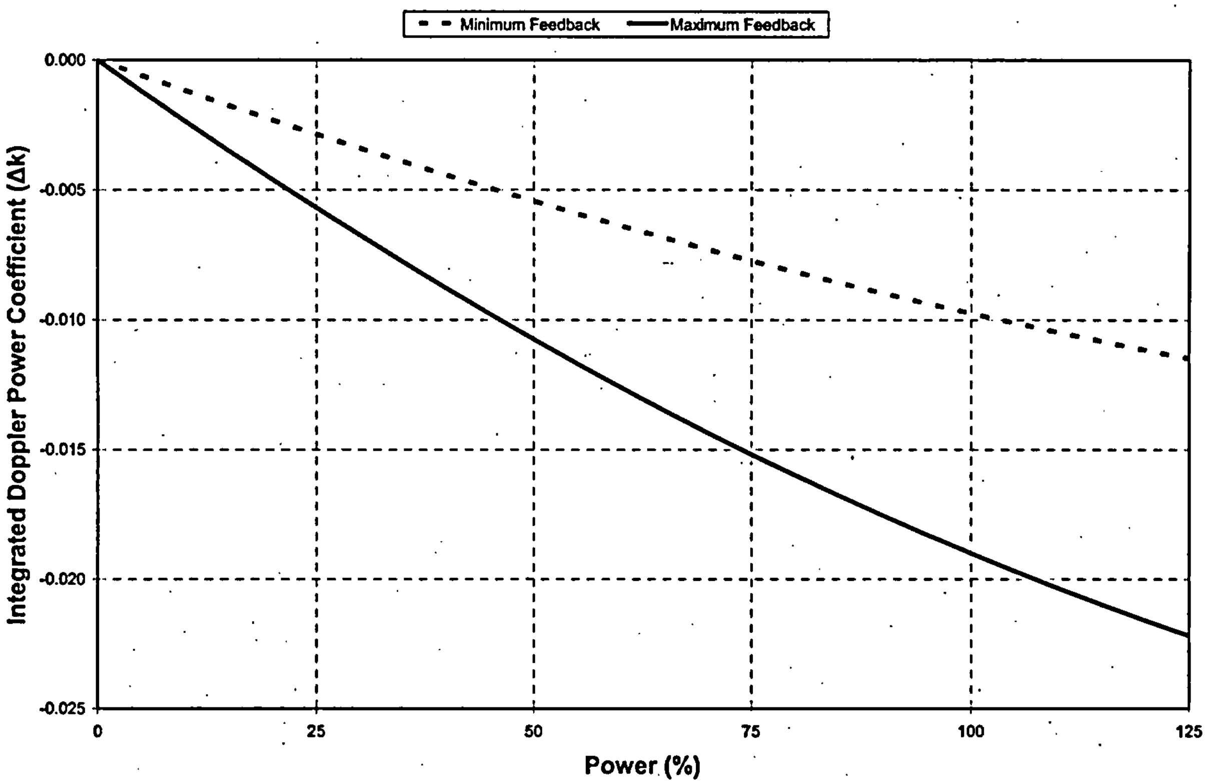


Figure 2.8.5.0-6
Integrated DPC Used in Non-LOCA Transient Analyses

2.8.5.1 Increase in Heat Removal by the Secondary System

2.8.5.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

2.8.5.1.1.1 Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature that increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase can result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The postulated initial core and reactor conditions
- The methods of thermal-hydraulic analyses
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor protection system
- The operator actions
- The results of the transient analyses

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences

- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems are designed with sufficient margin to ensure that the design condition of the reactor coolant pressure boundary is not exceeded during any condition of normal operation

- GDC-20, insofar as it requires that the reactor protection system is designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences

- GDC-26, insofar as it requires that a reactivity control system is provided, and is capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Specific review criteria are contained in the SRP, Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001, Revision 0.

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. However, for this event, the analyses performed demonstrate that the requirements specified by the GDC in 10CFR50, Appendix A are met. Specifically, the adequacy of the Ginna Station analyses for events resulting in increase in heat removal by the secondary system relative to conformance to:

- GDC-10 is described in Ginna UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control, and protection systems, provides margins to ensure the fuel is not damaged during MODES 1 and 2 or as the result of anticipated operational transients.

- 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 its associated auxiliary systems were designed with sufficient margins so that design conditions are not exceeded during MODES 1 and 2 including anticipated operational occurrences. Overpressurization is prevented by a combination of automatic control and pressure relief devices. Analyses of transients which can result in an increase in primary system heat removal by the secondary system are discussed in UFSAR section 15.1

- GDC-20 is described in Ginna UFSAR section 3.1.2.3.1, General Design Criterion 20 – Protection Systems Functions. As described in this UFSAR section, a plant protection system is provided to automatically initiate appropriate action whenever specific plant conditions reach pre-established limits to ensure that fuel design limits are not exceeded for anticipated operational occurrences. Other protective instrumentation is provided to initiate actions to mitigate the consequences of an accident. Plant protection systems are described in UFSAR section 7.2.

- GDC-26 is described in Ginna UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, two reactivity control systems of different principles are employed. Control rod drive mechanisms regulate the position of neutron absorbing control rods within the core. The control rods are designed to shut the reactor down with adequate margin for all anticipated operational occurrences so that fuel design limits are not exceeded. The chemical and volume control system provides boric acid neutron absorber to the reactor coolant which is capable of controlling the reactivity change resulting from planned normal power changes. Reactivity control system redundancy and capability are discussed in UFSAR section 4.3 and 9.3.4.

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, components associated with the control and mitigation of transients that could result in an increase in heat removal by the secondary system were evaluated within the system that contained them.

2.8.5.1.1.2 Technical Evaluation

2.8.5.1.1.2.1 Decrease in Feedwater Temperature

2.8.5.1.1.2.1.1 Introduction

The reduction in feedwater temperature is one means of increasing core power above authorized power. Such increases are attenuated by the thermal capacity in the secondary plant and in the RCS. The overpower-overtemperature protection functions (overpower ΔT and overtemperature ΔT trips) prevent any power increase that could lead to a DNBR less than the safety analysis limit.

An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the condensate bypass valve that diverts flow around the low-pressure feedwater heaters. In the event of an accidental opening of the condensate bypass valve, there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling would create a greater load demand on the primary system that can potentially lead to a reactor trip. The net anticipated effect on the RCS is similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

2.8.5.1.1.2.1.2 Description of Analyses and Evaluation

The opening of a low-pressure feedwater heater bypass valve causes a reduction in feedwater temperature that increases the thermal load on the primary system. The increased thermal load, due to the opening of the condensate bypass valve, results in a transient similar, but of a greatly reduced magnitude, to the steam system piping failure initiated from full-power conditions described in <u>LR section 2.8.5.1.2.2.2</u>. Thus, the feedwater temperature reduction transient is bounded by a steam system piping failure initiated from full power. The feedwater system malfunction is a Condition II event. Since the steam system piping failure is analyzed to Condition II acceptance criteria, no transient results are presented, as no explicit analysis is performed for the decrease in feedwater temperature case.

2.8.5.1.1.2.2 Increase in Feedwater Flow

2.8.5.1.1.2.2.1 Introduction

Excessive feedwater additions are means of increasing core power above authorized power. Such transients are attenuated by the thermal capacity of the RCS and the secondary side of the plant. The overpower/overtemperature protection functions (neutron high flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the limit value.

During normal operation the main feedwater bypass valves (MFBPVs) can be partially open at the same time that the main feedwater regulating valves (MFRVs) are open. Thus, for the increase in feedwater flow events, it is possible for a regulating valve to open fully when its bypass valve is open. Flow control failures causing the MFRV to fully open are considered an initiating event. With an increase in feedwater flow at power, the high steam generator water level setpoint is approached in the faulted loop(s). The high steam generator water level will close the MFRV and the feedwater bypass valve (if open) in the associated loop(s); the signal does not result in a turbine trip or a reactor trip. This temporarily terminates the addition of feedwater to the affected steam generator(s) and the water level begins to drop. When the water level drops below the high steam generator water level setpoint, the closure signal clears and the valves will reopen, potentially causing the steam generator water level to increase. The control can oscillate between full closed and open until a reactor trip signal or a safety injection signal is generated. If no protection setpoint is approached, the MFRVs will continue to cycle until the operator has had time to identify the problem and take the appropriate action, which could be to manually trip the reactor and isolate feedwater.

The failure of a processing controller in the advanced digital feedwater control system (ADFCS) is also postulated to cause the simultaneous failure (spurious opening) of the atmospheric relief valve(s) (ARV(s)), the MFRV(s), and the MFBPV(s). Therefore, cases combining ARV and MFRV/MFBPV failures are examined.

The spurious opening of a steam generator ARV is a credible steam line break. A credible steam line break results in a cooldown of the reactor coolant system due to the excessive heat removal caused by the increase in steam flow. Steam line break analyses assume maximum reactivity feedback, which result in addition of positive reactivity. This addition of positive reactivity can cause a return to criticality (return to power).

Thus, modeling an increase in feedwater flow (feedwater system malfunction) coincident with the spurious opening of an ARV (credible steamline break) causes a more severe cooldown than either a feedwater system malfunction or credible steamline break event by itself.

2.8.5.1.1.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The feedwater system malfunction event is analyzed to confirm that the minimum DNBR remains greater than the limit and to confirm the adequacy of the plant protection against steam generator overfill. Thus, the analysis uses the following key modeling characteristics:

- The *Revised Thermal Design Procedure* (RTDP) (Reference 1) is employed for the cases initiated from full-power. Initial reactor power, RCS pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state full power operation. Minimum measured flow (MMF) is modeled. Uncertainties in initial conditions are included in the DNBR limit as described in Reference 1.

- For the cases initiated at zero-power, initial reactor power, RCS pressure, and RCS temperature are assumed to be at levels corresponding to no-load conditions. Thermal design flow is modeled. In addition, the reactor is assumed to be at the minimum shutdown margin condition of 1.3%Δk.

- The nuclear steam supply system (NSSS) power level is set at 1817 MWt.

- For the feedwater malfunction accident at full-power conditions (with an open bypass valve) that results in an increase in feedwater flow to one steam generator, one MFRV is assumed to malfunction resulting in a step increase to 200% of nominal full-power feedwater flow to one steam generator.

- For the feedwater malfunction accident at full-power conditions (with an open bypass valve) that results in an increase in feedwater flow to both steam generators, both MFRVs are assumed to malfunction resulting in a step increase to 170% of nominal full-power feedwater flow to both steam generators.
- The increase in feedwater flow rate results in a decrease in the feedwater temperature due to the reduced efficiency of the feedwater heaters. For the full-power cases, a 25 Btu/lbm decrease in the feedwater enthalpy is conservatively assumed to occur coincident with the feedwater flow increase.
- For the feedwater malfunction accident at no-load conditions that results in an increase in feedwater flow to one steam generator, one MFRV is assumed to malfunction resulting in a step increase to 110% of nominal full-power feedwater flow to one steam generator.
- For the feedwater malfunction accident at no-load conditions that results in an increase in feedwater flow to both steam generators, both MFRVs are assumed to malfunction resulting in a step increase to 110% of nominal full-power feedwater flow to both steam generators.
- For the full-power cases, an initial water level of nominal-minus-uncertainty in both steam generators is modeled, while an initial level at nominal level minus uncertainties is evaluated for the zero-power cases.
- Pressurizer sprays and power-operated relief valves (PORVs) are modeled to reduce RCS pressure resulting in a conservative evaluation of the margin to the DNBR limit.
- The OPΔT reactor trip is modeled for all full-power cases.
- Closure of the MFRV(s) on high steam generator level at a setpoint of 100% of the narrow range level occurs on a loop-specific basis. Closure of both MFRVs occurs on low-pressurizer pressure SI at a setpoint of 1715 psia. Closure of the MFRV(s) is modeled 22 seconds after reaching a high steam generator level setpoint and 32 seconds after reaching a low-pressurizer pressure SI setpoint.

- Cases are analyzed with and without automatic rod control for the full-power cases.
- Ten minutes after transient initiation, operator actions to trip the reactor and to isolate the steamlines and feedlines are modeled for the full power cases.
- No credit is taken for the heat capacity of the RCS and steam generator metal mass in attenuating the resulting plant cooldown.

Based on its frequency of occurrence, the feedwater system malfunction event is considered a Condition II event as defined by the American Nuclear Society (ANS). As such, the applicable acceptance criteria for this incident are:

- Pressure in the RCS and MSS should be maintained below 110% of the design pressure
- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains greater than the 95/95 DNBR limit in the limiting fuel rods.
- An accident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The primary acceptance criterion used in this analysis is that the minimum DNBR remains greater than the safety analysis limit. The event does not challenge the primary and secondary side pressure limits since the increased heat removal tends to cool the RCS.

2.8.5.1.1.2.2.3 Description of Analyses and Evaluations

The excessive heat removal due to a feedwater system malfunction transient was analyzed with the RETRAN (Reference 2) computer code. This code simulates a multiloop RCS, core neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray and heaters, steam generators, and main steam safety valves (MSSVs). The code computes pertinent plant variables including temperatures, pressures, and power level.

The cases shown in Table 2.8.5.1.1.2.2-1 were considered. This differs slightly from the spectrum of cases considered in the Ginna Station pre-uprate UFSAR. The previous analysis cases modeling only ARV failures are clearly bounded by the steamline break analyses documented in LR section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment. Also, cases that modeled a single FCV failure and a single ARV failure on opposite loops are bounded by cases that model multiple FCV and ARV failures. Thus, these cases were not explicitly analyzed for the EPU.

The MFRVs re-opened on a loop-specific basis when the associated steam generator level falls below the high steam generator level setpoint. The re-opening of the MFRV was accounted for in each of the full-power cases either by conservatively maintaining the steam generator level in the faulted loop(s) at the level attained 22 seconds after reaching the high steam generator level setpoint (Cases 1, 2, 3, 4, 10, and 11) until operator-action at 10 minutes, or by conservatively maintaining the feedwater flow at the maximum faulted value until feedwater isolation (FWI) occurs on low-pressurizer pressure SI(Cases 8 and 9). For the no-load cases, the re-opening never occurred because the steam generator level never fell below 100% NRS before FWI occurred due to a SI signal on low-pressurizer pressure (Cases 5, 10, and 11).

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

In addition to the evaluations described in the Ginna UFSAR, the systems and components used to control and mitigate transients associated increases in heat removal by secondary systems were evaluated for the Ginna License Renewal within the systems that contain them. The evaluations are documented in the License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004. The aging management review for these systems and components are addressed in the License Renewal SER section 3.0. The EPU does not add any new structures/ components used to control or mitigate transients associated with an increase in heat removal by the secondary system.

2.8.5.1.1.2.2.4 Increase in Feedwater Flow Results

Case 8 modeling a failure of the MFRVs and ARVs in both loops with the reactor in manual rod control resulted in the least margin to the DNBR limit for the full-power cases. This case yields a minimum DNBR of 1.527 compared to a minimum DNBR from the previous analysis of 1.810. Primary and secondary pressure results are not noted here because these limits are not challenged for these events.

Table 2.8.5.1.1.2.2-2 shows the time sequence of events for this transient. Figures 2.8.5.1.1.2.2-1 through 2.8.5.1.1.2.2-4 show transient responses for various system parameters during this transient initiated from full-power conditions with manual rod control.

Although Case 11 was the most limiting case (manual rod control and a failure of the MFRVs and ARVs in both loops) initiated at zero-power conditions, this case was less limiting than the full-power cases considered herein. This case resulted in a minimum DNBR above the limit values of 1.613 (422V+ fuel) and 1.566 (OFA fuel).

For the excessive feedwater addition event, the results show that the DNBRs encountered are above the limit value; hence, no fuel damage is predicted.

The decrease in feedwater temperature transient due to the opening of a condensate bypass valve diverting flow around the low-pressure feedwater heaters is bounded by the steam system piping failure initiated from full-power.

The protection features presented in LR section 2.8.5.1.1.2.2.2 provide mitigation of the feedwater system malfunction transient such that the criteria of the same section are satisfied.

2.8.5.1.1.2.2.5 Increase in Feedwater Flow References

1. WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), *Revised Thermal Design Procedure*, Friedland, A. J. and Ray, S., April 1989.

2. WCAP-14882-P-A (Proprietary), April 1999 and WCAP-15234-A (Nonproprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, D. S., et al., May 1999.

Table 2.8.5.1.1.2.2-1 Cases Considered Using RETRAN				
Case	**Power Level**	**Failure**	**Affected Loop(s)**	**Rod Control**
1	HFP (Hot Full Power)	MFRV	Loop 1	Auto
2	HFP	MFRV	Loop 1	Manual
3	HFP	MFRV	Both	Auto
4	HFP	MFRV	Both	Manual
5	HZP (Hot Zero Power)	MFRV	Both	Manual
6	HFP	MFRV ARV	Loop 1 Loop 1	Manual
7	HFP	MFRV ARV	Loop 1 Loop 1	Auto
8	HFP	MFRV ARV	Both Both	Manual
9	HFP	MFRV ARV	Both Both	Auto
10	HZP	MFRV ARV	Loop 1 Loop 1	Manual
11	HZP	MFRV ARV	Both Both	Manual

Table 2.8.5.1.1.2.2-2
Time Sequence of Events – Excessive Heat Removal Due to Feedwater System Malfunctions

Event	Time (seconds)
Two MFRVs Fail Full Open Two ARVs Fail Full Open	0
OPΔT Setpoint Reached	45.5
Rod Motion Begins	47.5
Minimum DNBR Occurs	47.7
Low-Pressurizer Pressure SI Setpoint Reached	98.4
Loops 1 and 2 MFRV Closure on Low-Pressurizer Pressure SI	130.4

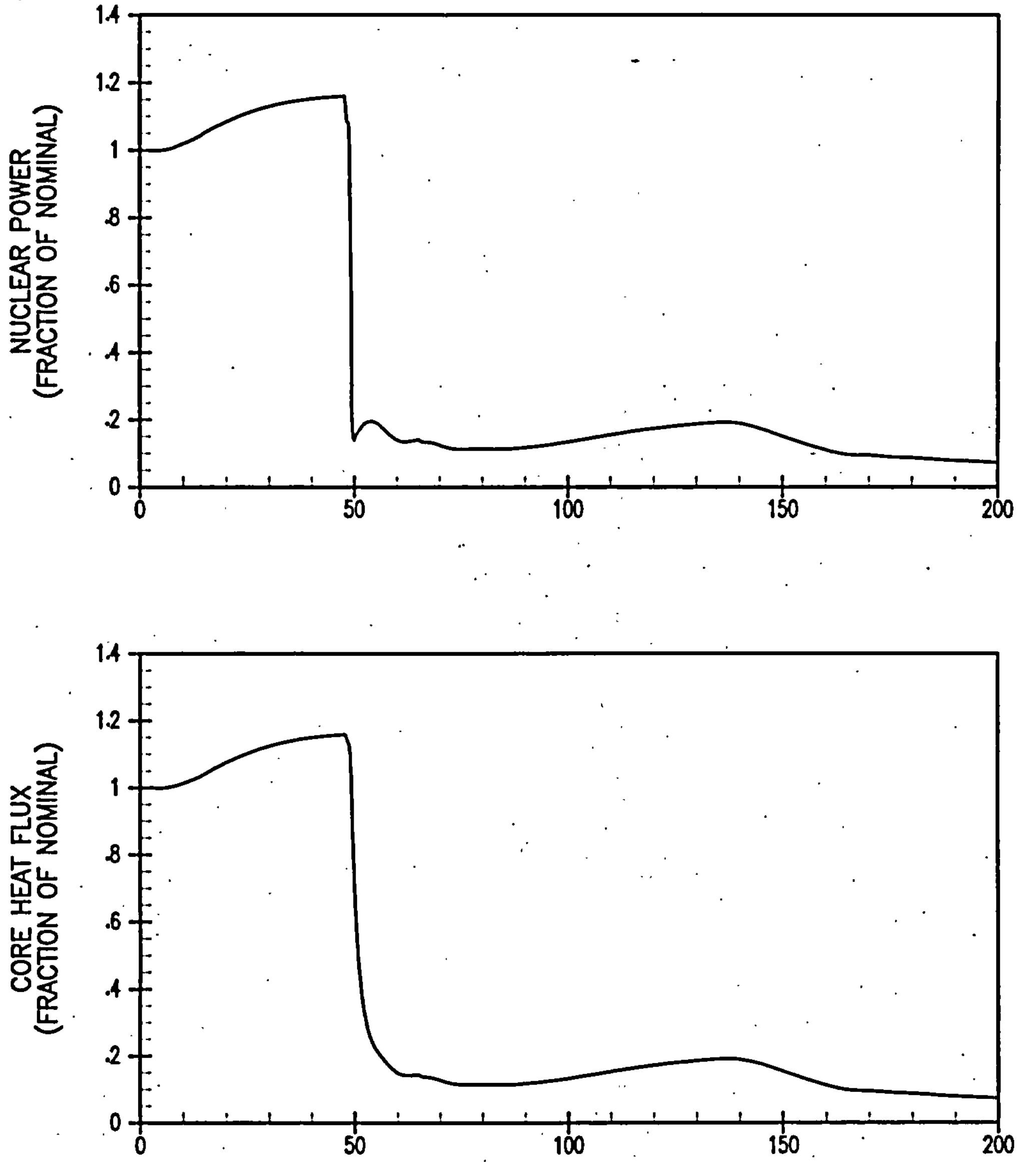


Figure 2.8.5.1.1.2.2-1
Feedwater System and ARV Malfunction at Full-Power
Nuclear Power and Core Heat Flux vs. Time

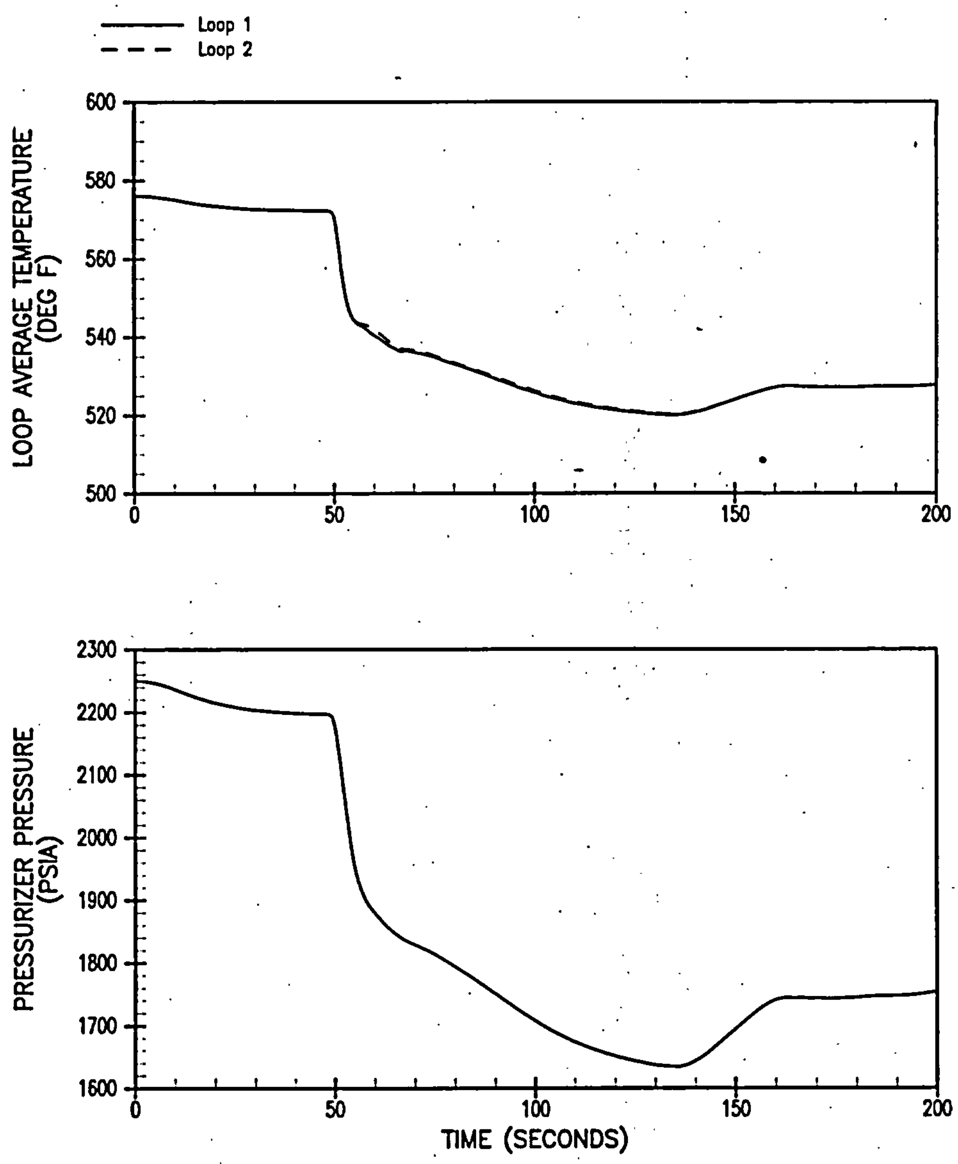


Figure 2.8.5.1.1.2.2-2
Feedwater System and ARV Malfunction at Full-Power
Loop Average Temperature and Pressurizer Pressure vs. Time

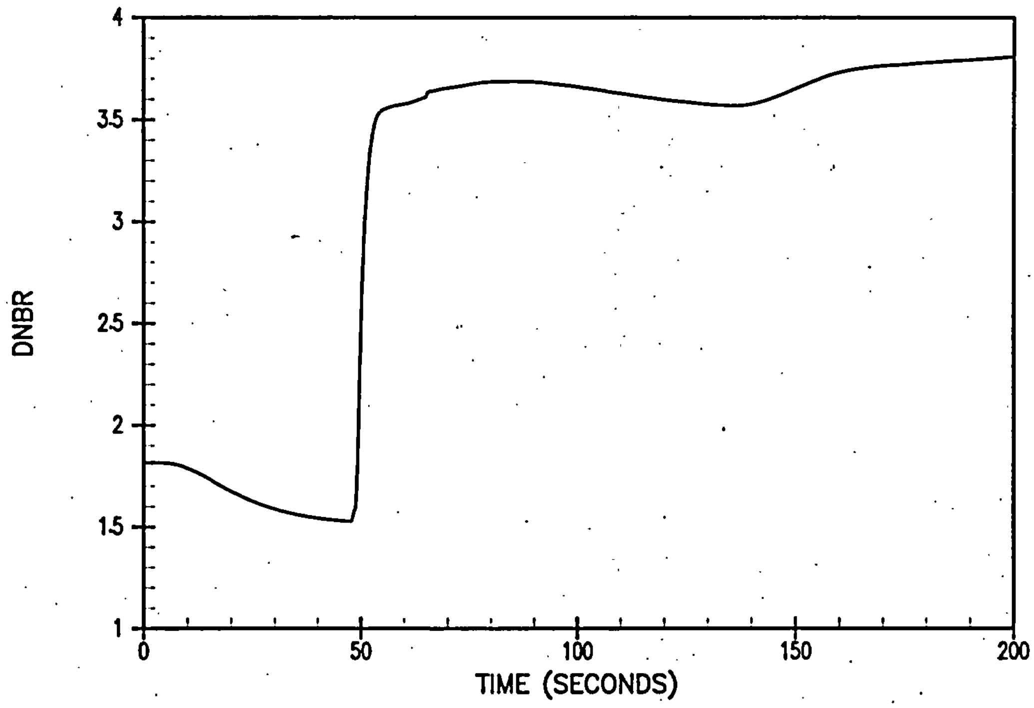


Figure 2.8.5.1.1.2.2-3
Feedwater System and ARV Malfunction at Full-Power
DNBR vs. Time

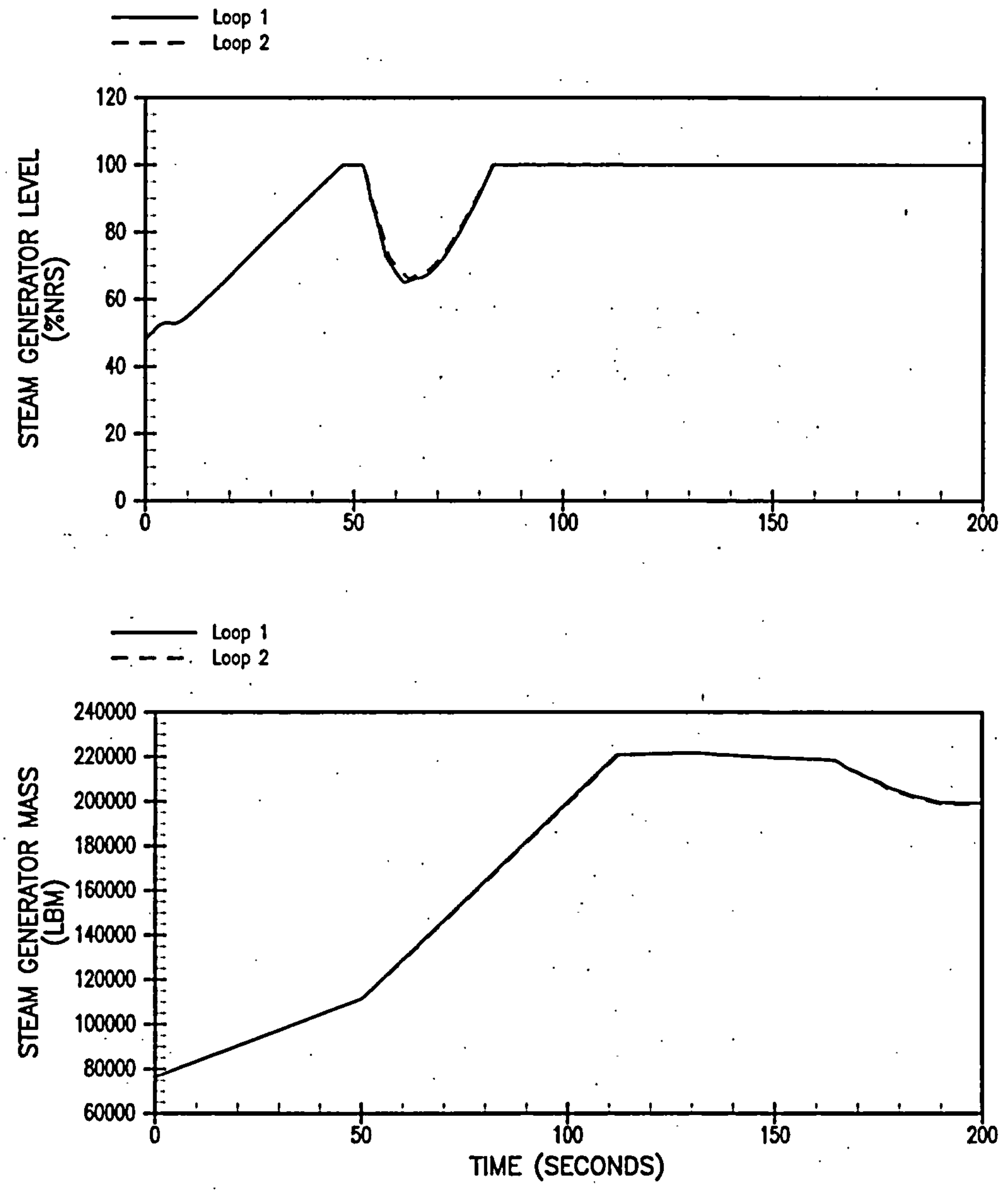


Figure 2.8.5.1.1.2.2-4
Feedwater System and ARV Malfunction at Full-Power
Steam Generator Level and Mass vs. Time

2.8.5.1.1.2.3 Increase in Steam Flow

2.8.5.1.1.2.3.1 Introduction

An excessive load increase incident is defined as a rapid increase in steam flow that causes a mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp-load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values can cause a reactor trip actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient would be terminated in sufficient time to prevent the DNB design basis from being violated.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control system, or turbine speed control.

During power operation, steam dump to the condenser is controlled by comparing the Reactor Coolant System (RCS) temperature to a reference temperature based on turbine power, where a high-temperature difference in conjunction with a loss-of-load or turbine trip indicates a need for steam dump. A single controller malfunction does not cause steam dump valves to open. Interlocks are provided to block the opening of the valves unless a large turbine load decrease or a turbine trip has occurred. In addition, the reference temperature and loss-of-load signals are developed by independent sensors.

Regardless of the rate of load increase, the reactor protection system will trip the reactor in time to prevent the DNBR from going below the limit value. Increases in steam load to more than design flow are analyzed as the steam line rupture event in LR section 2.8.5.1.2.2.2.

Protection against an excessive load increase accident, if necessary, is provided by the following reactor protection system signals:

- Overtemperature ΔT (OTΔT)
- Overpower ΔT (OPΔT)
- Power range high neutron flux
- Low-pressurizer pressure

2.8.5.1.1.2.3.2 Input Parameters, Assumptions, and Acceptance Criteria

The analysis includes the following conservative assumptions:

- This accident is analyzed with the *Revised Thermal Design Procedure* (RTDP) (Reference 1). Initial reactor power, RCS pressure, and RCS temperature are assumed to be at their nominal values, consistent with steady-state full-power operation. Minimum measured flow (MMF) is assumed. Uncertainties in initial conditions are included in the DNBR limit as described in Reference 1.

- The evaluation is performed for a step-load increase of 10% steam flow from 100% of nuclear steam supply system (NSSS) thermal power (1817 MWt).

- This event is evaluated in both automatic and manual rod control.

- The excessive load increase event is evaluated for both the beginning-of-life (BOL) (minimum reactivity feedback) and end-of-life (EOL) (maximum reactivity feedback) conditions. A small (zero) moderator density coefficient at BOL and a large value at EOL are used. A positive moderator temperature coefficient is not assumed since this would provide a transient benefit. For all cases, a small (absolute value) Doppler coefficient of reactivity is assumed.

Based on its frequency of occurrence, the excessive load increase accident is considered a Condition II event as defined by the American Nuclear Society (ANS). The following items summarize the acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

- Pressure in the RCS and main steam systems (MSS) should be maintained below 110% of the design pressures.

- The peak linear heat generation rate (expressed in kW/ft) should not exceed a value that would cause fuel centerline melt.

2.8.5.1.1.2.3.3 Description of Analyses and Evaluations

Historically, four cases are analyzed, and presented in the *Updated Final Safety Analysis Report* (UFSAR), to demonstrate the plant behavior following a 10% step-load increase from 100% load. These cases are as follows:

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback
- Reactor in automatic rod control with BOL (minimum moderator) reactivity feedback
- Reactor in automatic rod control with EOL (maximum moderator) reactivity feedback

At BOL, minimum-moderator feedback cases, the core has the least-negative moderator temperature coefficient of reactivity and the least-negative Doppler-only power coefficient curve, and, therefore, the least-inherent transient response capability. Since a positive-moderator temperature coefficient would provide a transient benefit, a zero-moderator temperature coefficient was assumed in the minimum feedback cases. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its most-negative value and the most-negative Doppler-only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature. Normal reactor control systems and engineered safety systems are not required to function.

A 10% step increase in steam demand was assumed and the analysis did not take credit for the operation of the pressurizer heaters. The cases that assumed automatic rod control were analyzed to ensure that the worst case was presented. The automatic function was not required. The reactor protection system was assumed to be operable; however, reactor trip was not encountered for the cases analyzed. No active failure is postulated because no single active failure in any system or component required for mitigation would adversely affect the consequences of this accident.

Because this event is very non-limiting with respect to the departure from nucleate boiling ratio (DNBR) safety analysis criterion, an explicit LOFTRAN/RETRAN analysis was not performed as part of the EPU program. Instead, an evaluation of this event was performed. The evaluation model consists of the generation of statepoints based on generic conservative data. These generic statepoints were generated from a conservative compilation of Excessive Load Increase analyses performed for various

Westinghouse 2-loop, 3-loop and 4-loop plants. The statepoints are in the form of changes from nominal so they are applicable regardless of a plant's operating parameters. The statepoints are then compared to the core thermal limits to ensure that the DNBR limit is not violated. Three cases were included in the evaluation. These are:

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback
- Reactor in automatic rod control (both minimum/maximum moderator) reactivity feedback

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

See LR section 2.8.5.1.1.2.2.3.

2.8.5.1.1.2.3.4 Increase in Steam Flow Results

The evaluation confirmed that for an excessive load increase, the minimum DNBR during the transient will not go below the safety analysis limit value and the peak linear heat generation does not exceed the limit value, thus demonstrating that the applicable acceptance criteria for critical heat flux and fuel centerline melt are met. Following the initial load increase, the plant reaches a stabilized condition. With respect to peak pressure, the excessive load increase accident is bounded by the loss-of-electrical-load/ turbine-trip analysis. The loss-of-electrical-load/turbine-trip analysis is described in LR section 2.8.5.2.1.

2.8.5.1.1.2.3.5 Increase in Steam Flow Conclusions

The evaluation performed for the EPU demonstrates that the DNBR does not decrease below the safety analysis limit value at any time during the transient for an excessive load increase incident. Thus, no fuel or clad damage is predicted. The peak primary and secondary system pressures remain below their respective limits at all times. All applicable acceptance criteria are therefore met.

The protection features presented in LR section 2.8.5.1.1.2.3.1 provide mitigation for the excessive load increase incident such that the above criteria are satisfied.

2.8.5.1.1.2.3.6 Increase in Steam Flow References

1. WCAP-11397-P-A, (Proprietary) and WCAP-11397-A (Nonproprietary), *Revised Thermal Design Procedure*, Friedland, A. J., and Ray, S., April 1989.

2.8.5.1.1.2.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve event is equivalent to a small steamline break. It is always bounded by the analysis of the large steamline break (referred to as the hypothetical steamline break) presented in the *Ginna Updated Final Safety Analysis Report* (UFSAR), Section 15.1.4. The hypothetical steamline break is a Condition IV event that is conservatively analyzed to Condition II acceptance criteria. The inadvertent opening of a steam generator relief or safety valve is a Condition II event. Since the more severe Condition IV event is shown to meet the more restrictive Condition II acceptance criteria, it can be concluded that the inadvertent opening of a steam generator relief or safety valve also meets the Condition II acceptance criteria. As such, no explicit analysis of this event has been performed. The analysis documented in <u>LR section 2.8.5.1.2.2.1</u>, Steam System Piping Failure at Hot Zero Power (and UFSAR, Section 15.1.5) demonstrates that all applicable acceptance criteria are met for the hypothetical steam line break and, consequently, all acceptance criteria are met for this event.

2.8.5.1.1.3 Results

The Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve events have been evaluated or analyzed for the EPU. In all cases, the minimum DNBR has been shown to remain above the DNBR limit of 1.38. None of these events challenge the primary or secondary pressure limits of 2748.5 psia and 1208.5 psia, respectively.

2.8.5.1.1.4 Conclusion

The Ginna staff has reviewed the analyses of the increased heat removal by the secondary system events described above and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of these events. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-10, GDC-15, GDC-20, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

2.8.5.1.2.1 Regulatory Evaluation

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The postulated initial core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed responses of the reactor coolant and auxiliary systems
- The functional and operational characteristics of the reactor protection system
- The operator actions
- The core power excursion due to power demand created by excessive steam flow
- The variables influencing neutronics
- The results of the transient analyses

The NRC's acceptance criteria are based on:

- GDC-27, insofar as it requires that the reactivity control systems are designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.

- GDC-28, insofar as it requires that the reactivity control systems are designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

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

- GDC-35, insofar as it requires the reactor coolant system (RCS) and associated auxiliaries are designed to provide abundant emergency core cooling.

Specific review criteria are contained in the SRP, Section 15.1.5, and other guidance provided in Matrix 8 of RS-001, Revision 0.

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. However, for this event, the analyses performed demonstrate that the requirements specified by the GDC in 10CFR50, Appendix A are met. Specifically, the adequacy of the Ginna Station analyses for steam system piping failure events relative to conformance to:

- GDC-27 is described in Ginna UFSAR section 3.1.2.3.8, General Design Criterion 27 – Combined Reactivity Control System Capability. As described in this UFSAR section, the reactivity control system, in conjunction with boron addition through the emergency core cooling system, has the capability of controlling reactivity changes under postulated accident condition with appropriate margin for stuck rods. Analysis of the spectrum of steam system piping failures is provided in UFSAR section 15.1.5.

- GDC-28 is described in Ginna UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. As described in this UFSAR section, the maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling. This design basis is further discussed in UFSAR section 4.2.1.

- GDC-31 is described in Ginna UFSAR 3.1.2.4.2, General Design Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary. As described in this UFSAR section, the reactor coolant pressure boundary components were fabricated, inspected and tested in accordance with codes that were applicable at the time of fabrication and installation. The reactor vessel and reactor coolant system components are subject to an on-going inservice inspection program as required by Technical Specification. Operating limits are imposed on heat-up and cool-down rates for the reactor coolant system in accordance with the fracture toughness requirements of ASME, Section III, Appendix G.

- GDC-35 is described in Ginna UFSAR section 3.1.2.4.6, General Design Criterion 35 – Emergency Core Cooling. As described in this UFSAR section, the emergency core cooling system is available to provide an abundant supply of cooling water to the core in the event of a steam system pipe rupture. Sufficient cooling water can be supplied at a rate to maintain the core in a coolable geometry and to ensure that the clad metal-water reaction is limited. Emergency core cooling is discussed in UFSAR section 6.3.

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, components associated with the control and mitigation of transients that could result in an increase in heat removal by the secondary system were evaluated within the system that contained them.

2.8.5.1.2.2 Technical Evaluation

Steamline breaks initiated from either hot full power (HFP) conditions or from hot zero power (HZP) conditions are conservatively chosen to be analyzed to Condition II acceptance criteria. The specific acceptance criteria applied by Ginna for these events are as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met satisfies the Ginna Station current licensing basis requirements with respect to GDC-27.
- Primary and secondary pressures must remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met, including allowance made for the worst stuck rod, satisfies the Ginna Station current licensing basis requirements with respect to GDC-28 and GDC-31.
- Only the HZP case assumes emergency core cooling system (ECCS) actuation (i.e., safety injection (SI) flow) for mitigation. The analysis performed demonstrates that the SI system has sufficient capacity to mitigate the event. The HFP transient is terminated via a reactor trip. The post-trip portion of the HFP transient is bounded by the HZP case. Thus, demonstrating adequate capacity for the HZP case also demonstrates adequate capacity for the post-reactor trip portion of the HFP transient. The analyses demonstrate that the Ginna Station current licensing basis requirements with respect to GDC-35 are met.

The discussion below demonstrates that all applicable acceptance criteria are met for these events by Ginna Station at EPU conditions.

2.8.5.1.2.2.1 Steam System Piping Failure at Hot Zero Power

2.8.5.1.2.2.1.1 Introduction

The steam release from a major rupture of a main steam pipe will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a positive reactivity insertion and subsequent reduction in core shutdown margin. If the most-reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a concern primarily because of the high-power peaking factors that would exist assuming the most-reactive RCCA is stuck in its fully withdrawn position. The core is ultimately shut down by boric acid injection delivered by the ECCS.

The major rupture of a main steam pipe is the most-limiting cooldown transient. It is analyzed at HZP conditions with no decay heat (decay heat would retard the cooldown, thus reducing the return to power). A detailed discussion of this transient with the most limiting break size is presented below.

The primary design features which provide protection for steam pipe ruptures are:

- Actuation of the SI system from any of the following:
  - Two-out-of-three pressurizer low-pressure signals.
  - Two-out-of-three low-pressure signals in any steam line.
  - Two-out-of-three high-containment pressure signals.
- If the reactor trip breakers are closed, reactor trip can be actuated from overpower neutron flux, overpower delta T (OPΔT), or upon actuation of the SI system.
- Redundant isolation of the main feedwater lines to prevent sustained high-feedwater flow that will cause additional cooldown. In addition to the normal control action which will close the main feedwater control valves, an SI signal will also rapidly close all feedwater control valves as well as the feedwater isolation valves. Details of the operation of the control and isolation valves are provided in Reference 3.
- Trip of the fast-acting main steamline isolation valves (MSIVs), on the following:

- Two-out-of-three high containment pressure signals.
- One-out-of-two high-high steam flow signals in a steam line in coincidence with any safety injection signal.
- One-out-of-two high-steam flow signals in a steam line in coincidence with two-out-of-four indications of low-reactor coolant T_{avg} and any SI signal.

Each steam line is provided with a main steam isolation valve which isolates flow in the forward direction, and a main steam non-return valve, which isolates flow in the reverse direction. Thus, even with a single failure of any valve, no more than one steam generator can blow down, no matter where the break is postulated. The unaffected steam generator is still available for dissipation of decay heat after the initial transient is over.

Following blowdown of the faulted steam generator, the unit can be brought to a stabilized hot-standby condition through control of the auxiliary feedwater (AFW) flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system (AFWS).

2.8.5.1.2.2.1.2 Input Parameters, Assumptions, and Acceptance Criteria

The following summarizes the major input parameters and/or assumptions used in the main steam line rupture event:

- HZP conditions were modeled with two loops in service with and without offsite power available. A case with one loop in service with offsite power available was also modeled.
- For Ginna, a 1.4 ft^2 break was analyzed for the Babcock and Wilcox (BWI) steam generators, since they are designed with a flow restrictor built into the steam exit nozzle. The assumed steam generator tube plugging level was 0%.
- All control rods were inserted except the most reactive RCCA, which was assumed to be stuck out of the core.
- The shutdown margin was 1.30% Δk/k and 1.80% Δk/k for the two-loop and one-loop operation cases, respectively.

For acceptance criteria see LR section 2.8.5.1.2.2, above.

2.8.5.1.2.2.1.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (reference 1) computer code was performed in order to determine the plant transient conditions following a main steam line break. The code models the core neutron kinetics, RCS, pressurizer, steam generators, SI system and the AFWS; and computes pertinent variables, including the core heat flux, RCS temperature, and pressure. A conservative selection of those conditions were then used to develop core models which provide input to the detailed thermal and hydraulic digital computer code, VIPRE (reference 2), to determine if the DNB design basis is met.

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

In addition to the evaluations described in the Ginna UFSAR, the systems and components used to control and mitigate transients associated increases in heat removal by secondary systems were evaluated for the Ginna License Renewal within the systems that contain them. The evaluations are documented in the License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May 2004. The aging management review for these systems and components are addressed in the License Renewal SER section 3.0. The EPU does not add any new structures/ components used to control or mitigate transients associated with an increase in heat removal by the secondary system.

2.8.5.1.2.2.1.4 Steam System Piping Failure at HZP Results

For Ginna, the most limiting main steamline rupture at HZP is the case with two-loops in service in which offsite power is assumed to be available since the steam generator inventory is highest and the reactor coolant pumps are available to circulate RCS flow.

The calculated sequence of events for the most limiting case is shown in Table 2.8.5.1.2.2.1-1.

Figures 2.8.5.1.2.2.1-1 through 2.8.5.1.2.2.1-4 show the transient results for the most limiting case for Ginna. These figures show transient results following a 1.4 ft^2 main steamline rupture at initial no-load conditions with offsite power available. Since offsite power is assumed available, there is full reactor coolant flow.

Should the core be critical at near zero power when the rupture occurs, the initiation of SI via a low-steam line pressure signal will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the main steam isolation valves in conjunction with the main steam non-return valves.

As shown in Figure 2.8.5.1.2.2.1-3 the core attains criticality with the RCCAs inserted (i.e., with the plant shutdown assuming one stuck RCCA) before boron solution from the ECCS enters the RCS.

A DNB analysis was performed for the limiting point in the transient which determined that the DNB design basis is met. The peak heat flux (13.3%) and minimum DNBR (2.58) occur approximately 54 seconds after the break occurs. The DNBR Limit used for this event is 1.566. Primary and secondary pressure limits are not challenged because primary and secondary pressures decrease from their initial values during the transient.

The only criterion that could be challenged during this event is the one that states that the critical heat flux should not be exceeded. The analysis demonstrated that this criterion was met by showing that the minimum DNBR did not go below the limit value at any time during the transient.

The results of the major rupture of a main steam pipe event indicate that the DNB design basis is met. The calculated minimum DNBR is 2.58 compared to a limit of 1.566. Primary and secondary pressure limits are not challenged because primary and secondary pressures decrease from their initial values during the transient. Therefore, this event does not adversely affect the core or the RCS, and all applicable acceptance criteria are met.

The HZP steam line break analysis was also performed for a break size just large enough to actuate a High Steam Flow (HSF) signal in both loops (1.5E6 lbs/hr), coincident with a Safety Injection (SI) signal on Low Pressurizer Pressure (1700 psig) and Low Tave (530°F). The break was assumed to be located downstream of the MSIVs in order to blow down both steam generators and maximize cool down. The break location upstream of the MSIVs is bounded by the HZP steam line break analysis of the maximum break size described above since only one steam generator is available to blow down in that case. The analysis for this (minimum) break size was performed with no credit for automatic steam line isolation. After 10 minutes it was assumed that operators manually close the MSIVs to terminate the event. The results of this analysis were bounded by the HZP steam line break analysis of the maximum break size described above. Therefore, for any breaks smaller than the minimum break size analyzed, operator action in 10 minutes is sufficient to provide protection. For breaks larger than the minimum break analyzed, automatic steam line isolation will occur and the transient will be bounded by the maximum break size HZP analysis. This minimum break size HZP steam line break analysis thereby establishes an acceptable basis for the High Steam Flow Steam Line Isolation setpoint coincident with SI and Low Tave.

2.8.5.1.2.2.1.5 Steam System Piping Failure at HZP References

1. WCAP-14882-P-A (Proprietary), April 1999 and WCAP-15234-A (Nonproprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, D. S., et al., May 1999.

2. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X. et al., October 1999.

3. Letter from Mary G. Korsnick (Ginna) to Donna M. Skay (NRC), dated April 29, 2005, Subject: License Amendment Request Regarding Main Feedwater Isolation Valves.

Table 2.8.5.1.2.2.1-1 Ginna Station Time Sequence of Events – Steam System Piping Failure		
Case	**Event**	**Time (sec)**
Reactor at HZP with Offsite Power Available (Unisolable Steam Release Paths Case)	Double-Ended Guillotine Break Occurs	0.0
	Low Steam Pressure SI System Actuation Setpoint Reached	1.4
	MSIVs Closed 7 Seconds After SI System Actuation Signal	8.4
	High-Head SI Pump At Rated Speed 12 Seconds After SI System Actuation Signal	15.4
	Reactor Becomes Critical	22.7
	Main Feedwater Flow Isolated 32 Seconds After SI System Actuation Signal	33.4
	Power Reaches Maximum Level	53.0
	Time of Minimum DNBR	54.2
	Reactor Returns Subcritical	67.5

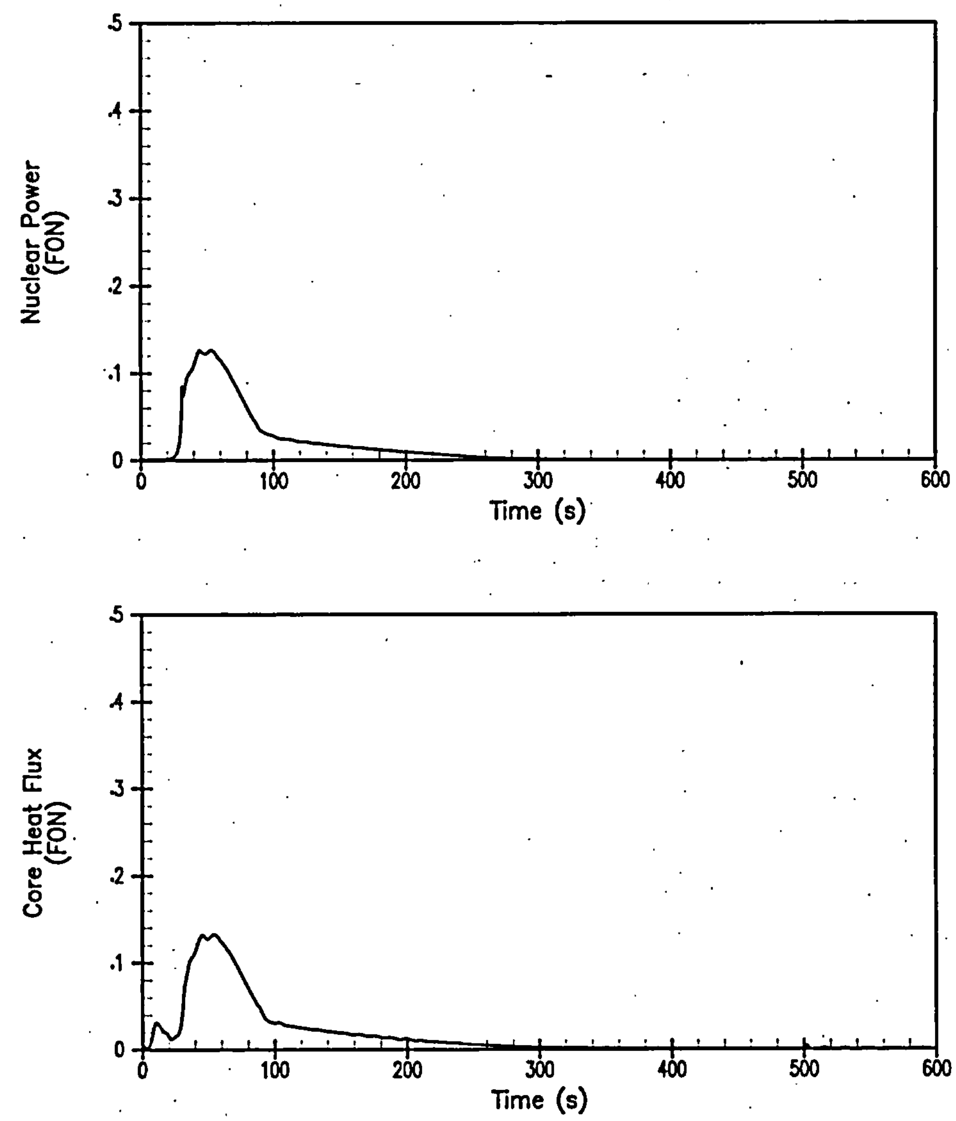


Figure 2.8.5.1.2.2.1-1
Ginna Station
Steam System Piping Failure at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Nuclear Power, and Core Heat Flux vs. Time

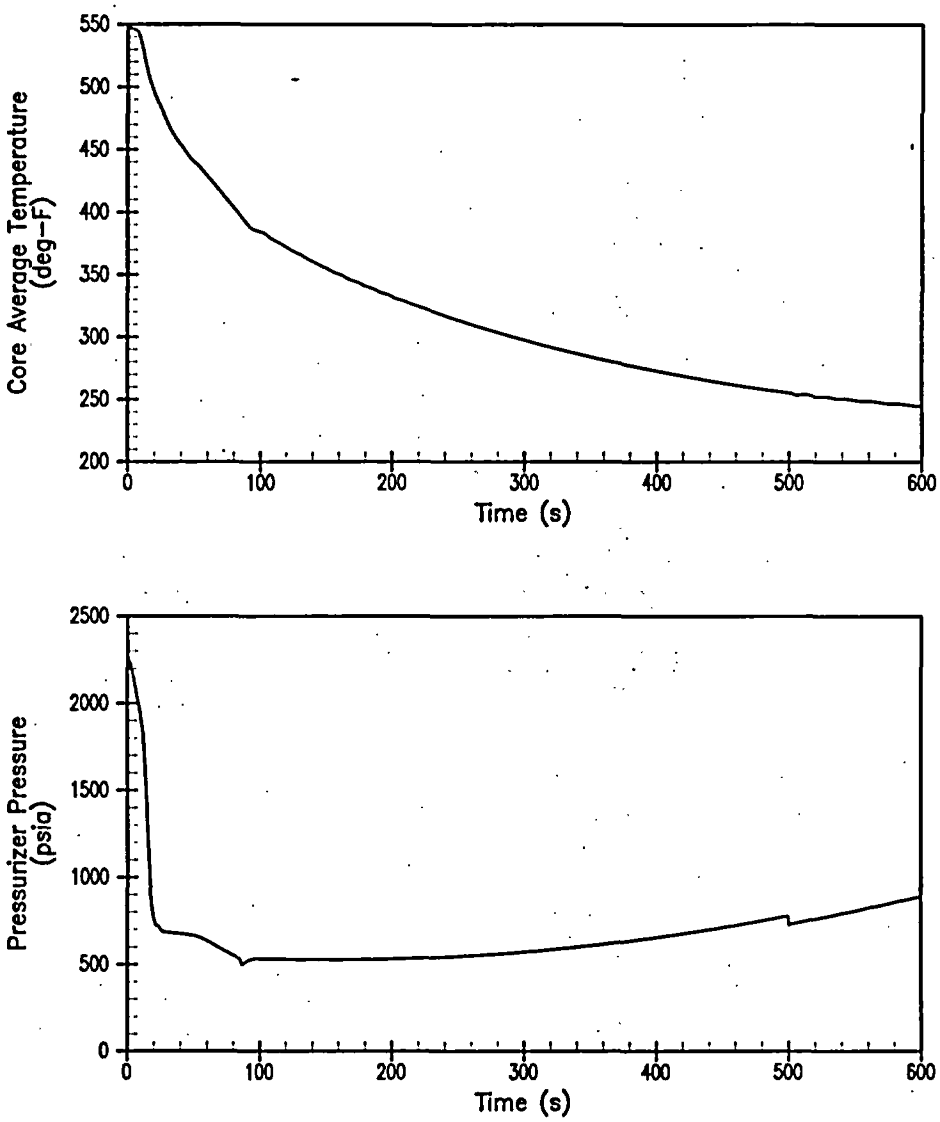


Figure 2.8.5.1.2.2.1-2
Ginna Station
Steam System Piping Failure at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Core Average Temperature, and Pressurizer Pressure vs. Time

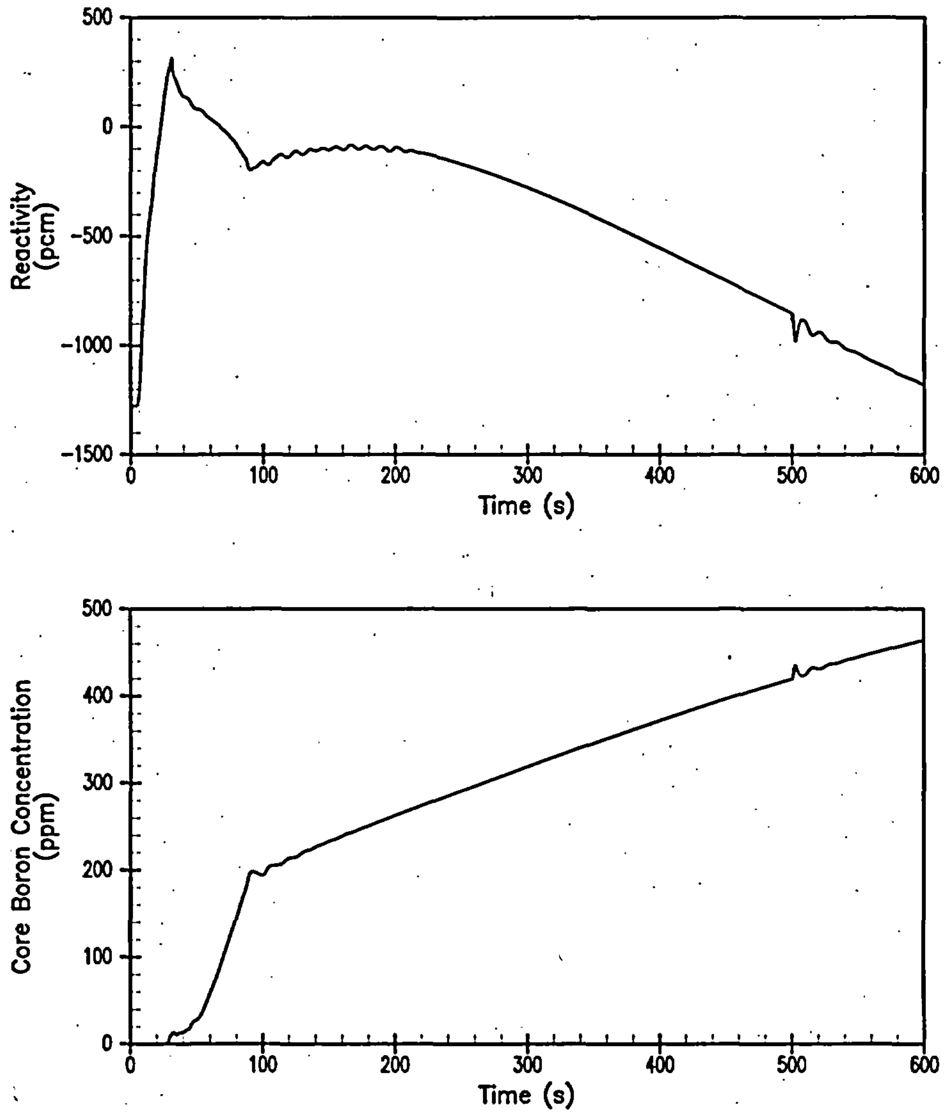


Figure 2.8.5.1.2.2.1-3
Ginna Station
Steam System Piping Failure at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Reactivity, and Core Boron Concentration vs. Time

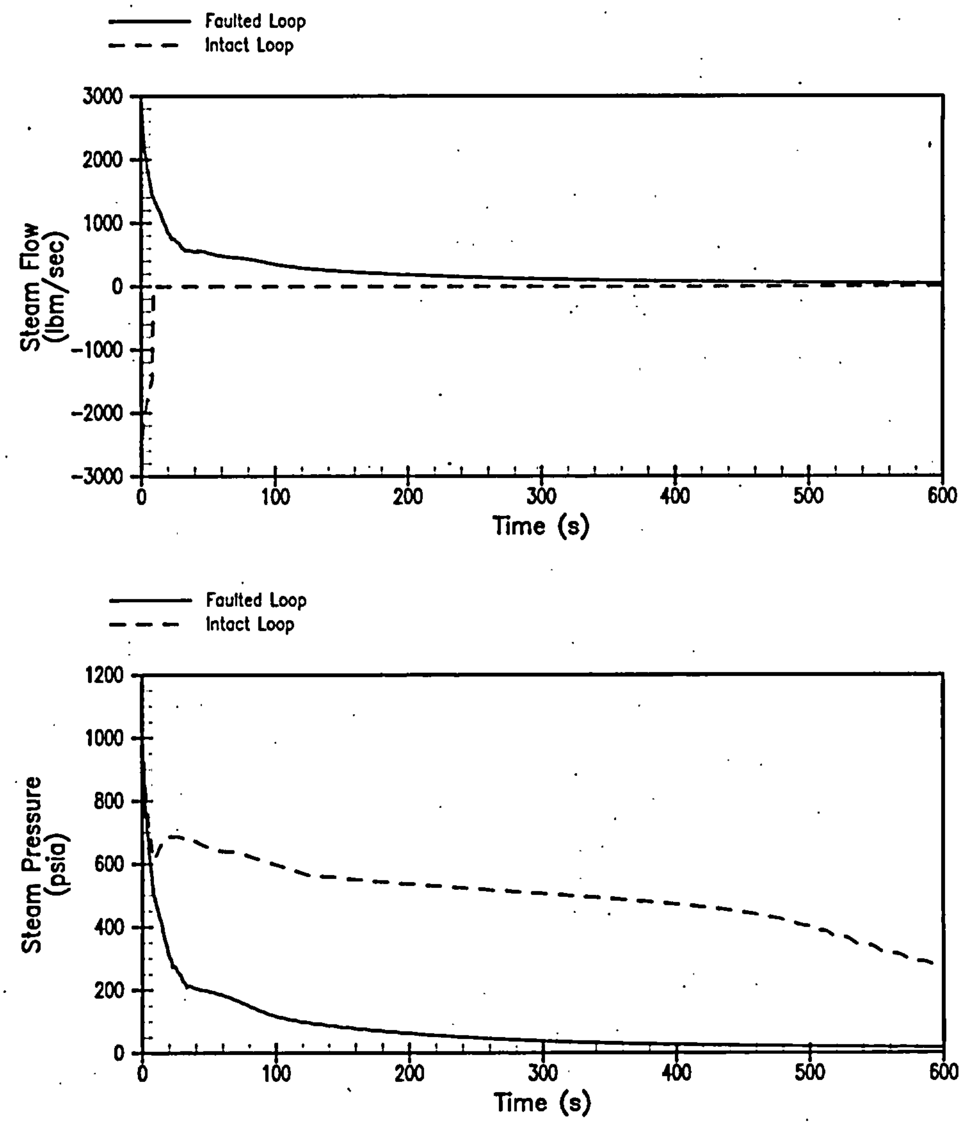


Figure 2.8.5.1.2.2.1-4
Ginna Station
Steam System Piping Failure at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Steam Flow, and Steam Pressure vs. Time

2.8.5.1.2.2.2 Steam System Piping Failure at Full-Power

2.8.5.1.2.2.2.1 Introduction

This section describes the analysis of a steam system piping failure occurring from at-power initial conditions to demonstrate that core protection is maintained prior to and immediately following reactor trip. The at-power case is currently not analyzed for Ginna but has been added for EPU for completeness.

2.8.5.1.2.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Limiting transient condition statepoints were generated using the *Revised Thermal Design Procedure* (RTDP) (Reference 1). For RTDP applications, uncertainties on reactor coolant system (RCS) initial conditions (temperature, pressure, power, and flow) are included in the development of the departure from nucleate boiling ratio (DNBR) limit value. When RTDP is not applicable, uncertainties are included in the initial conditions or are conservatively applied to the limiting transient condition in the calculation of the minimum DNBR.

- Initial conditions – The initial core power, RCS temperature, and RCS pressure are assumed to be at their nominal steady-state, full-power values when generating the transient statepoints. Uncertainties are already explicitly included in the DNBR limit calculations.

- RCS flow – Minimum measured RCS flow is assumed when generating the transient statepoints and in the DNBR calculations. The flow uncertainty is included in the DNBR limit calculations. The initial loop flows are assumed to be symmetric.

- RCS average temperature – The full-power RCS T_{avg} range is from 564.6° to 576.0°F. Since the full-power steamline-rupture-core-response event is a DNB event, assuming a maximum RCS average temperature of 576.0°F is limiting.

- Feedwater temperature – The main feedwater analytical temperature range is from 390° to 435°F. A nominal feedwater temperature of 435°F is assumed for this event. Sensitivity studies have shown that HFP SLB results are not influenced by the assumed initial feedwater temperature.

- Break size – The event is analyzed over a spectrum of break sizes in order to identify the most limiting overpower condition, which is typically the largest break to produce a reactor trip on overpower delta T (OPΔT). The Babcox & Wilcox (BWI) steam generators used in Ginna have a steam exit nozzle flow restrictor that limits the flow area to 1.396 ft^2. Therefore the analysis modeled break sizes up to 1.4 ft^2. In addition, the largest break size for which there is no reactor trip is examined to determine if it is more limiting with respect to peak power level.

- Reactivity coefficients – The analysis assumed maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.

- Protection system – The protection system features that mitigate the effects of a steamline break are described in LR section 2.8.5.1.2.2.1. This analysis only considers the initial phase of the transient from at-power conditions. Protection in this phase of the transient is provided by reactor trip, if necessary. The fluid conditions at the time of reactor trip for a hot full power case are less severe than the initial conditions for a hot zero power case with respect to the potential for a return to critical. Thus, the post-trip portion of an at-power steamline break is bounded by the hot zero power steamline break analysis. LR section 2.8.5.1.2.2.1, Steam System Piping Failure at Hot Zero Power, presents the analysis of the bounding transient following reactor trip, where other protection system features are actuated to mitigate the effects of the steamline break.

- Control systems – The only control system that is assumed to function during a full-power-steamline-rupture-core-response event is the main feedwater system. For this event, the feedwater flow is set to match the steam flow.

Depending on the size of the break, this event is classified as either a Condition III (infrequent fault) or Condition IV (limiting fault) event. However, the analysis was done to the more conservative Condition II acceptance criteria. The acceptance criteria for this event are consistent with those stated in LR section 2.8.5.1.2.2..

2.8.5.1.2.2.2.3 Description of Analysis and Evaluations

The analysis of the steamline break at-power for the EPU was performed as follows:

- The RETRAN code (Reference 2) was used to calculate the nuclear power, core heat flux, and RCS temperature and pressure transients resulting from the cooldown following the steamline break.

- The core radial and axial peaking factors were determined using the thermal-hydraulic conditions from RETRAN as input to the nuclear core models. A detailed thermal-hydraulic code, VIPRE (Reference 3), was used to calculate the DNBR for the limiting time during the transient. The DNBR calculations were performed using the WRB-1 DNB correlation and RTDP.

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

See LR section 2.8.5.1.2.2.1.3.

2.8.5.1.2.2.2.4 Steam System Piping Failure at Full-Power Results

The limiting break size from the spectrum of break sizes analyzed is 1.4 ft^2, with a minimum DNBR of 1.395/1.392 (thimble/typical), and a peak fuel rod power of 22.67 kW/ft. The sequence of events for the limiting case with a 1.4 ft^2 break is shown in Table 2.8.5.1.2.2.2-1. Plots for this limiting case are provided in Figures 2.8.5.1.2.2.2-1 through 2.8.5.1.2.2.2-4.

The 1.4 ft^2 break size is the most limiting break size with respect to peak heat flux and minimum DNBR for the full-power-steamline-rupture-core-response event.

The DNB design basis is met. Therefore, this event does not adversely affect the core or RCS, and all applicable criteria are met.

The results and conclusions of the analysis performed for the steam system piping failure at full-power for the nuclear steam supply system (NSSS) power of 1817 MWt bound and support the implementation of EPU. Furthermore, the results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload process.

2.8.5.1.2.2.2.5 Steam System Piping Failure at Full-Power References

1. WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), *Revised Thermal Design Procedure, Friedland*, A. J. and Ray, S., April 1989.

2. WCAP-14882-P-A (Proprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, D.S., et al., April 1999.

3. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X. et al., October 1999.

Table 2.8.5.1.2.2.2-1 Time Sequence of Events – Steam System Piping Failure at Full-Power (Core Response – 1.4 ft^2 break)	
Event	**Time (sec)**
Steam Line Ruptures	0.01
Overpower ΔT Reactor Trip Setpoint Reached	10.90
Rods Begin to Drop	12.90
Minimum DNBR Occurs	13.25
Peak Core Heat Flux Occurs	13.25

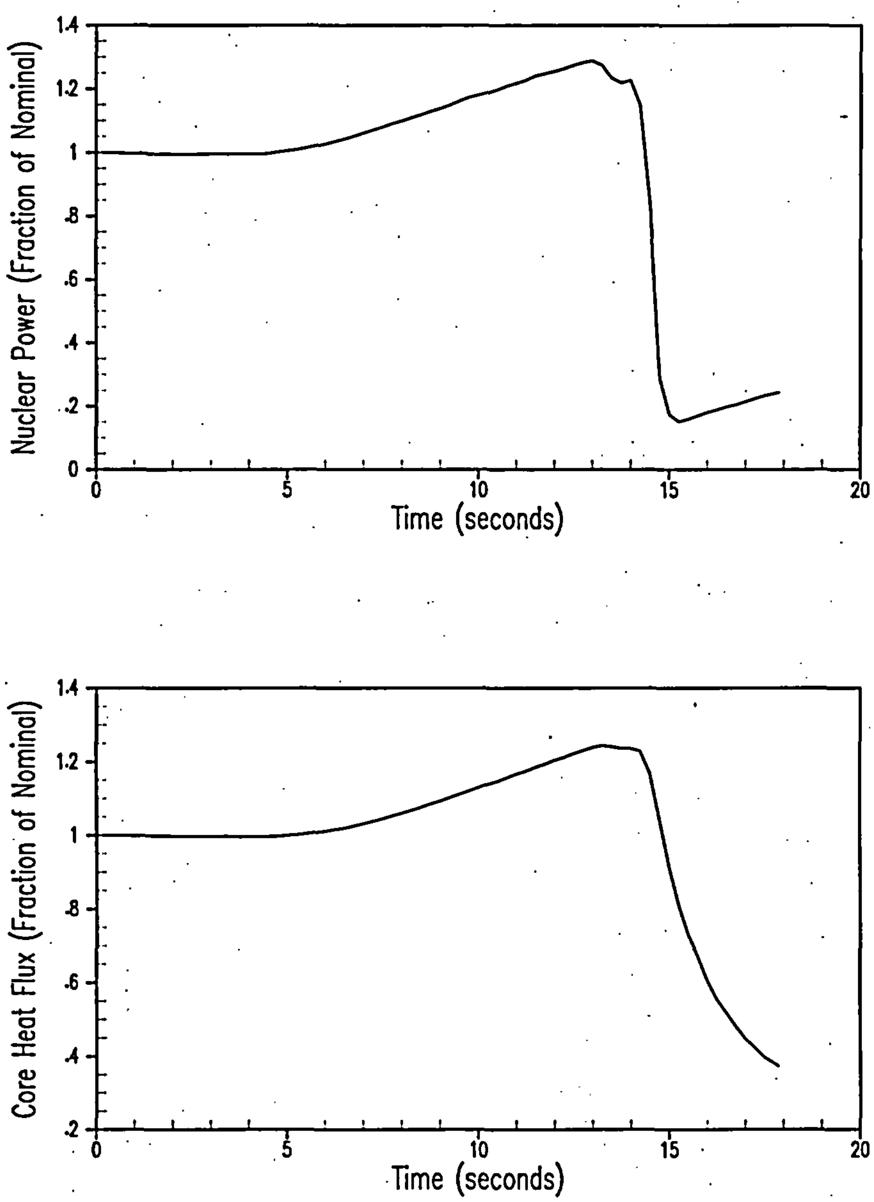


Figure 2.8.5.1.2.2.2-1
Steam System Piping Failure at Full-Power – 1.4 ft² Break
Nuclear Power and Core Heat Flux vs. Time

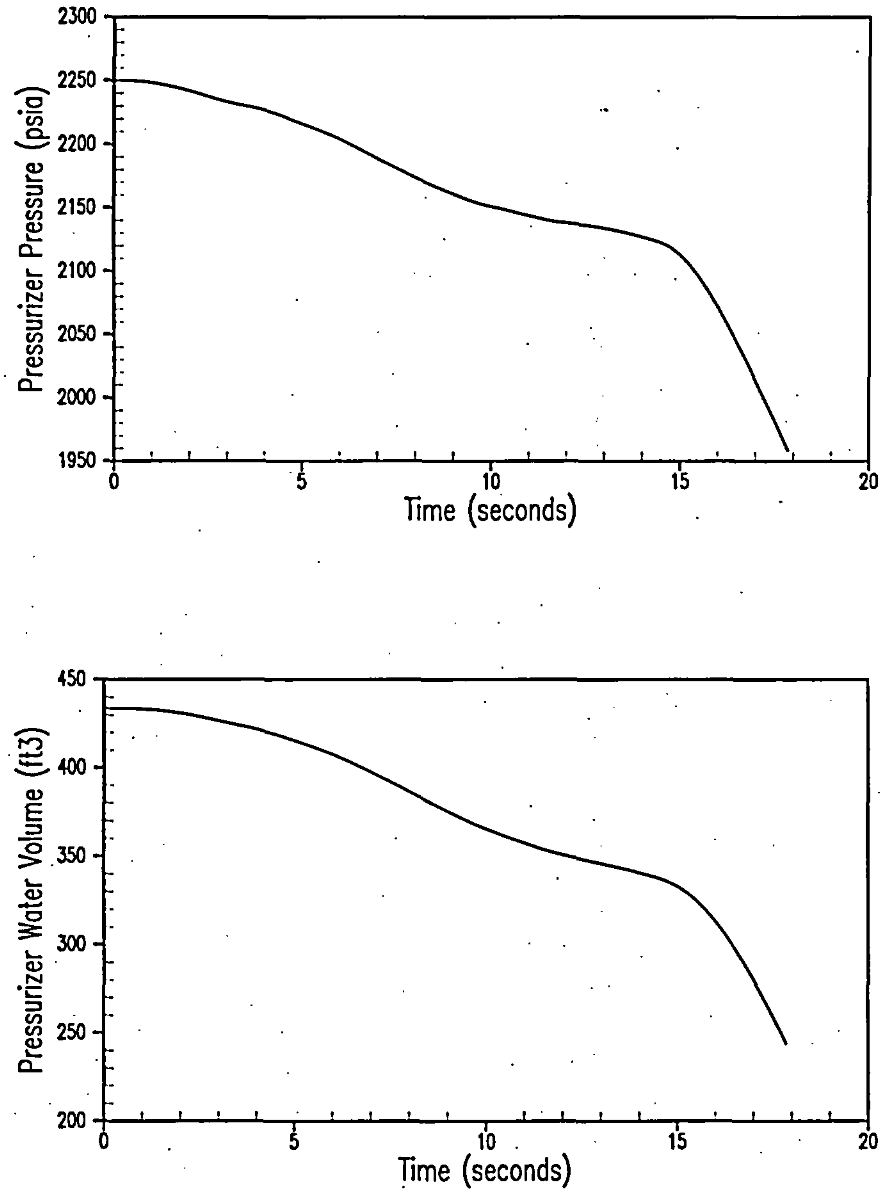


Figure 2.8.5.1.2.2.2-2
Steam System Piping Failure at Full-Power – 1.4 ft² Break
Pressurizer Pressure and Pressurizer Water Volume vs. Time

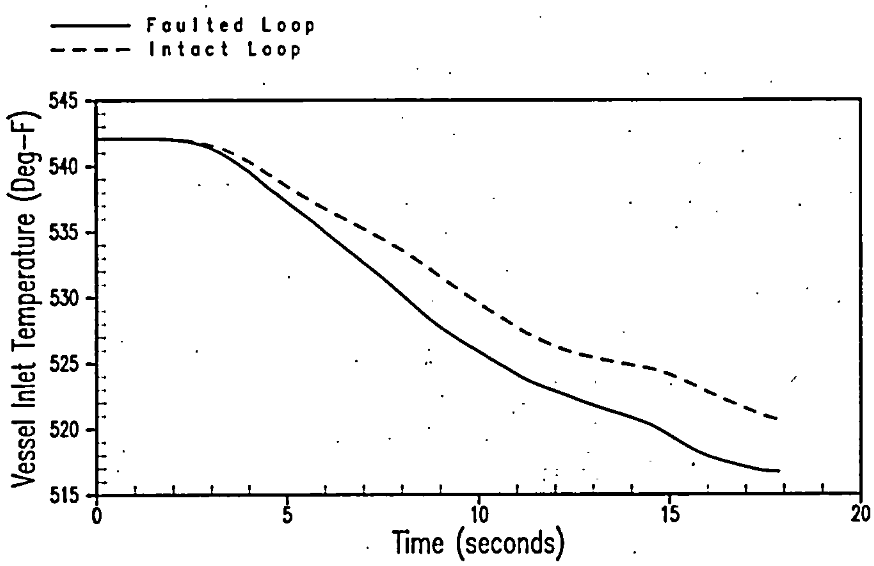
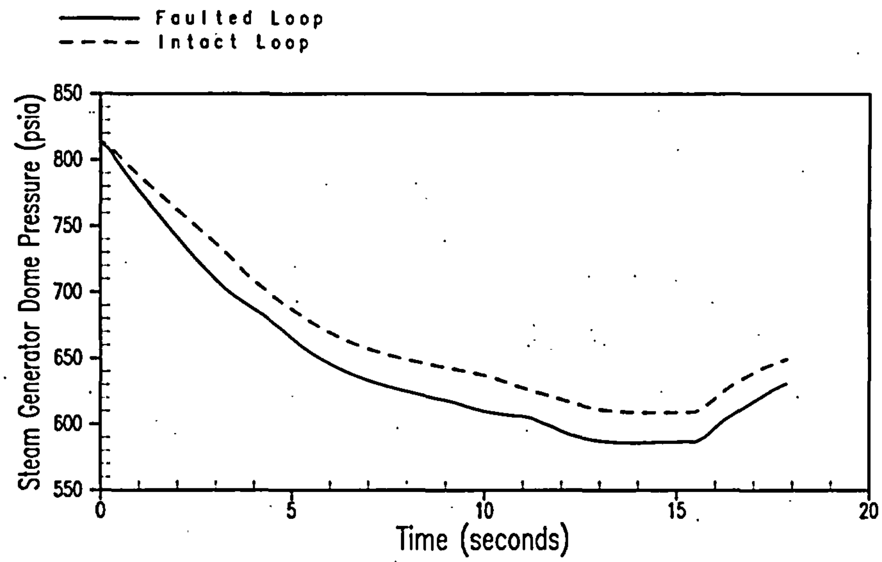


Figure 2.8.5.1.2.2.2-3
Steam System Piping Failure at Full-Power – 1.4 ft² Break
Vessel Inlet Temperature vs. Time



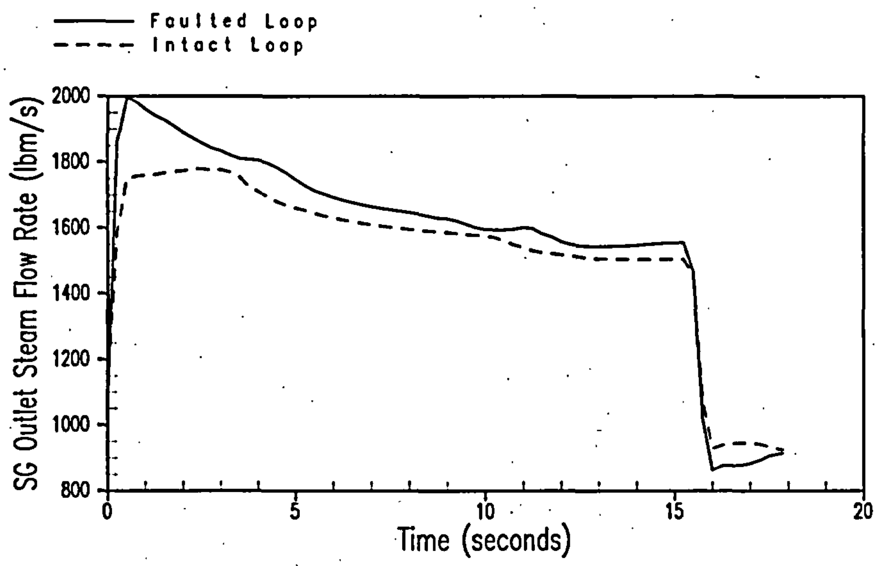


Figure 2.8.5.1.2.2.2-4
Steam System Piping Failure at Full-Power – 1.4 ft² Break
Steam Generator Dome Pressure and Steam Generator Outlet Steam Flow Rate vs. Time

2.8.5.1.2.3 Conclusion

The Ginna staff has reviewed the analyses of the steam system piping failure events described above and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the reactor coolant pressure boundary pressure limits will not be exceeded, the reactor coolant pressure boundary will behave in a nonbrittle manner, the probability of propagating fracture of the reactor coolant pressure boundary is minimized, and adequate core cooling will be provided. Based on this, the Ginna staff concludes that the Ginna Station will continue to meet its current licensing basis with respect to the requirements of GDC-27, GDC-28, GDC-31, and GDC-35 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal By the Secondary System

2.8.5.2.1 Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum

2.8.5.2.1.1 Regulatory Evaluation

A number of initiating events can result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient.

Loss of external electrical load can cause a sudden heat addition to the reactor coolant system (RCS) resulting in an increase in RCS temperature and pressure and an increase in pressurizer level and affect fuel design parameters and core reactivity. Similar effects to the RCS will be experienced following instantaneous turbine trip or loss of condenser vacuum during power operation.

The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The NRC acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems is designed with sufficient margin to ensure that the design condition of the reactor coolant pressure boundary is not exceeded during any condition of normal operation
- GDC-26, insofar as it requires that a reactivity control system is provided, and is capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Specific review criteria are contained in the SRP, Section 15.2.1-5, and other guidance provided in Matrix 8 of RS-001.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), 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 are published in NURG-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 primary and secondary systems design relative to conformance to

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 –Reactor Design. As described in this UFSAR section, the reactor core design in combination with coolant, control, and protection systems, provides margins that ensure that fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Further discussion of this design is provided in UFSAR Chapter 4.

- GDC-15 is described in 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. The analysis of the ability of the plant to safely undergo all anticipated transients with peak pressures below 2485 psig is provided in UFSAR Chapter 15. In addition, overpressure protection is prevented by a combination of automatic control and pressure relief devices.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, two means of controlling reactivity are employed at the Ginna Station. Control rod drive mechanisms regulate the position of neutron absorbers within the core. The control rods are designed to shut the reactor down with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The other reactivity control system employs the chemical and volume control system to regulate the concentration of boric acid, a neutron absorber, in the reactor coolant system. Reactivity control system redundancy and capability are further discussed in UFSAR sections 4.3 and 9.3.4.

In addition to the evaluations described in the UFSAR, the analyses of events at Ginna Station that can result in a sudden reduction in steam flow, and hence increase in reactor coolant system pressure, 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.

2.8.5.2.1.2 Technical Evaluation

2.8.5.2.1.2.1 Introduction

A major load loss on the plant (see UFSAR Section 15.2.2) can result from either a loss-of-external-electrical load or from a turbine trip. A loss-of-external-electrical load can result from an abnormal variation in network frequency or other adverse network operating condition. In either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps (RCPs).

As discussed in LR section 2.5.5.3, "Turbine Bypass," the plant is designed to accept a 50% rapid decrease (200% per minute) in electrical load while operating at full power, or a complete loss of load while operating below 50% power without actuating a reactor trip with all nuclear steam supply system (NSSS) control systems in automatic. A 50% load loss is handled by the steam dump system , the rod control system , and the pressurizer (which absorbs the change in coolant volume due to the heat addition resulting from the load rejection). Should a complete loss of load occur from full power, the reactor trip system will automatically actuate a reactor trip.

The most likely source of a complete loss of load on the NSSS is a trip of the turbine generator. In this case, there is a direct reactor trip signal derived from either the turbine auto-stop oil pressure or a closure of the turbine stop valves, provided the reactor is operating above 50% power. Reactor temperature and pressure do not increase significantly if the steam dump system and pressurizer pressure control system are functioning properly. However, the RCS and main steam system (MSS) pressure-relieving capacities are designed to ensure the safety of the plant without requiring the use of automatic rod control, pressurizer pressure control, and/or steam dump control systems. In this analysis, the behavior of the plant is evaluated for a complete loss-of-steam load from full power without direct reactor trip in order to demonstrate the adequacy of the pressure-relieving devices and core protection margins.

In the event the steam dump valves fail to open following a large loss of load, the main steam safety valves (MSSVs) can lift and the reactor can be tripped by the high pressurizer pressure signal, the overtemperature ΔT signal, or the overpower ΔT signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves (PSVs) and MSSVs are sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves (PORVs), automatic rod control, or the direct reactor trip on turbine trip.

2.8.5.2.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Three cases were analyzed for a total loss of load from full-power conditions:

- With automatic pressure control - departure from nucleate boiling ratio (DNBR) case
- With automatic pressure control and minimum steam generator tube plugging (SGTP) - SGTP and zero steam generator tube fouling (MSS pressure case)
- Without automatic pressure control - RCS pressure case

The primary concern for the case analyzed with pressure control is the minimum DNBR. The primary concern for the case analyzed with pressure control, minimum SGTP, and zero steam generator tube fouling was maintaining MSS pressure below 110% of the secondary side design pressure. The primary concern for the case analyzed without pressure control was maintaining RCS pressure below 110% of the primary side design pressure.

The key attributes of the analyses are summarized as follows.

Initial Operating Conditions

The DNBR case was analyzed using the revised thermal design procedure (RTDP) (reference 1). RCS temperature and pressure were assumed to be at their nominal values consistent with steady-state, full-power operation. Minimum measured flow was modeled. Uncertainties in initial conditions were included in the DNBR limit as described in WCAP-11397 (reference 1).

The remaining cases were analyzed using the standard thermal design procedure (STDP). Initial uncertainties on reactor coolant flow, temperature, and pressure were applied in the conservative direction to obtain the initial plant conditions for the transient. The analysis modeled thermal design flow.

The nominal NSSS full power including uncertainties was 1817 MWt.

Reactivity Coefficients

The total loss-of-load transient was analyzed conservatively with minimum reactivity feedback (beginning of core life). All cases assumed the least-negative Doppler power coefficient and a 0 pcm/°F moderator temperature coefficient, which bounded part power conditions, assuming a positive moderator temperature coefficient. Minimum reactivity conditions were conservative since reactor power was maintained until the time of reactor trip, which exacerbated the calculated minimum DNBR and maximum RCS and MSS pressures.

Reactor Control

Manual rod control was modeled for all cases. If the reactor had been in automatic rod control, the control rod banks would have driven into the core prior to reactor trip, thereby reducing the severity of the transient.

Pressurizer Spray, PORVs, and Safety Valves

The loss-of-load event was analyzed both with and without pressurizer pressure control. The pressurizer PORVs and sprays were assumed operable for the DNBR case to minimize the increase in primary pressure, which was conservative for the DNBR criterion. The pressurizer PORVs and sprays were assumed operable for the MSS peak pressure case to minimize the increase in primary pressure, which delayed reactor trip, resulting in a conservatively high calculated peak secondary side pressure. The RCS pressure case was analyzed without pressure control to conservatively maximize the RCS pressure increase. In all cases, the MSSVs and pressurizer safety valves were operable.

The pressurizer safety valve (PSV) model included the effects of the PSV loop seals. In the peak RCS pressure case, opening the (PSVs) was delayed by 0.8 seconds to purge the water-filled loop seals. The remaining cases did not model the loop seal purge delay since the intent of those cases was to minimize RCS pressure.

A total PSV setpoint tolerance of -3%/+2.3% was supported in the analysis. For the DNBR case and MSS peak pressure case (pressurizer pressure control cases), the negative tolerance was applied to conservatively reduce the setpoint. For the case analyzed for peak RCS pressure, the positive tolerance was applied to conservatively increase the setpoint pressure. An additional +1% pressure uncertainty was included to account for setpoint shift due to the presence of the loop seals.

Feedwater Flow

Main feedwater flow to the steam generators was assumed to be lost at the time of turbine trip. No credit was taken for auxiliary feedwater flow, however, auxiliary feedwater flow was eventually initiated and a stabilized plant condition reached.

Reactor Trip

Only the overtemperature ΔT, high-pressurizer pressure, and overpower ΔT (OTΔT) reactor trips are assumed operable for the purposes of this analysis. No credit is taken for a reactor trip on high pressurizer level or the direct reactor trip on turbine trip. The DNBR case and MSS peak pressure case typically trip on the OTΔT signal, while the RCS peak pressure case trips on the high pressurizer pressure signal.

Secondary Side Steam Release

No credit is taken for the operation of the steam dump system or steam generator atmospheric relief valves (ARVs). This assumption maximizes secondary pressure. The MSSV model for all cases includes an allowance of +1.4% for safety valve setpoint tolerance and an accumulation model that assumes that the safety valves are wide open once the pressure exceeds the setpoint (plus tolerance) by 5 psi.

Single Failures

The limiting single failure is failure of one train of the reactor trip system (RTS). The remaining (operable) train trips the reactor. The MSSVs and pressurizer safety valves are considered passive components and are assumed not to fail to open on demand.

Steam Generator Conditions

Maximum (10%) steam generator tube plugging is assumed in the DNBR case and RCS peak pressure case since it maximizes the RCS temperature transient following event initiation. However, the MSS peak pressure case is analyzed at zero steam generator tube plugging and zero steam generator tube fouling since this conservatively maximizes the initial steam generator pressure (i.e., the initial pressure is closer to the MSSV opening setpoint). This assumption is slightly more limiting with respect to the secondary side pressure transient.

The specific acceptance criteria applied by Ginna for these events were as follows:

- The departure from nucleate boiling ration (DNBR) remains above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met meets the Ginna Station current licensing basis requirements with respect to GDC-10.

- Primary and secondary pressures remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-15.

- Reactivity changes are reliably controlled to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ensuring that appropriate margin for malfunctions, such as stuck rods, is accounted for in the safety analysis assumptions. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-26.

Based on its frequency of occurrence, the loss-of-external-electrical load/turbine trip accident is considered a Condition II event as defined by the American Nuclear Society. The specific acceptance criteria for this accident, as stated in the SRP, are as follows:

- Pressure in the reactor coolant and main steam systems are maintained below 110% of the design values (an RCS pressure limit of 2748.5 psia and secondary side pressure limit of 1208.5 psia).

- Fuel cladding integrity is maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit for PWRs (the applicable safety analysis DNBR limit is 1.38).

- An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

This criterion is satisfied by verifying that the pressurizer does not fill (i.e., total pressurizer water volume remains less than 818.6 ft^3 including the surge line).

- An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, fuel failure is assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There is no loss of function of any fission product barrier other than the fuel cladding.

These criteria are satisfied by verifying that DNBR remains above the 95/95 DNBR limit.

2.8.5.2.1.2.3 Description of Analyses and Evaluations

For the loss-of-external-electrical load/turbine trip event, the behavior of the unit is analyzed for a complete loss of steam load from full power without a direct reactor trip.

A detailed analysis using the RETRAN (Reference 2) computer code is performed to determine the plant transient conditions following a total loss of load. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system. RETRAN computes pertinent variables, including the pressurizer pressure, steam generator pressure, and reactor coolant average temperature. Additional discussion of the RETRAN code is contained in LR section 2.8.5.0.9, "Computer Codes Utilized."

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

The NRC issued the Ginna License Renewal Safety Evaluation Report (SER), NUREG-1786, in May 2004. The plant systems and system components whose performance is relied upon to support the inputs, assumptions, and results of the analyses described in this section for transients resulting in unplanned sudden decreases in heat removal by the secondary system are not being modified for the proposed EPU. Those systems and system components are described in SER sections 2.3, "Mechanical Systems," and 2.5, "Electrical, Instrumentation and Control Systems," for Scoping and Screening Results and SER sections 3.1, "Reactor Coolant System," 3.2, "Engineered Safety Features," 3.4, "Steam & Power Conversion Systems, and 3.6, "Electrical, Instrumentation and Control Systems" for Aging Management Review. EPU activities do not add any new components nor do they introduce any new functions for existing components associated with these analyses that would change the license renewal evaluation boundaries. The primary and secondary systems performance capability described in this LR section for the proposed EPU involves analytical techniques and methodology are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of SRP 15.2.1-5 with respect to GDC- 10, 15, and 26. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to plant components associated with the analyses described in this LR section.

2.8.5.2.1.3 Results

The calculated sequence of events for the three loss-of-external-electrical load/turbine trip cases are presented in Table 2.8.5.2.1-1. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.2.1-2. In all cases, the EPU analyses are more limiting than the previous analyses.

Case 1: DNBR Case

The transient response calculated for the total loss-of-load event (DNBR case) is shown in Figures 2.8.5.2.1-1 through 2.8.5.2.1-3.

The reactor was tripped via an OTΔT signal. The nuclear power slightly increased until the reactor was tripped and the pressurizer PORVs and sprays minimized the primary pressure transient, which was conservative for DNBR. Although the DNBR value decreases below the initial value, it remains well above the safety analysis limit throughout the entire transient. The peak pressurizer water volume remains below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition. The MSSVs actuated to maintain the secondary side pressure below 110% of the design value.

Case 2: MSS Peak Pressure Case

The transient response calculated for the total loss-of-load event (MSS peak pressure case) is shown in Figures 2.8.5.2.1-4 through 2.8.5.2.1-6.

The reactor was tripped via an OTΔT signal. The nuclear power slightly increased until the reactor was tripped and the pressurizer PORVs and sprays minimized the primary pressure transient, which was conservative to delay reactor trip and exacerbate the peak secondary side pressure. The MSSVs actuated to maintain the secondary side pressure below 110% of the design value. The peak pressurizer water volume remained below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition.

Case 3: RCS Peak Pressure Case

The transient response calculated for the total loss-of-load event (RCS peak pressure case) is shown in Figures 2.8.5.2.1-7 through 2.8.5.2.1-9.

The reactor was tripped on the high-pressurizer pressure reactor trip function. The nuclear power remained essentially constant at full power until the reactor was tripped. The PSVs actuated and confirmed that the primary side pressure was maintained below 110% of the design value. The MSSVs were also actuated and secondary side pressure was maintained below 110% of the design value. The peak pressurizer water volume remained below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition.

Summary

The results of this analysis showed that the plant design is such that a total loss-of-external-electrical load without a direct reactor trip presents no hazard to the integrity of the RCS or the MSS. All of the applicable acceptance criteria were met. The minimum DNBR remained greater than the applicable safety analysis limit value and the peak primary and secondary system pressures remained below 110% of their respective design pressures at all times as shown in Table 2.8.5.2.1-2. The protection features presented in LR section 2.8.5.2.1.2.2 adequately mitigated the loss-of-external-electrical load/turbine trip transient such that the above acceptance criteria were satisfied.

Turbine Trip

The analysis of the consequences of an instantaneous turbine trip by closure of the turbine stop valves was bounded by the analyses performed for the loss of external electrical load event.

Loss-of-Condenser Vacuum

Loss-of-condenser vacuum can occur from failure of the circulating water system or excessive air in-leakage through turbine gland packing. In the event of loss of condenser vacuum, the turbine is tripped and, therefore, the event is bounded by the turbine trip event noted above.

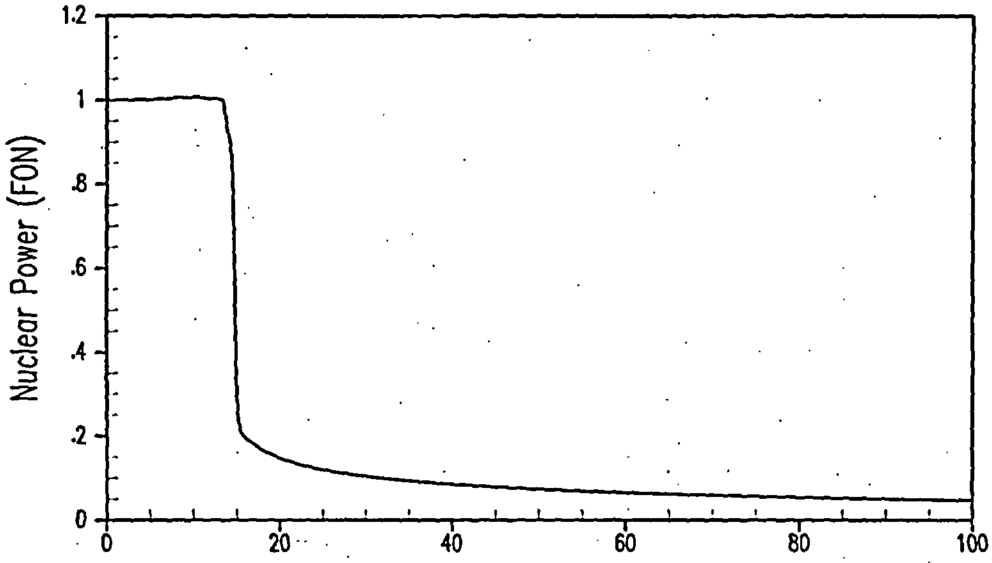
2.8.5.2.1.4 References

1. WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), *Revised Thermal Design Procedure*, April 1989.

2. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, April 1999.

Table 2.8.5.2.1-1 Time Sequence of Events – Loss of External Electrical Load and/or Turbine Trip		
Case	**Event**	**Time (sec)**
DNBR Case (auto pressurizer pressure control, RTDP initial conditions)	Loss of Electrical Load/Turbine Trip	0.0
	Overtemperature ΔT Reactor Trip Setpoint Reached	11.6
	Rods Begin to Drop	13.1
	Minimum DNBR Occurs	14.6
MSS Peak Pressure Case (auto pressurizer pressure control, STDP initial conditions)	Loss of Electrical Load/Turbine Trip	0.0
	Overtemperature ΔT Reactor Trip Setpoint Reached	10.9
	Rods Begin to Drop	12.4
	Peak Secondary Side Pressure Occurs	15.9
RCS Peak Pressure Case (no pressurizer pressure control, STDP initial conditions)	Loss of Electrical Load/Turbine Trip	0.0
	High-Pressurizer Pressure Reactor Trip Setpoint Reached	5.4
	Rods Begin to Drop	7.4
	Peak RCS Pressure Occurs	8.5

Table 2.8.5.2.1-2 Loss of External Electrical Load and/or Turbine Trip – Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Minimum DNBR	1.61	1.82	1.38 (EPU)
Peak Primary System Pressure (psia)	2746.8	2739	2748.5
Peak Secondary System Pressure	1208.0	1191.	1208.5



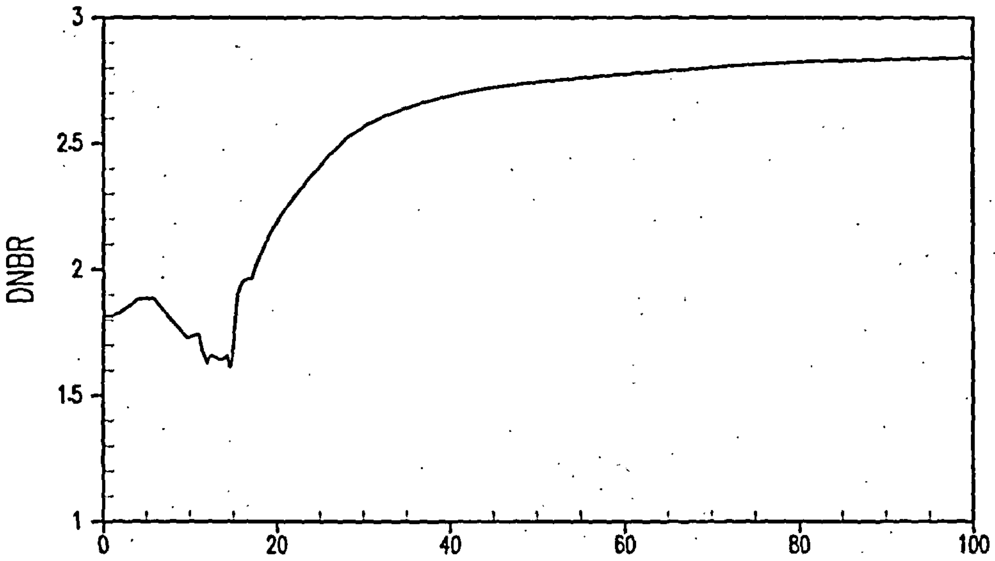
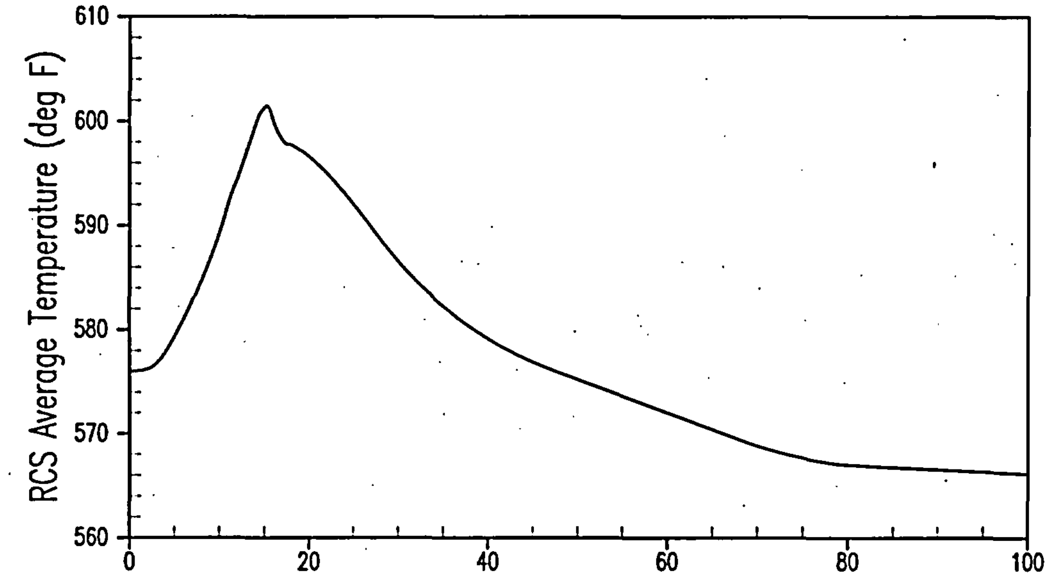


Figure 2.8.5.2.1-1
Loss of Load/Turbine Trip DNBR Case
Nuclear Power and DNBR vs. Time



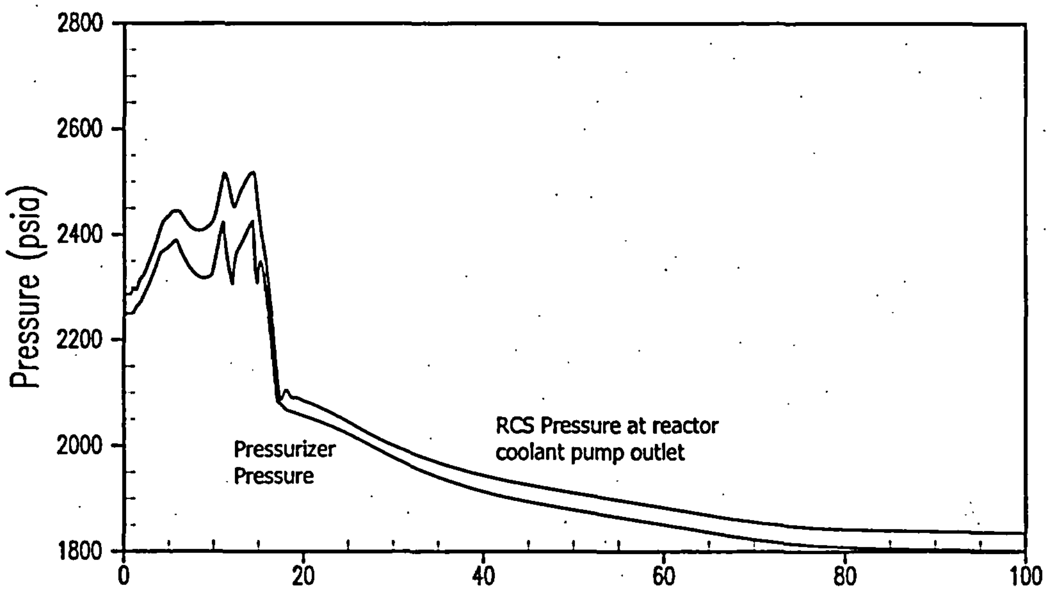


Figure 2.8.5.2.1-2
Loss of Load/Turbine Trip DNBR Case
RCS Average Temperature and Pressurizer/Maximum RCS Pressure vs. Time

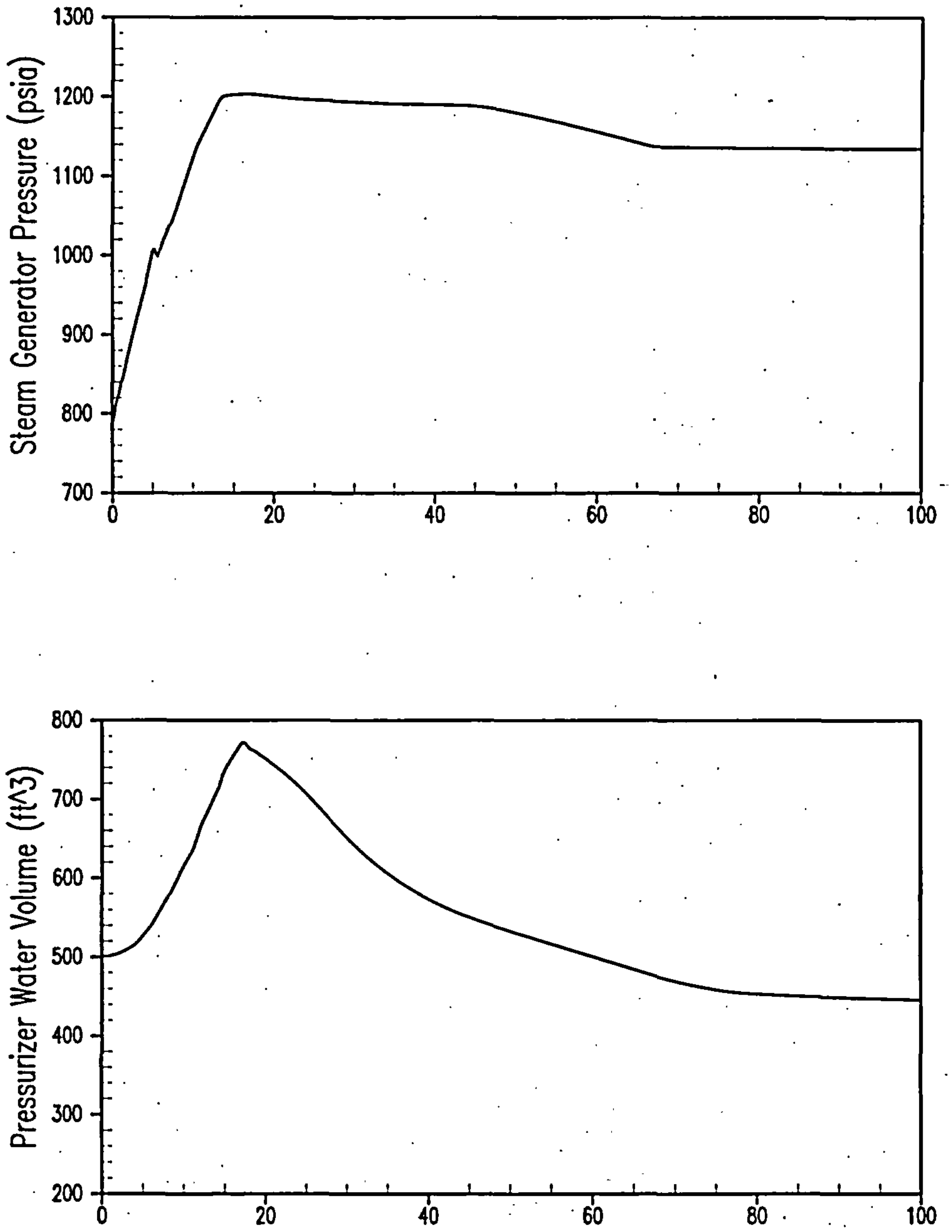


Figure 2.8.5.2.1-3
Loss of Load/Turbine Trip DNBR Case
Steam Generator Pressure and Pressurizer Water Volume vs. Time

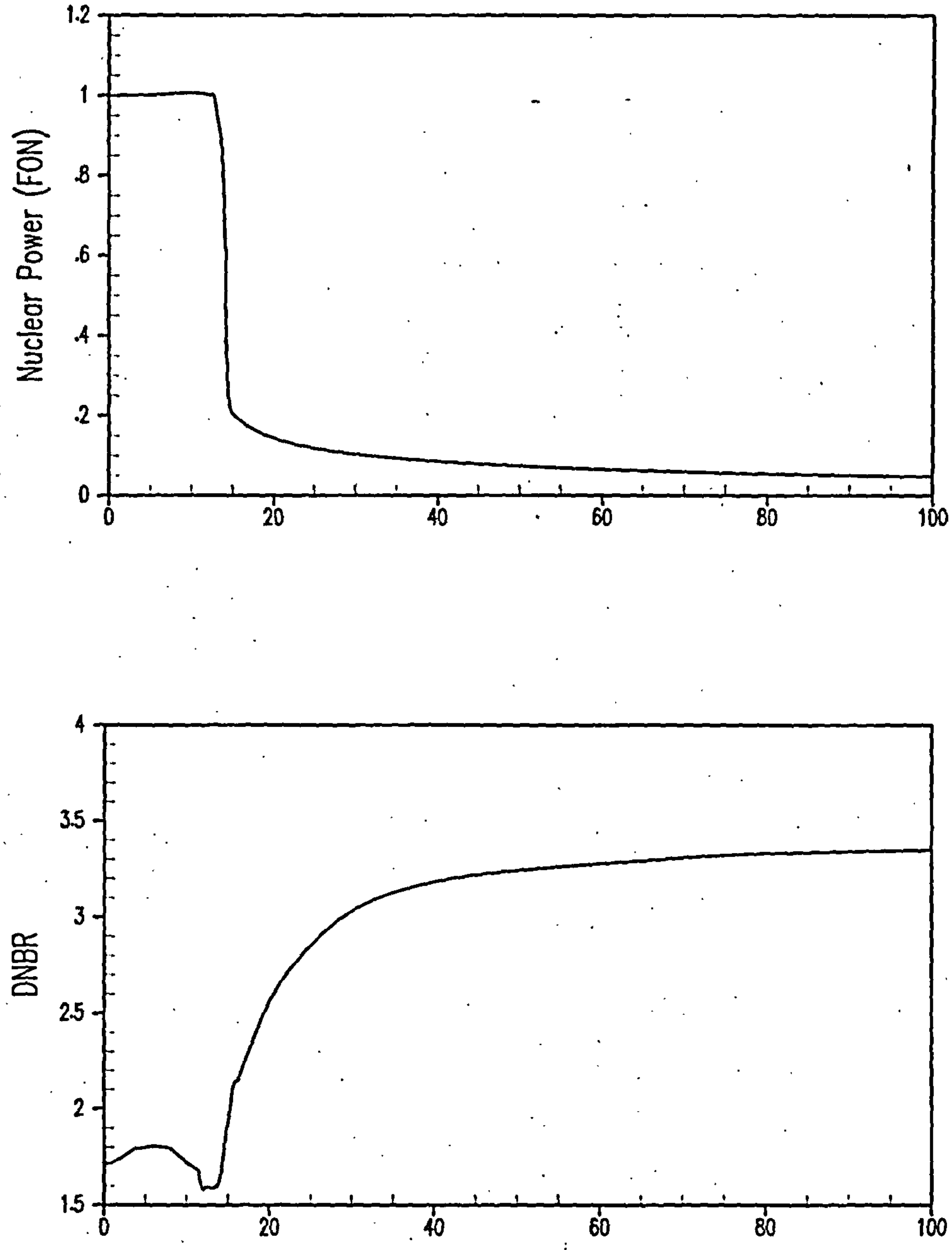
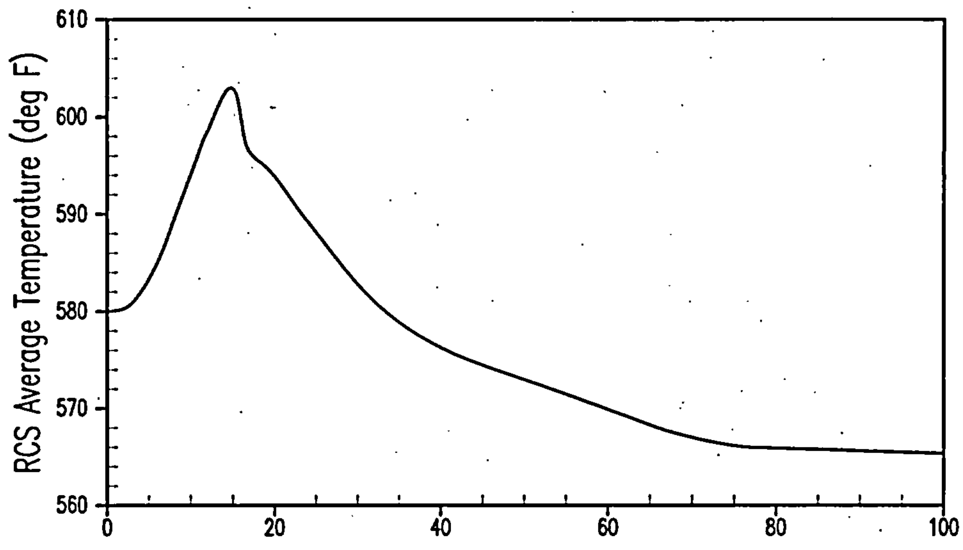


Figure 2.8.5.2.1-4
Loss of Load/Turbine Trip MSS Peak Pressure Case
Nuclear Power and DNBR vs. Time



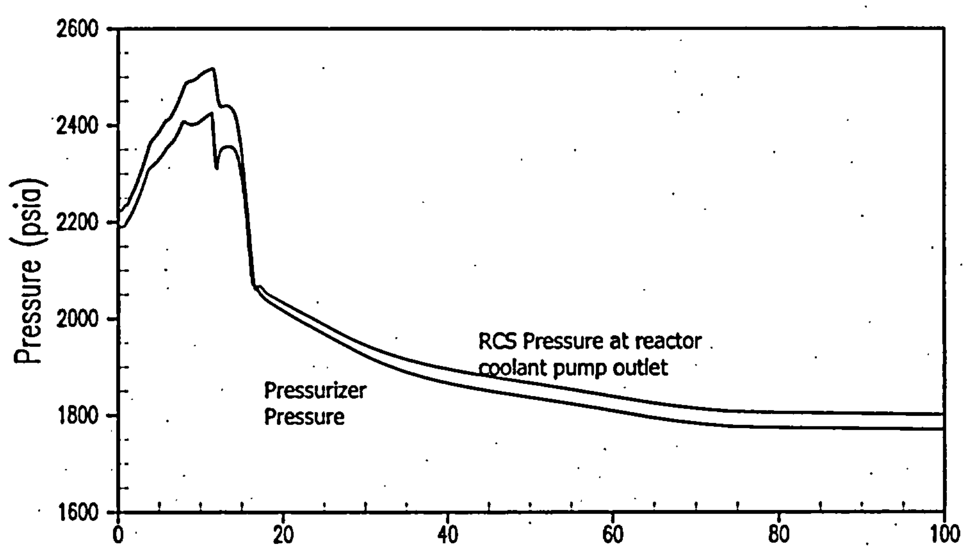


Figure 2.8.5.2.1-5
Loss of Load/Turbine Trip MSS Peak Pressure Case
RCS Average Temperature and Pressurizer/Maximum RCS Pressure vs. Time

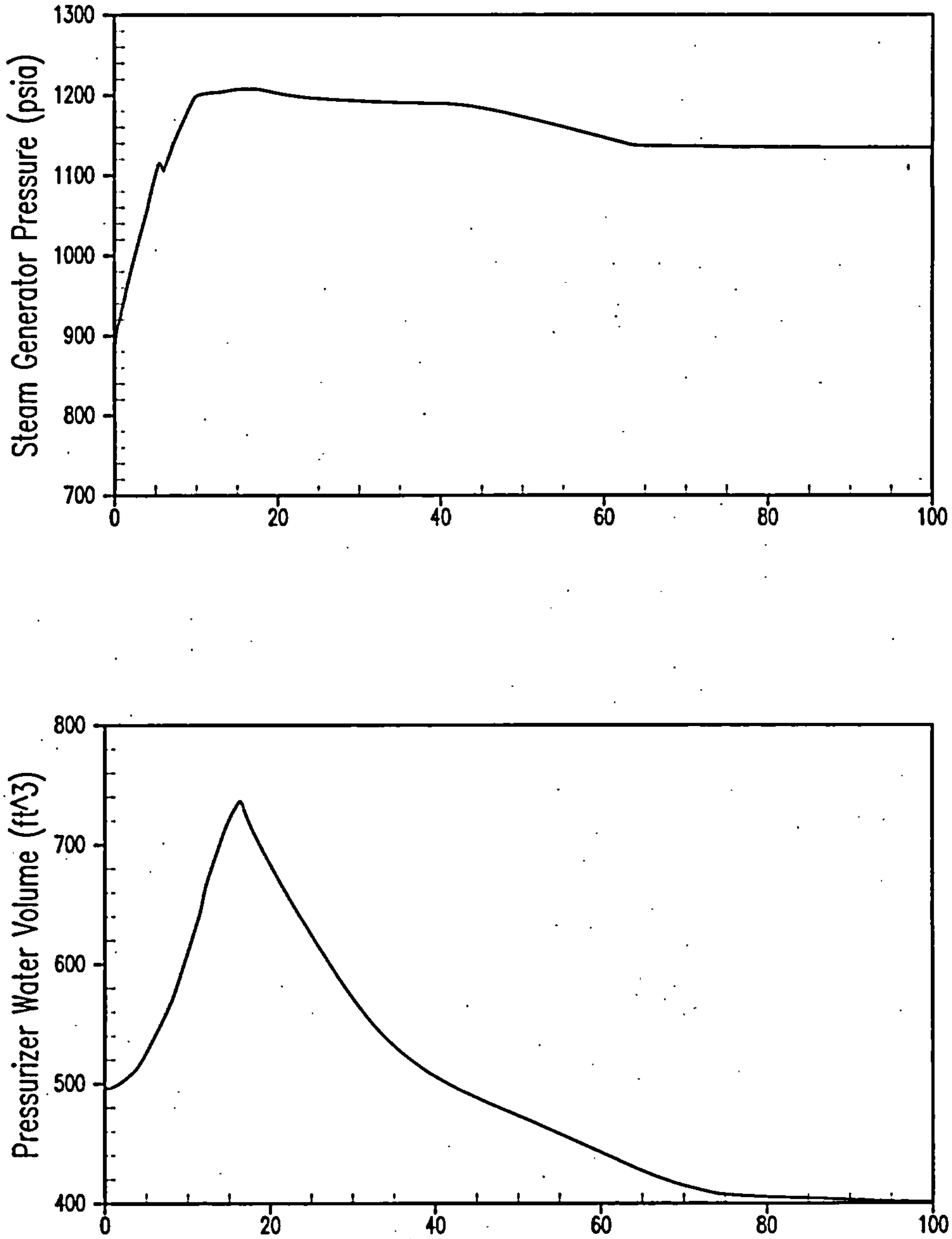


Figure 2.8.5.2.1-6
Loss of Load/Turbine Trip MSS Peak Pressure Case
Steam Generator Pressure and Pressurizer Water Volume vs. Time

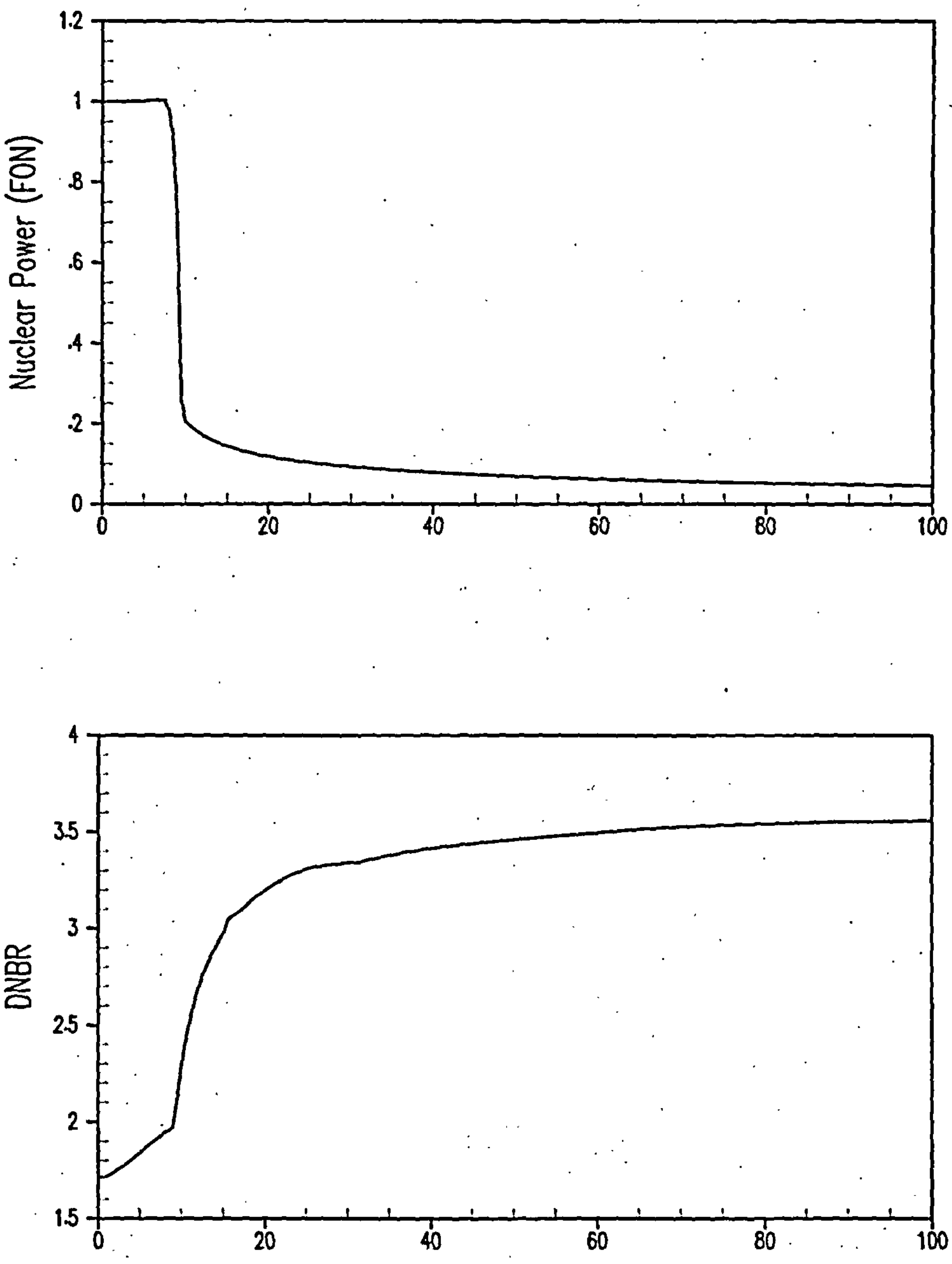
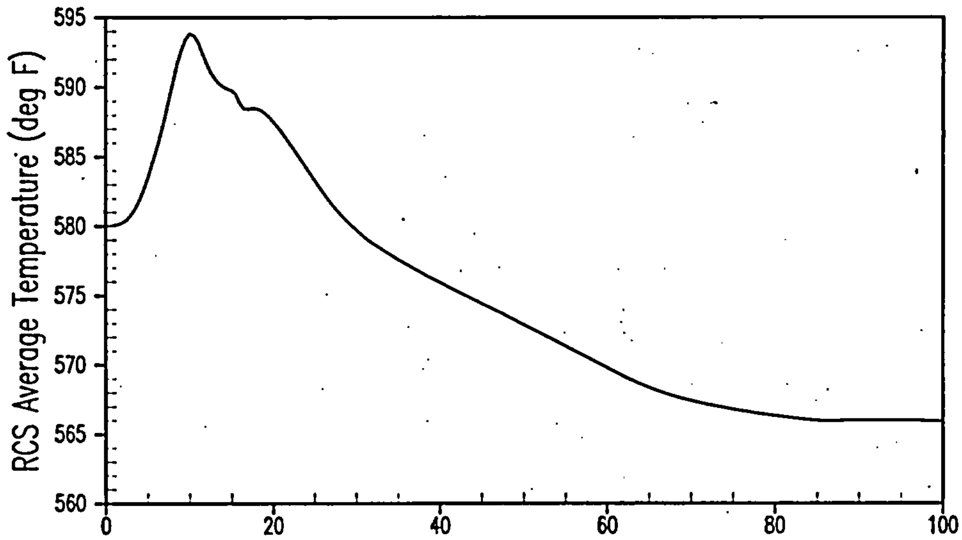


Figure 2.8.5.2.1-7
Loss of Load/Turbine Trip RCS Peak Pressure Case
Nuclear Power and DNBR vs. Time



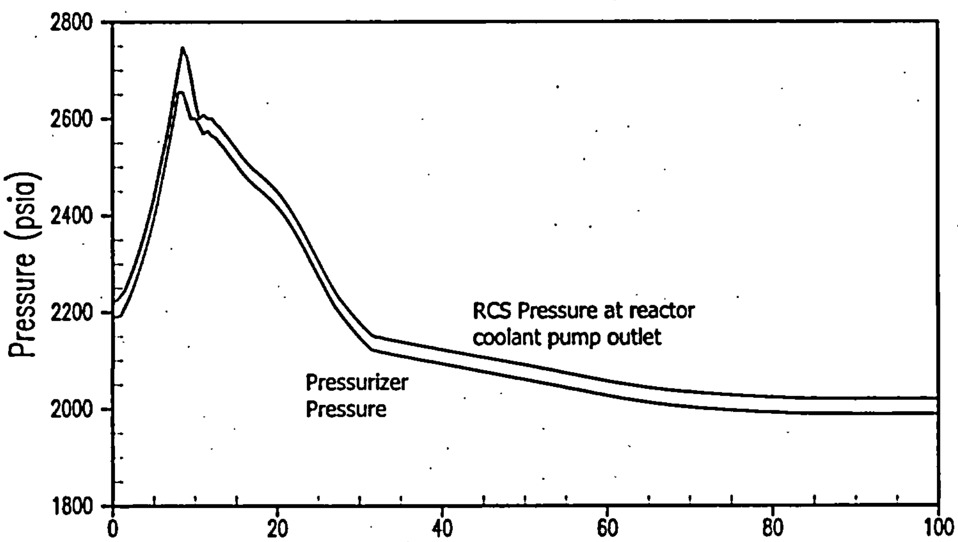


Figure 2.8.5.2.1-8
Loss of Load/Turbine Trip RCS Peak Pressure Case
RCS Average Temperature and Pressurizer/Maximum RCS Pressure vs. Time

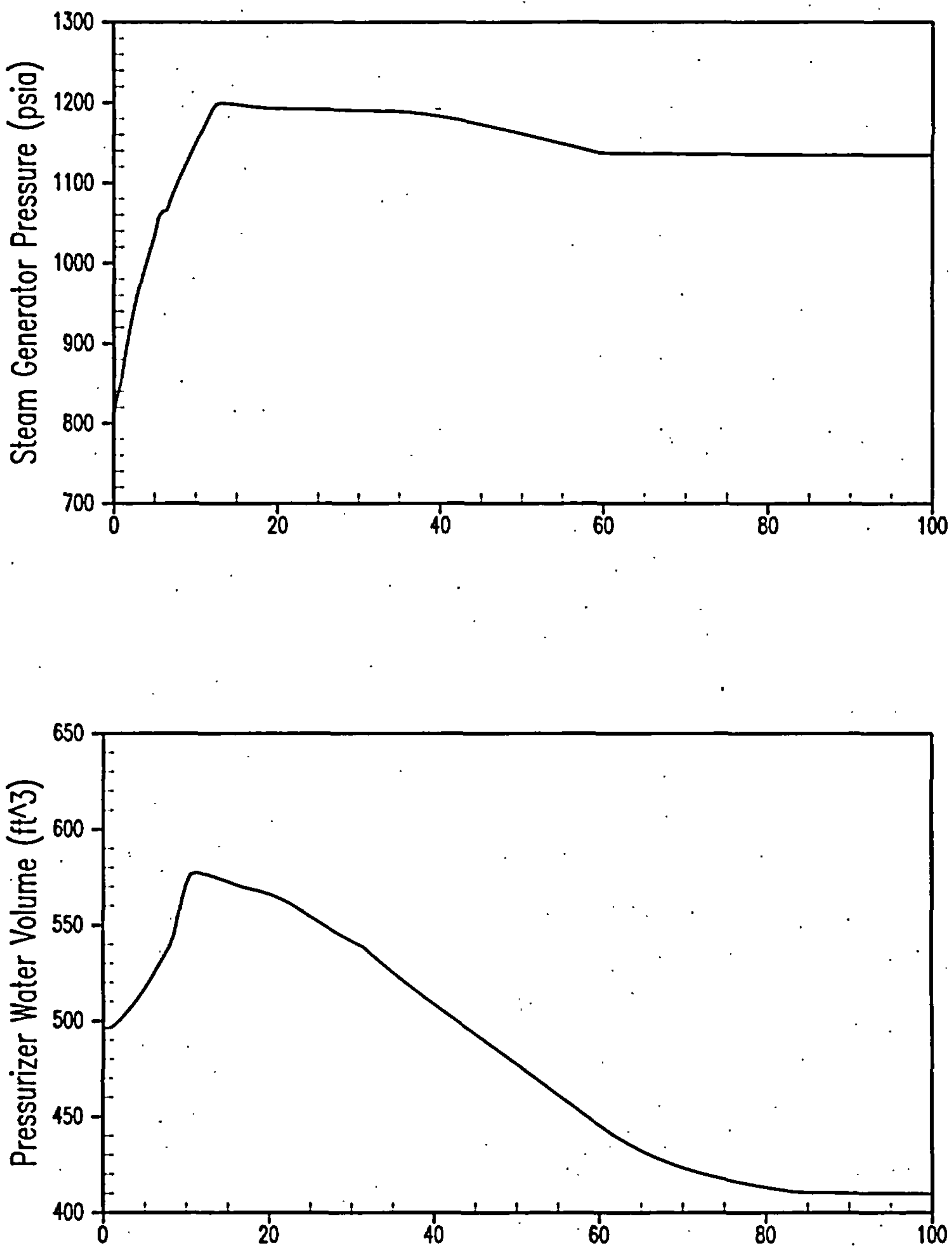


Figure 2.8.5.2.1-9
Loss of Load/Turbine Trip RCS Peak Pressure Case
Steam Generator Pressure and Pressurizer Water Volume vs. Time

2.8.5.2.1.5 Conclusion

Ginna staff has reviewed the analyses of the decrease in heat removal events described above and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of these events. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

2.8.5.2.2.1 Regulatory Evaluation

The loss-of-non-emergency-ac-power event is assumed to result in the loss-of-all power to the station auxiliaries and the simultaneous tripping of all reactor coolant pumps (RCPs) This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytic models, and the results of the transient analyses.

The NRC acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences

- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the reactor coolant pressure boundary (RCPB) is not exceeded during any condition of normal operation

- GDC-26, insofar as it requires that a reactivity control system be provided and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Specific review criteria are contained in SRP, Section 15.2.6, and other guidance provided in Matrix 8 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), 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 are published in NURG-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 primary and secondary systems design relative to conformance to

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 –Reactor Design. As described in this UFSAR section, the reactor core design in combination with coolant, control, and protection systems, provides margins that ensure that fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Further discussion of this design is provided in UFSAR Chapter 4.

- GDC-15 is described in 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. The analysis of the ability of the plant to safely undergo all anticipated transients with peak pressures below 2485 psig is provided in UFSAR Chapter 15. In addition, overpressure protection is prevented by a combination of automatic control and pressure relief devices.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, two means of controlling reactivity are employed at the Ginna Station. Control rod drive mechanisms regulate the position of neutron absorbers within the core. The control rods are designed to shut the reactor down with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The other reactivity control system employs the chemical and volume control system to regulate the concentration of boric acid, a neutron absorber, in the reactor coolant system. Reactivity control system redundancy and capability are further discussed in UFSAR sections 4.3 and 9.3.4.

In addition to the evaluations described in the UFSAR, the analysis of a loss of non-emergency ac power to station auxiliaries event at Ginna Station 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.

2.8.5.2.2.2 Technical Evaluation

2.8.5.2.2.2.1 Introduction

A complete loss of non-emergency ac power (UFSAR 15.2.6) will result in a loss of power to the plant auxiliaries, i.e., the RCPs, main feedwater pumps, condensate pumps, etc. The loss-of-power can be caused by a complete loss-of-the-offsite grid accompanied by a turbine generator trip at the station, or by a loss-of-the-onsite-ac distribution system. The events following a loss-of-ac power with turbine and reactor trip are described in the sequence listed below:

- Plant vital instruments are supplied by emergency dc power sources.
- The atmospheric relief valves (ARVs) can be automatically opened to the atmosphere as the steam system pressure rises following the trip. The condenser is assumed unavailable for steam dump. If the relief capacity of the ARVs is inadequate, the main steam safety valves (MSSVs) can lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The ARVs (or MSSVs, if the ARVs are inadequate or unavailable) are used to dissipate the residual decay heat and to maintain the plant at the MODE 3 (hot shutdown) condition as the no-load temperature is approached.
- The emergency diesel generators start on loss of voltage to the engineered safety features buses and begin to supply safeguards loads in the event offsite power is also lost.

As discussed in UFSAR 15.2.6.3.1, the following provide the necessary protection following a loss of all ac power:

- The reactor can be tripped on one or more of the following reactor trip signals:
  - Pressurizer-high pressure trip signal if any two-of-three pressure channels exceed a fixed setpoint
  - Pressurizer-high water level trip signal if any two-of-three level channels exceed a fixed setpoint

- Overtemperature ΔT trip signal if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressurizer pressure to protect against departure from nucleate boiling (DNB).

- Low-low steam generator water level trip signal if any two-out-of-three level channels in either steam generator are below a fixed setpoint

- Two motor-driven auxiliary feedwater (MDAFW) pumps are started on:

  - Low-low water level in two-out-of-three level channels in any steam generator
  - Trip of both main feedwater pumps (i.e., opening of both main feedwater pump breakers)
  - Safety injection
  - Manual actuation

- One turbine-driven auxiliary feedwater (TDAFW) pump is started on any of the following:

  - Low-low water level in two-out-of-three channels in both steam generators
  - Loss of voltage on both RCPs
  - Manual actuation

- The main steam safety valves (MSSVs) open to provide an additional heat sink and protection against secondary side overpressure.

- The pressurizer safety valves (PSVs) may open to provide protection against overpressure of the RCS.

The auxiliary feedwater (AFW) system is initiated as discussed in the loss-of-normal feedwater analysis (see LR section 2.8.5.2.3, Loss of Normal Feedwater Flow). The reactor trip system and AFW system design provide reactor trip and AFW flow following any loss of normal feedwater.

Following the loss of power to the RCPs, heat removal is maintained by natural circulation in the RCS loops. Following the RCP coastdown, the natural circulation capability of the RCS will remove decay heat from the core, aided by the AFW flow in the secondary system. Demonstrating that acceptable results can be obtained for this event proves that the resultant natural circulation flow in the RCS is adequate to remove decay heat from the core.

The first few seconds after a loss-of-ac-power to the RCPs closely resembles the analysis of the complete loss-of-flow event (see LR section 2.8.5.3.1, "Loss-of-Forced-Reactor-Coolant Flow") in that the RCS experiences a rapid flow reduction transient. This aspect of the loss-of-ac-power event is bounded by the analysis performed for the complete loss-of-flow event that demonstrates that the DNB design basis is met. The analysis of the loss-of-ac-power event demonstrates that RCS natural circulation and the AFW system are capable of removing the stored and residual heat and consequently will prevent RCS or main steam system (MSS) overpressurization and core uncovery. The plant is therefore able to return to a safe condition.

2.8.5.2.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The major assumptions used in this analysis were identical to those used in the loss-of-normal-feedwater analysis described in LR section 2.8.5.2.3, "Loss of Normal Feedwater Flow," with the exception that power is assumed to be lost to the RCPs coincident with rod motion. Details of that assumption are as follows:

- Loss of ac power was assumed to occur soon after the time of reactor trip on low-low steam generator water level. No credit was taken for the immediate insertion of the control rods as a result of the loss of ac power to the station auxiliaries.

- Power was assumed to be lost to the RCPs. To maximize the amount of stored energy in the RCS, the power to the RCPs was not assumed to be lost until after the start of rod motion.

- The plant was initially operating at a NSSS power of 1817 MWt. Since power to the RCPs was lost, a nominal RCP heat of 6.0 MWt and core power of 1811 MWt were assumed. A nominal RCP heat of 6.0 MWt was assumed to be conservative since the RCPs coasted down and ceased to add heat to the primary coolant while the core decay heat was based on a slightly higher initial core power.

- The RCPs were assumed to lose power and coastdown shortly after reactor trip, and the post-trip heat removal from the core relied upon natural circulation flow in the RCS loops.

- The RCS flow coastdown was based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance was combined with the continuity equation, a pump momentum balance, the as-built pump characteristics, and conservative estimates of system pressure losses.

- The most limiting loss-of-non-emergency-ac-power case with respect to overfill for EPU was with a conservative temperature uncertainty subtracted from the high nominal (window) T_{avg} (i.e., 576° - 4°F), conservative pressure uncertainty added to the nominal value (i.e., 2250 psia + 60 psi), while modeling low (390°F) main feedwater temperature conditions.

Based on its frequency of occurrence, the loss-of-non-emergency-ac-power accident was considered a Condition II event as defined by the American Nuclear Society. A restrictive acceptance criterion that the pressurizer does not become water solid has been used for this event. This criterion establishes the acceptable capacity of the AFW system, ensuring that the pressure criteria and minimum DNBR criterion remained satisfied for the long-term portion of the event, and demonstrated that a more serious plant condition is precluded.

2.8.5.2.2.2.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (reference 1) computer code is performed to determine the plant transient following a loss-of-all-ac–power event. The code models the core neutron kinetics, RCS (including natural circulation), pressurizer, pressurizer PORVs and sprays, steam generators, MSSVs, and the AFW system. RETRAN computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature. Additional discussion of the RETRAN code is contained in LR section 2.8.5.0.9, "Computer Codes Utilized."

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

The NRC issued the Ginna License Renewal Safety Evaluation Report (SER), NUREG-1786, in May 2004. The plant systems and system components whose performance is relied upon to support the inputs, assumptions, and results of the analyses described in this section for transients resulting in unplanned sudden decreases in heat removal by the secondary system are not being modified for the proposed EPU. Those systems and system components are described in SER sections 2.3, "Mechanical Systems," and 2.5, "Electrical, Instrumentation and Control Systems," for Scoping and Screening Results and SER sections 3.1, "Reactor Coolant System," 3.2, "Engineered Safety Features," 3.4, "Steam & Power Conversion Systems, and 3.6, "Electrical, Instrumentation and Control Systems" for Aging Management Review. EPU activities do not add any new components nor do they introduce any new functions for existing components associated with these analyses that would change the license renewal evaluation boundaries. The primary and secondary systems performance capability described in this LR section for the proposed EPU involves analytical techniques and methodology are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of SRP 15.2.1-5 with respect to GDC- 10, 15, and 26. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to plant components associated with the analyses described in this LR section.

2.8.5.2.2.3 Results

Figures 2.8.5.2.2-1 through 2.8.5.2.2-9 present transient plots of plant parameters following a loss-of-non-emergency-ac-power event with the assumptions listed in LR section 2.8.5.2.2.2.2. The calculated sequence of events for this accident is listed in Table 2.8.5.2.2-1. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.2.2-2. The most limiting case is initiated with the average RCS temperature at the low end of the temperature window. Margin was generated for the EPU analyses by defining a new pressurizer level program for operation at the low end of the temperature window. For a full power Tavg of 576.0°F, the nominal pressurizer level is 60% narrow range setpoint (NRS). For a full power Tavg of 564.6°F, the nominal pressurizer level is 44.3% NRS. The previous analyses assume 54% NRS for all cases.

The first few seconds after the loss of non-emergency-ac-power to the RCPs, the flow transient closely resembles the complete loss-of-flow incident, where core damage due to rapidly increasing core temperatures is prevented by the reactor trip. For a loss-of-non-emergency-ac-power event, the DNBR results were less limiting since the reactor was already tripped when RCP coastdown began. After the reactor trip, stored and residual heat had to be removed to prevent damage to the core and the RCP and MSS. The RETRAN code results show that the natural circulation and AFW flow available are sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

Figure 2.8.5.2.2-6 illustrates that the pressurizer does not reach a water solid condition, hence, no water relief from the pressurizer occurs.

With respect to DNB, the loss-of-non-emergency-ac-power event was bounded by the complete loss-of-flow event described in LR section 2.8.5.3.1, "Loss-of Forced-Reactor-Coolant Flow," which demonstrates that the minimum DNBR is greater than the safety analysis limit value of 1.38. Also, with respect to primary and secondary overpressurization, the loss of load event described in LR section 2.8.5.2.1, "Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum," demonstrates that the primary and secondary pressure limits of 2748.5 psia and 1208.5 psia are met.

The results of the analysis show that the pressurizer does not reach a water solid condition (800 ft^3). Therefore, the loss-of-offsite-power event does not adversely affect the core, the RCS, or the MSS.

2.8.5.2.2.4 Reference

1. WCAP-15234-A (Nonproprietary), May 1999 and WCAP-14882-P-A (Proprietary), April 1999, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, D.S., et al.

Table 2.8.5.2.2-1 Time Sequence of Events – Loss of Non-Emergency AC Power to the Plant Auxiliaries	
Event	**Time (seconds)**
Main Feedwater Flow Stops	20.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	60.8
Rods Begin to Drop	62.8
RCPs Begin to Coastdown	64.8
Flow from two MDAFW Pumps is Initiated	120.8
Long-Term Peak Water Level in Pressurizer Occurs	266.0
Core Decay Heat Decreases to AFW Heat Removal Capacity	~700

Table 2.8.5.2.2-2 Loss of Non-Emergency AC Power to the Plant Auxiliaries Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Peak Pressurizer Water Volume from the limiting case (ft^3)	635	592	800

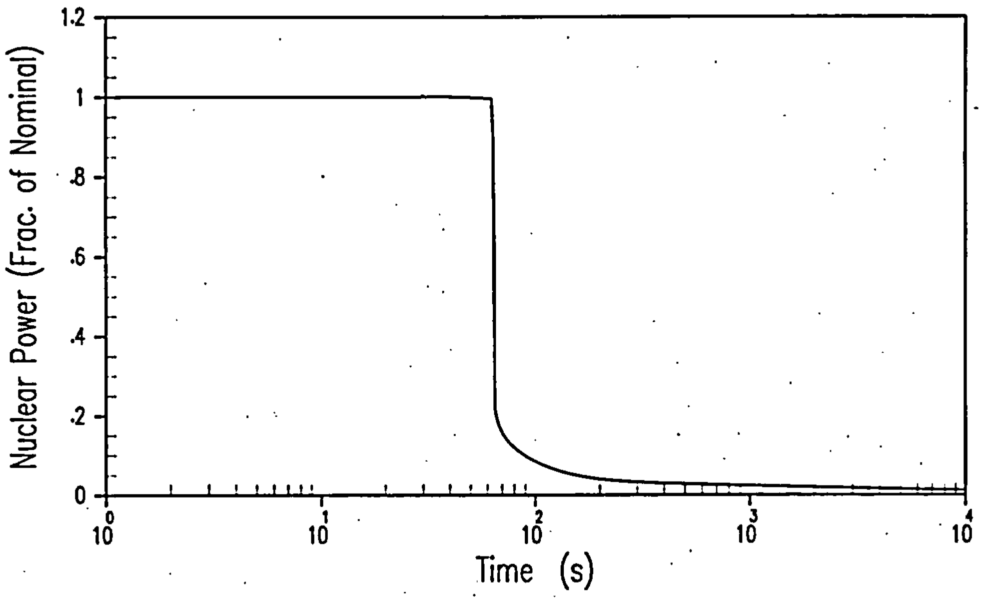


Figure 2.8.5.2.2-1
Loss of All Offsite AC Power to the Plant Auxiliaries Nuclear Power vs. Time

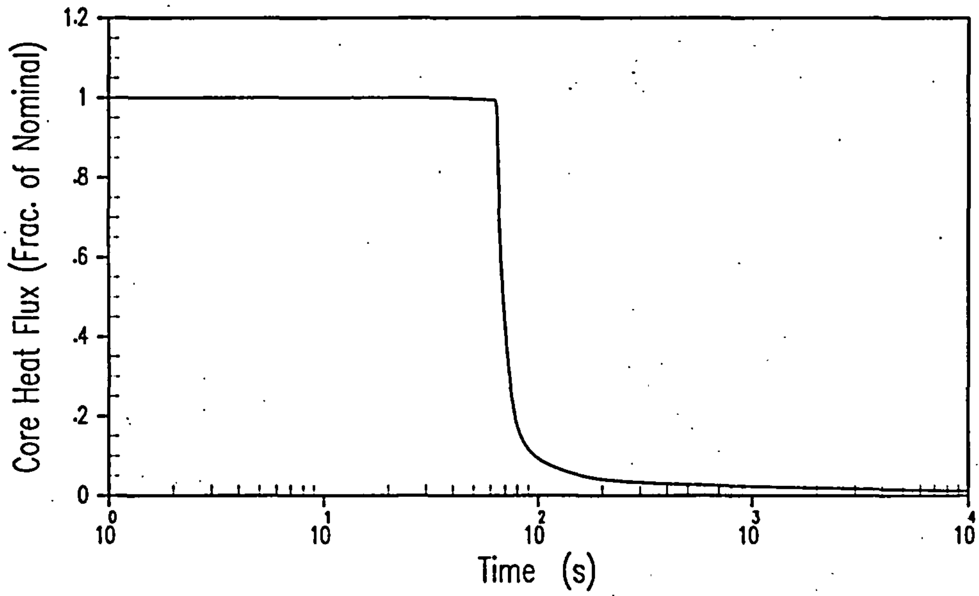


Figure 2.8.5.2.2-2
Loss of All Offsite AC Power to the Plant AuxiliariesCore Average Heat Flux vs. Time

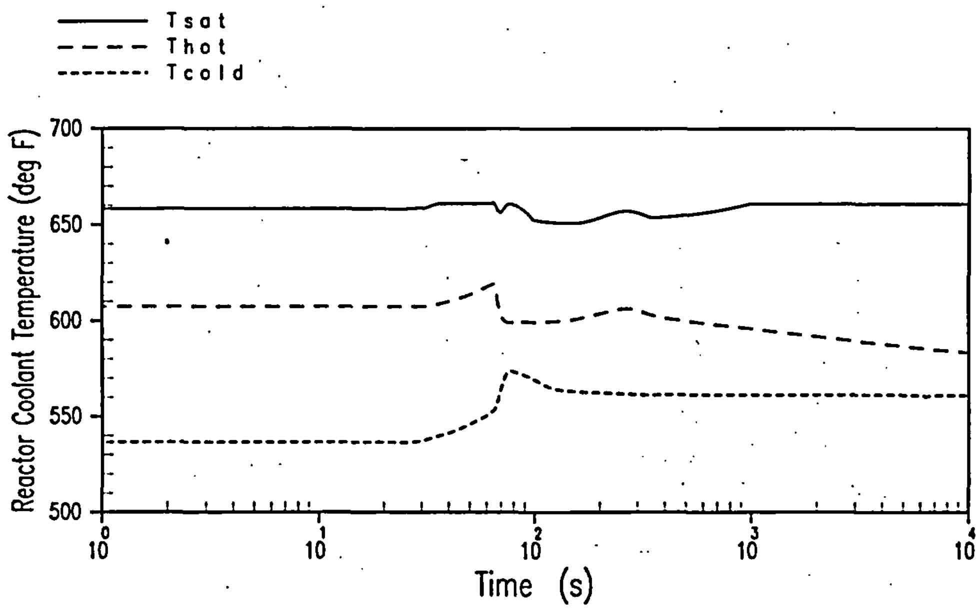


Figure 2.8.5.2.2-3
Loss of All Offsite AC Power to the Plant Auxiliaries
RCS Temperatures vs. Time

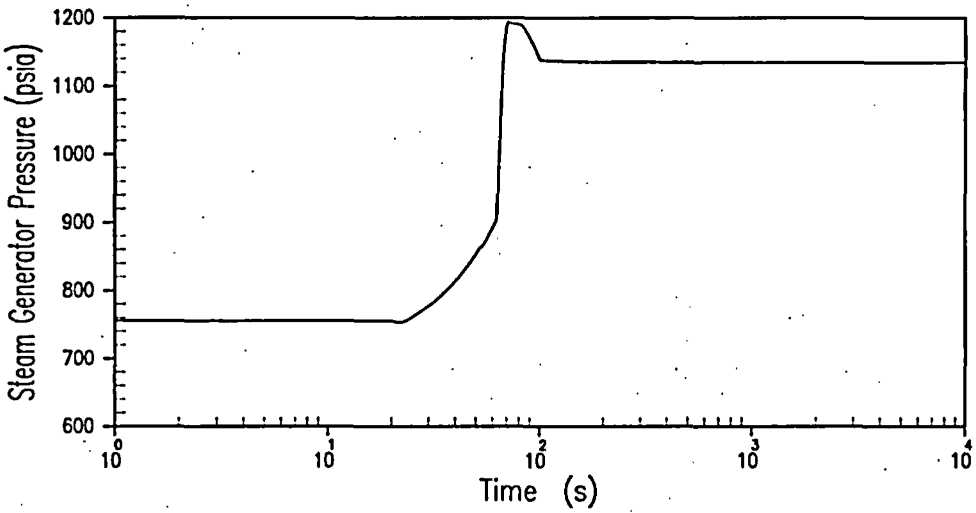


Figure 2.8.5.2.2-4
Loss of All Offsite AC Power to the Plant Auxiliaries
Steam Generator Pressure vs. Time

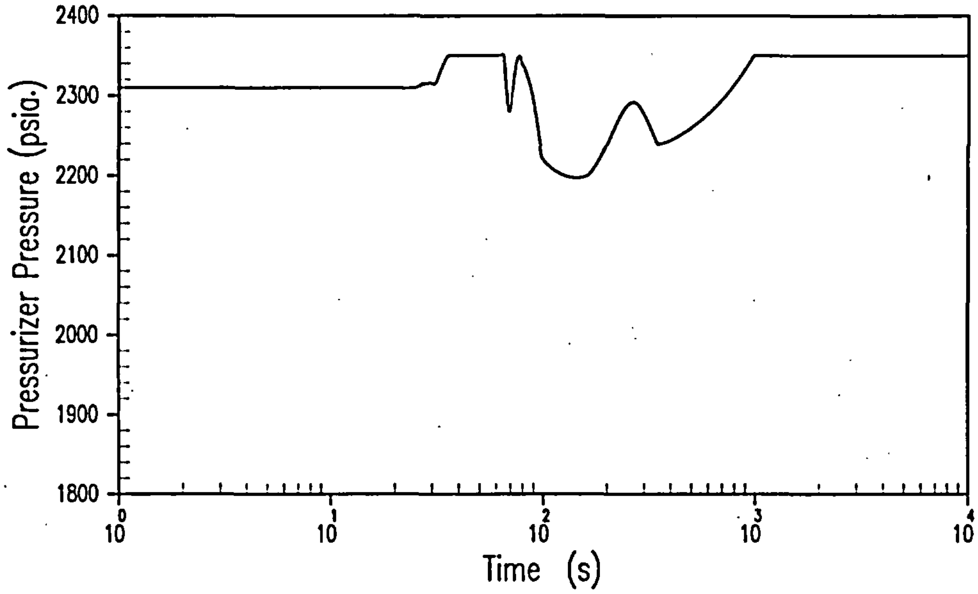


Figure 2.8.5.2.2-5
Loss of All Offsite AC Power to the Plant Auxiliaries
Pressurizer Pressure vs. Time

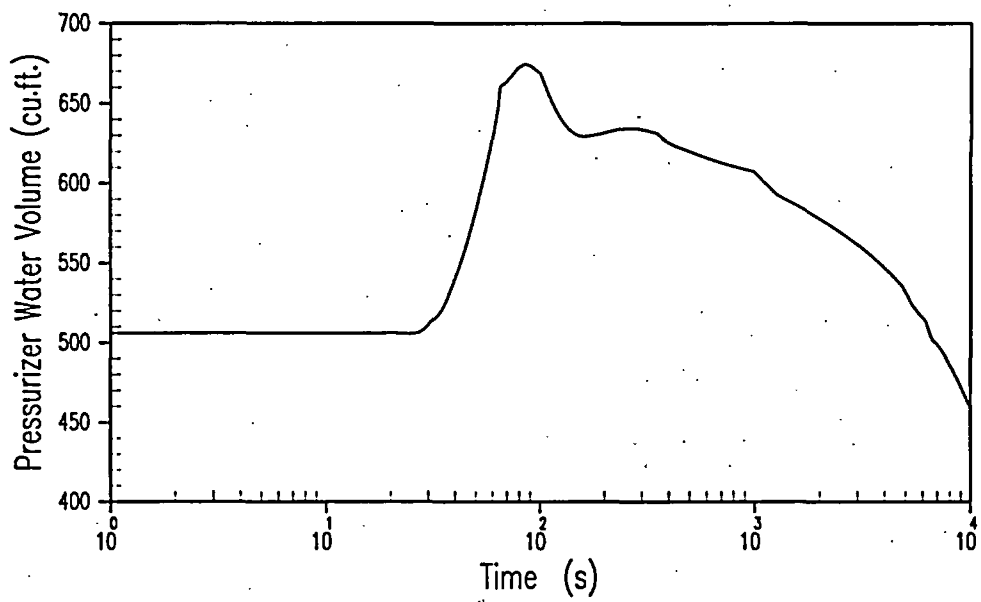


Figure 2.8.5.2.2-6
Loss of All Offsite AC Power to the Plant Auxiliaries
Pressurizer Water Volume vs. Time

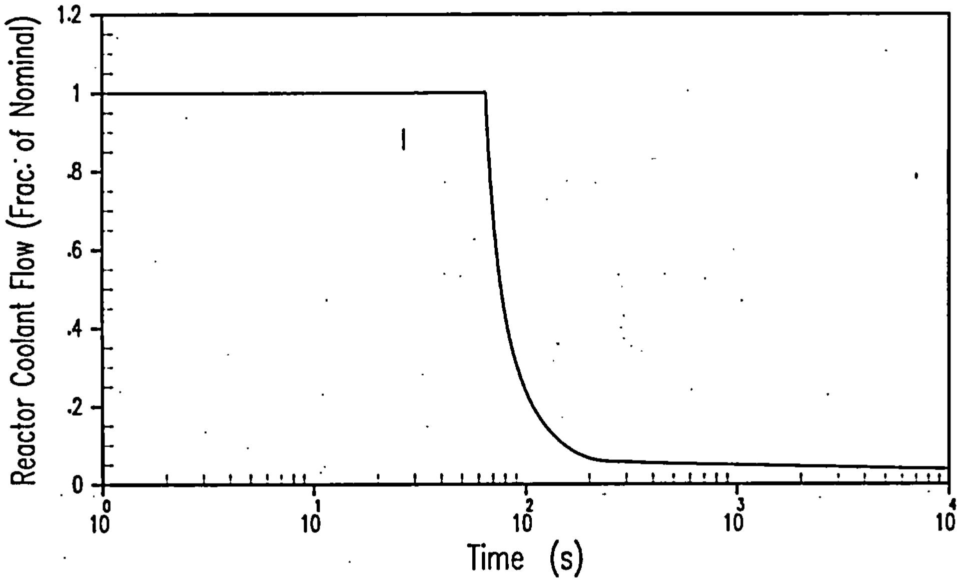


Figure 2.8.5.2.2-7
Loss of All Offsite AC Power to the Plant Auxiliaries
Reactor Coolant Flow Rate vs. Time

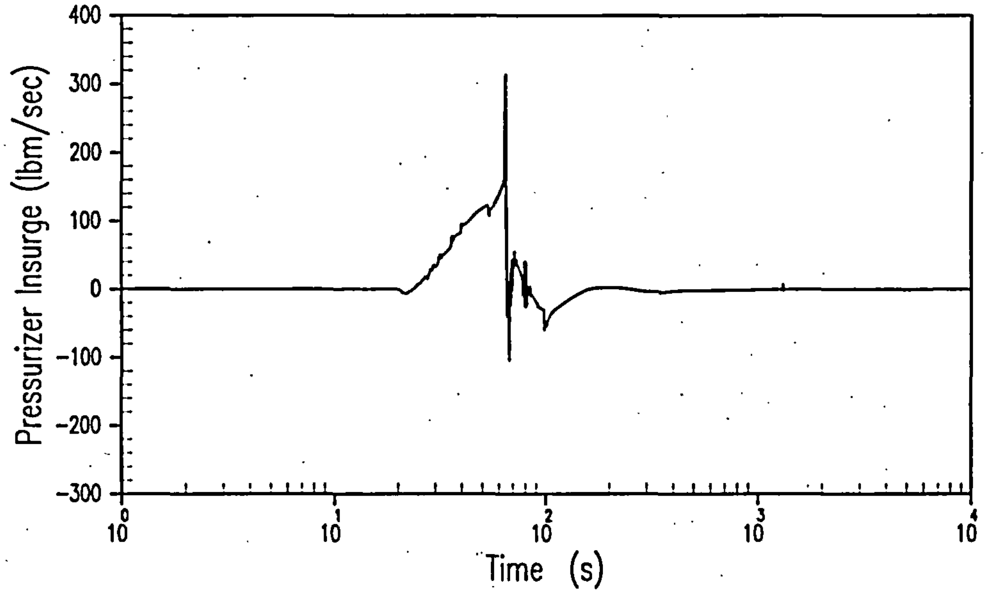


Figure 2.8.5.2.2-8
Loss of All Offsite AC Power to the Plant Auxiliaries
Pressurizer Insurge vs. Time

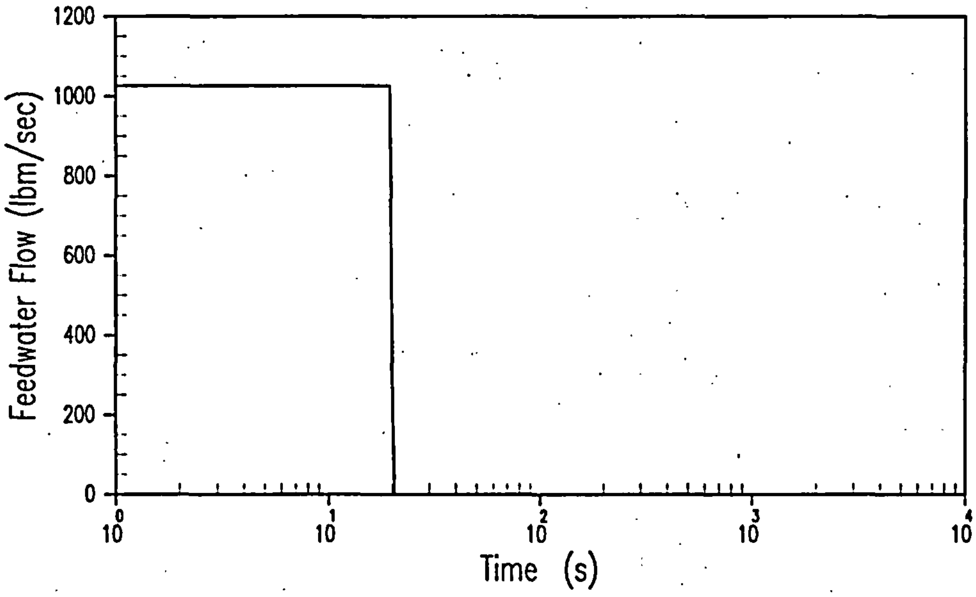


Figure 2.8.5.2.2-9
Loss of All Offsite AC Power to the Plant Auxiliaries
Feedline Flow vs. Time

2.8.5.2.2.5 Conclusion

The Ginna staff has reviewed the analyses of the loss of non-emergency ac power to station auxiliaries event and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the loss of non-emergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

2.8.5.2.3.1 Regulatory Evaluation

A loss of normal feedwater flow (LONF) could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). Loss of feedwater flow results in an increase in reactor coolant temperature and pressure that eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a LONF. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The Ginna Nuclear Power Plant, LLC (Ginna) staff review covered

- the sequence of events,
- the analytical models used for the analyses,
- the values of parameters used in the analytical models, and
- the results of the transient analyses.

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems are designed with margin sufficient to ensure that the design condition of the reactor coolant pressure boundary (RCPB) is not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system is provided and is capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded.

Specific review criteria are contained in the SRP, Section 15.2.7, and other guidance provided in Matrix 8 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), 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 are published in NURG-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 primary and secondary systems design relative to conformance to

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 –Reactor Design. As described in this UFSAR section, the reactor core design in combination with coolant, control, and protection systems, provides margins that ensure that fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Further discussion of fuel system design is provided in UFSAR section 4.2.1.

- GDC-15 is described in 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. The analysis of the ability of the plant to safely undergo all anticipated transients with peak pressures below 2485 psig is provided in UFSAR Chapter 15. In addition, overpressure protection is prevented by a combination of automatic control and pressure relief devices. Analysis of the loss of normal feedwater flow event is described in UFSAR section 15.2.6. In addition, the loss of normal feedwater event was evaluated during the SEP review (SEP Topic XV-5) and found to be acceptable.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, two independent means of controlling reactivity of different design principles are employed at the Ginna Station. Control rod drive mechanisms regulate the position of neutron absorbers within the core. The control rods are designed to shut the reactor down with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The other reactivity control system employs the chemical and volume control system (CVCS) to regulate the concentration of boric acid, a neutron absorber, in the reactor coolant system (RCS). Reactivity control system redundancy and capability are further discussed in UFSAR sections 4.3 and 9.3.4.

In addition to the evaluations described in the UFSAR, the Ginna loss of normal feedwater flow analysis 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 loss of normal feedwater flow analysis is not within the scope of license renewal.

2.8.5.2.3.2 Technical Evaluation

2.8.5.2.3.2.1 Introduction

As described in UFSAR Section 15.2.6, a LONF (from pipe breaks, pump failures, valve malfunctions, or a complete loss of all ac power to station auxiliaries) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater is not supplied, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer could occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables do not approach a condition that causes a departure from nucleate boiling ration (DNBR) limit violation.

The LONF that occurs as a result of the loss of ac power is discussed in LR section 2.8.5.2.2, "Loss of Non-Emergency AC Power to the Station Auxiliaries."

The following events occur following the reactor trip for the LONF as a result of main feedwater pump failures or valve malfunctions:

- The ARVs are automatically opened to the atmosphere as the MSS pressure rises following a loss of feedwater. The condenser is assumed unavailable for steam dump. If the relief capacity of the ARVs is inadequate, the MSSVs can lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

- As the no-load temperature is approached, the ARVs (or MSSVs, if the ARVs are unavailable) are used to dissipate the residual decay heat and to maintain the plant at the MODE 3 (hot shutdown) condition.

The following provide the necessary protection in the event of a LONF:

- The reactor can be tripped on one or more of the following reactor trip signals:

  – Pressurizer-high pressure trip signal if any two-of-three pressure channels exceed a fixed setpoint

  – Pressurizer high water level trip signal if any two-of-three level channels exceed a fixed setpoint

  – Overtemperature ΔT trip signal if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressurizer pressure to protect against DNB

  – Low-low steam generator water level trip signal if any two-out-of-three level channels in either steam generator are below a fixed setpoint.

- Two motor-driven auxiliary feedwater (MDAFW) pumps are started on:

  – Low-low water level in two-out-of-three level channels in any steam generator
  – Trip of both main feedwater pumps (i.e., opening of both main feedwater pump breakers)
  – Safety injection
  – Manual actuation

- One turbine-driven auxiliary feedwater (TDAFW) pump is started on any of the following:

  – Low-low water level in two-out-of-three channels in both steam generators
  – Loss of voltage on both RCPs
  – Manual actuation

- The MSSVs open to provide an additional heat sink and protection against secondary side overpressure.

- The pressurizer safety valves (PSVs) may open to provide protection against overpressure of the RCS.

The analysis showed that following a LONF, the AFW system is capable of removing the stored and residual heat, thus preventing overpressurization of the RCS, overpressurization of the secondary side, water relief from the pressurizer, and uncovery of the reactor core.

2.8.5.2.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The following assumptions were made in the analysis:

- The plant is initially operating at a NSSS power of 1817 MWt. A conservative maximum RCP heat of 10 MWt was included in the analysis. The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the thermal design flow value. Although not assumed in the analysis, the RCPs could be manually tripped at some later time in the transient to reduce the heat addition to the RCS caused by the operation of the pumps.

- Main feedwater temperature conditions at 390°F and 435°F were analyzed.

- The full power vessel average temperature (T_{avg}) window of 564.6°F to 576.0°F is considered, with a temperature uncertainty of +/- 4°F applied. Since the nominal pressurizer level follows a program that varies as nominal T_{avg} changes, cases applying both plus and minus uncertainties at both ends of the operating window are examined to confirm the most limiting initial condition is considered. The most limiting LONF case with respect to pressurizer filling was with the temperature uncertainty subtracted from the low nominal (window) T_{avg} value (i.e., 564.6°F - 4°F), pressure uncertainty added to the nominal value (i.e., 2250 psia + 60 psi), while modeling high (435°F) main feedwater temperature conditions. Note that there are two peaks in the pressurizer water level for a loss of normal feedwater event. The first peak is a function of the initial conditions and the second peak is an indication of the capability of the AFW system to perform long term heat removal. Thus, the magnitude of the second peak is used to determine the limiting case.

- Reactor trip occurs on steam generator low-low water level at 0% of the narrow range span.

- It was assumed that two MDAFW pumps are available to supply a minimum flow of 340 gpm split equally to both steam generators, 60 seconds following a low-low steam generator water level signal. (The worst single failure, which was modeled in the analysis, was the loss of the TDAFW pump.) The two AFW line purge volumes were conservatively assumed to be 96 ft^3 and 213 ft^3, and the initial AFW enthalpy was assumed to be 74.87 Btu/lbm (corresponding with the maximum AFW temperature and pressure conditions).

- The pressurizer sprays and PORVs were assumed operable. This maximized the pressurizer water volume. If these control systems did not operate, the PSVs would prevent the RCS pressure from exceeding the RCS design pressure limit during this transient. The pressurizer heaters were modeled to exacerbate the heatup and volumetric expansion of the water in the pressurizer.

- Secondary system steam relief is achieved through the self-actuated MSSVs. Note that steam relief is provided by the steam generator ARVs or condenser dump valves for most cases of LONF. However, the condenser dump valves and the ARVs were assumed to be unavailable.

- The MSSVs were modeled assuming a 1.5% tolerance and an accumulation model that assumes that the valves were wide open once the pressure exceeded the setpoint (plus tolerance) by 5 psi (accumulation).

- Core residual heat generation was based on the 1979 version of ANS 5.1 (reference 1). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip was assumed.

- Steam generator tube plugging (SGTP) levels of both 0 and 10% were analyzed.

The specific acceptance criteria applied by Ginna for this event are as follows:

- The departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met meets the Ginna Station current licensing basis requirements with respect to GDC-10.

- Primary and secondary pressures remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-15.

- GDC-26 requires reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ensuring that appropriate margin for malfunctions, such as stuck rods, are accounted for in the safety analysis assumptions. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-26.

The discussion below demonstrates that all applicable acceptance criteria are met for this event at Ginna Station at EPU conditions.

Based on its frequency of occurrence, the LONF accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event:

- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.

- Pressure in the RCS and MSSs is maintained below 110% of the design pressures.

- An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

With respect to DNB and overpressurization, the LONF accident was bounded by the loss of load accident reported in LR section 2.8.5.2.1, "Loss of External Electrical Load,

A restrictive acceptance criterion that the pressurizer does not become water solid was used for this event. This criterion established the acceptable capacity of the AFW system, ensuring that the pressure criteria and minimum DNBR criterion remained satisfied for the long-term portion of the event, and demonstrated that a more serious plant condition was precluded.

2.8.5.2.3.2.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (reference 2) computer code was performed to determine the plant transient conditions following a LONF. The code modeled the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, steam generators, MSSVs, and the AFW system. RETRAN computes pertinent variables, including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

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

The NRC issued the Ginna License Renewal Safety Evaluation Report (SER), NUREG-1786, in May 2004. The plant systems and system components whose performance is relied upon to support the inputs, assumptions, and results of the analyses described in this section for transients resulting in unplanned sudden decreases in heat removal by the secondary system are not being modified for the proposed EPU. Those systems and system components are described in SER sections 2.3, "Mechanical Systems," and 2.5, "Electrical, Instrumentation and Control Systems," for Scoping and Screening Results and SER sections 3.1, "Reactor Coolant System," 3.2, "Engineered Safety Features," 3.4, "Steam & Power Conversion Systems, and 3.6, "Electrical, Instrumentation and Control Systems" for Aging Management Review. EPU activities do not add any new components nor do they introduce any new functions for existing components associated with these analyses that would change the license renewal evaluation boundaries. The primary and secondary systems performance capability described in this LR section for the proposed EPU involves analytical techniques and methodology are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of SRP 15.2.1-5 with respect to GDC-10, GDC-15, and GDC-26. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to plant components associated with the analyses described in this LR section.

2.8.5.2.3.3 Results

The calculated sequence of events for this accident is listed in Table 2.8.5.2.3-1. Figures 2.8.5.2.3-1 through 2.8.5.2.3-5 present transient plots of the significant plant parameters following a LONF, with the assumptions listed in LR section 2.8.5.2.3.2.2. The analysis demonstrates that 340 gpm of auxiliary feedwater split equally between the two steam generators is adequate to remove decay heat and pump heat such that no pressurizer filling will occur. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.2.3-2. The most limiting case is initiated with the average RCS temperature at the low end of the temperature window. Margin was generated for the EPU analyses by defining a new pressurizer level program for operation at the low end of the temperature window. For a full power T_{avg} of 576.0°F, the nominal pressurizer level is 60% NRS. For a full power T_{avg} of 564.6°F, the nominal pressurizer level is 44.3% NRS. The previous analyses assume 54% NRS for all cases.

Following the reactor and turbine trip from full load, the water level in the steam generators fell due to reduction of the steam generator void fraction and because steam flow through the safety valves continued to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the MDAFW pumps automatically started, consequently reducing the rate at which the steam generator water level was decreasing.

The capacity of the MDAFW pumps enabled sufficient heat transfer from each steam generator to dissipate the core residual heat without the pressurizer reaching a water solid condition (as shown in Figure 2.8.5.2.3-3). This precluded any water relief through the RCS pressurizer relief valves or PSVs.

With respect to DNB, the LONF accident was bounded by the loss of load event (see LR section 2.8.5.2.1, "Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum"), demonstrating that the minimum DNBR was greater than the safety analysis limit value of 1.38. Also, with respect to primary and secondary overpressurization, the loss of load event described in LR section 2.8.5.2.1, "Loss of External Electrical Load, Turbine, and Loss of Condenser Vacuum," demonstrates that the primary and secondary pressure limits of 2748.5 psia and 1208.5 psia are met.

2.8.5.2.3.4 References

1. ANSI/ANS-5.1 – 1979, *American National Standard for Decay Heat Power in Light Water Reactors*, August 1979.

2. WCAP-14882-P-A (Proprietary), April 1999 and WCAP-15234-A (Nonproprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*), Huegel, D.S., et al., May 1999.

Table 2.8.5.2.3-1 Time Sequence of Events – LONF	
Event	**Time (sec)**
Main Feedwater Flow Stops	20.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	54.6
Rods Begin to Drop	56.6
Flow from Two MDAFW is Initiated	114.6
Long-Term Peak Water Level in Pressurizer Occurs	896.0
Core decay and RCP Heat Decreases to AFW Heat Removal Capacity	~1000

Table 2.8.5.2.3-2 Loss of Normal Feedwater - Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Peak Pressurizer Water Volume from the limiting case (ft^3)	537	592	800

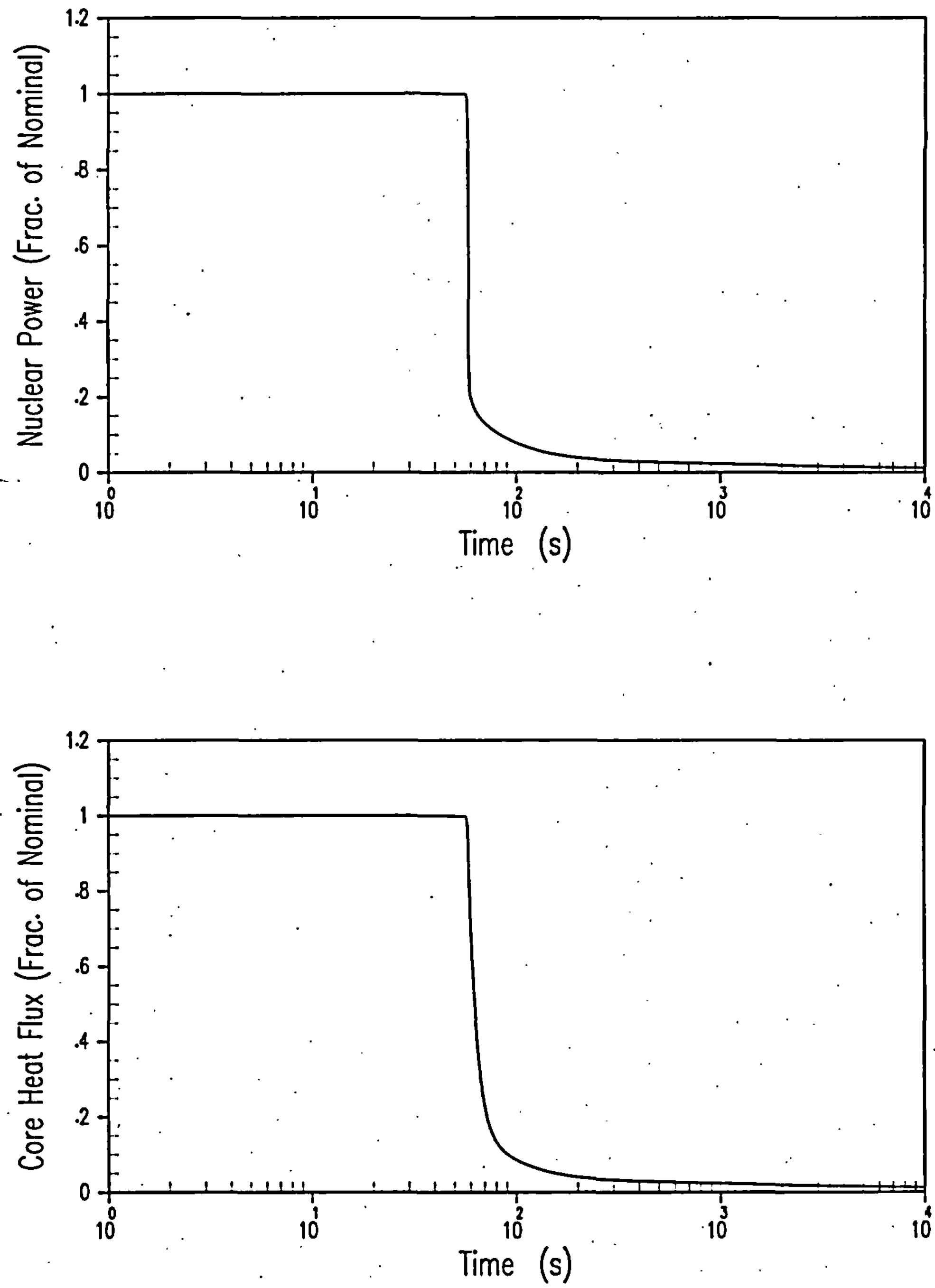


Figure 2.8.5.2.3-1
LONF Nuclear Power and Core Average Heat Flux vs. Time

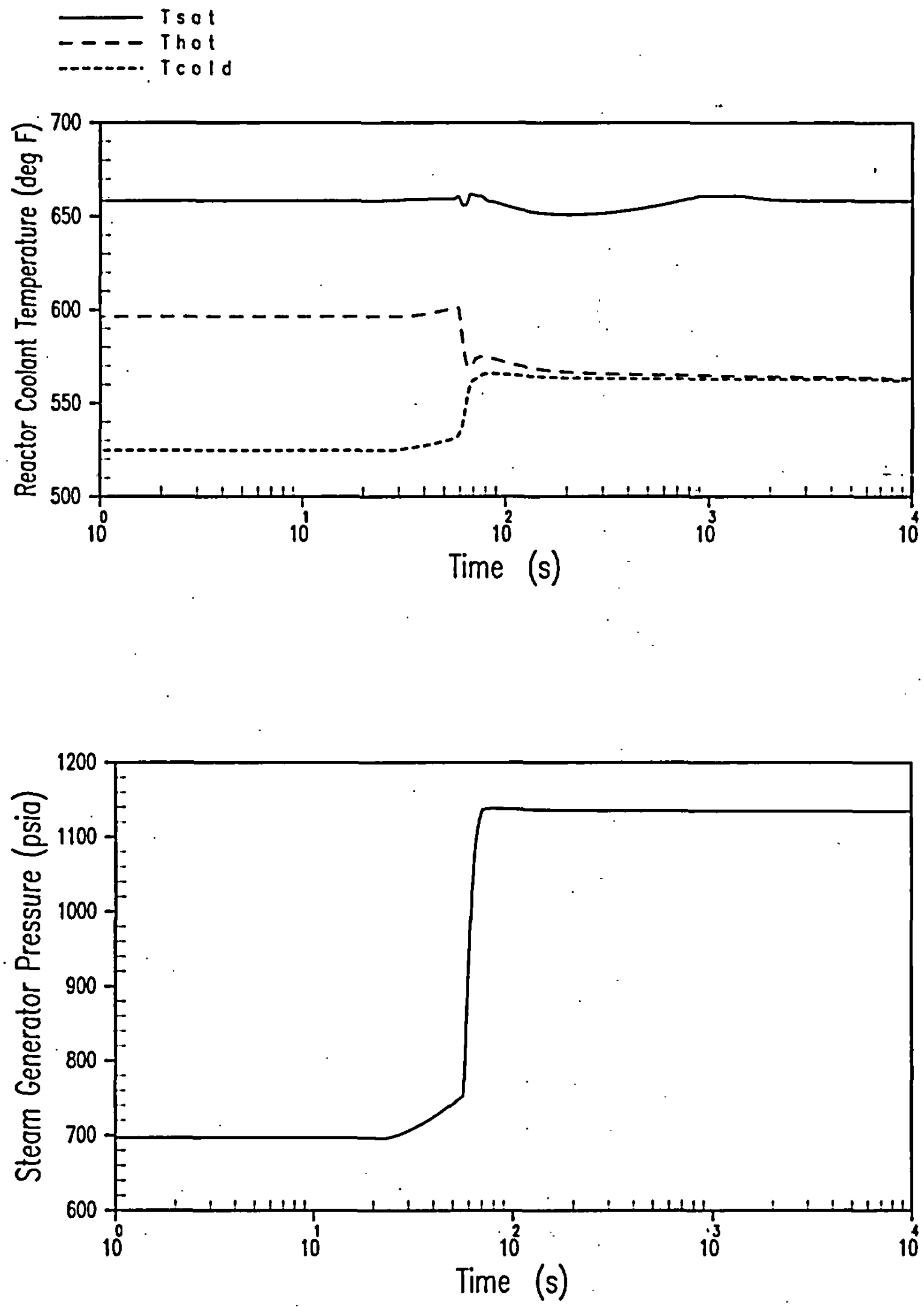
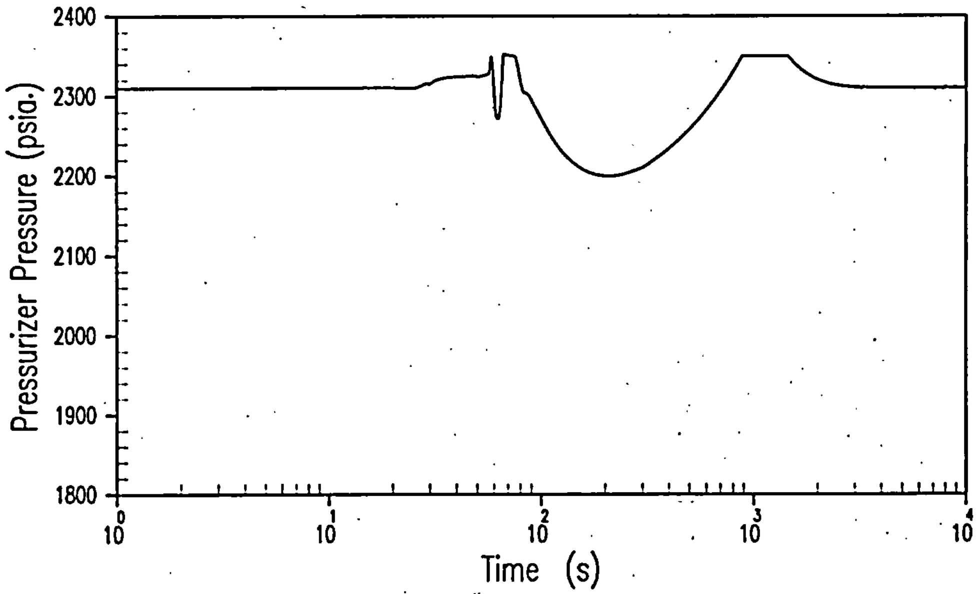


Figure 2.8.5.2.3-2
LONF RCS Temperatures and Steam Generator Pressure vs. Time



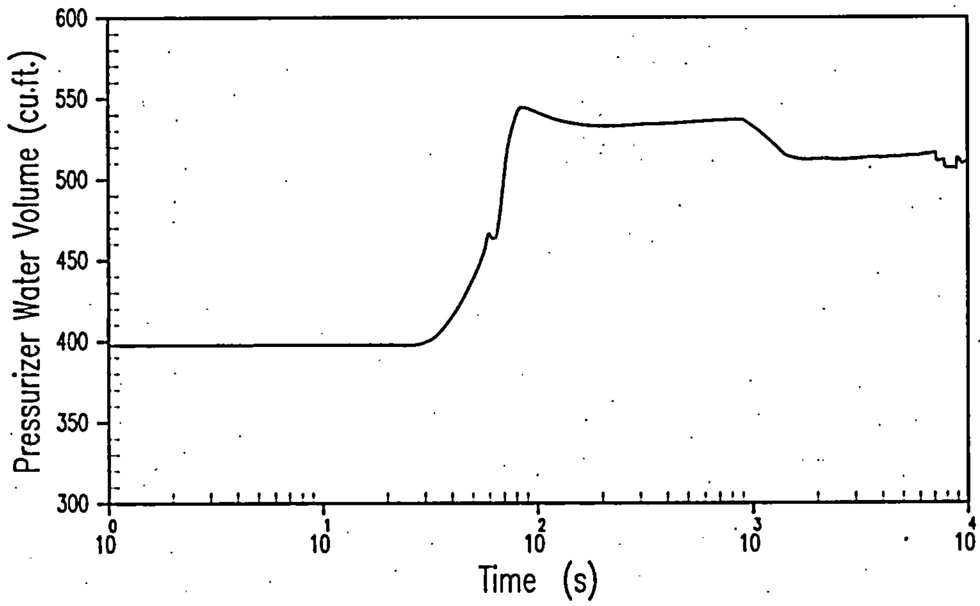


Figure 2.8.5.2.3-3
LONF Pressurizer Pressure and Water Volume vs. Time

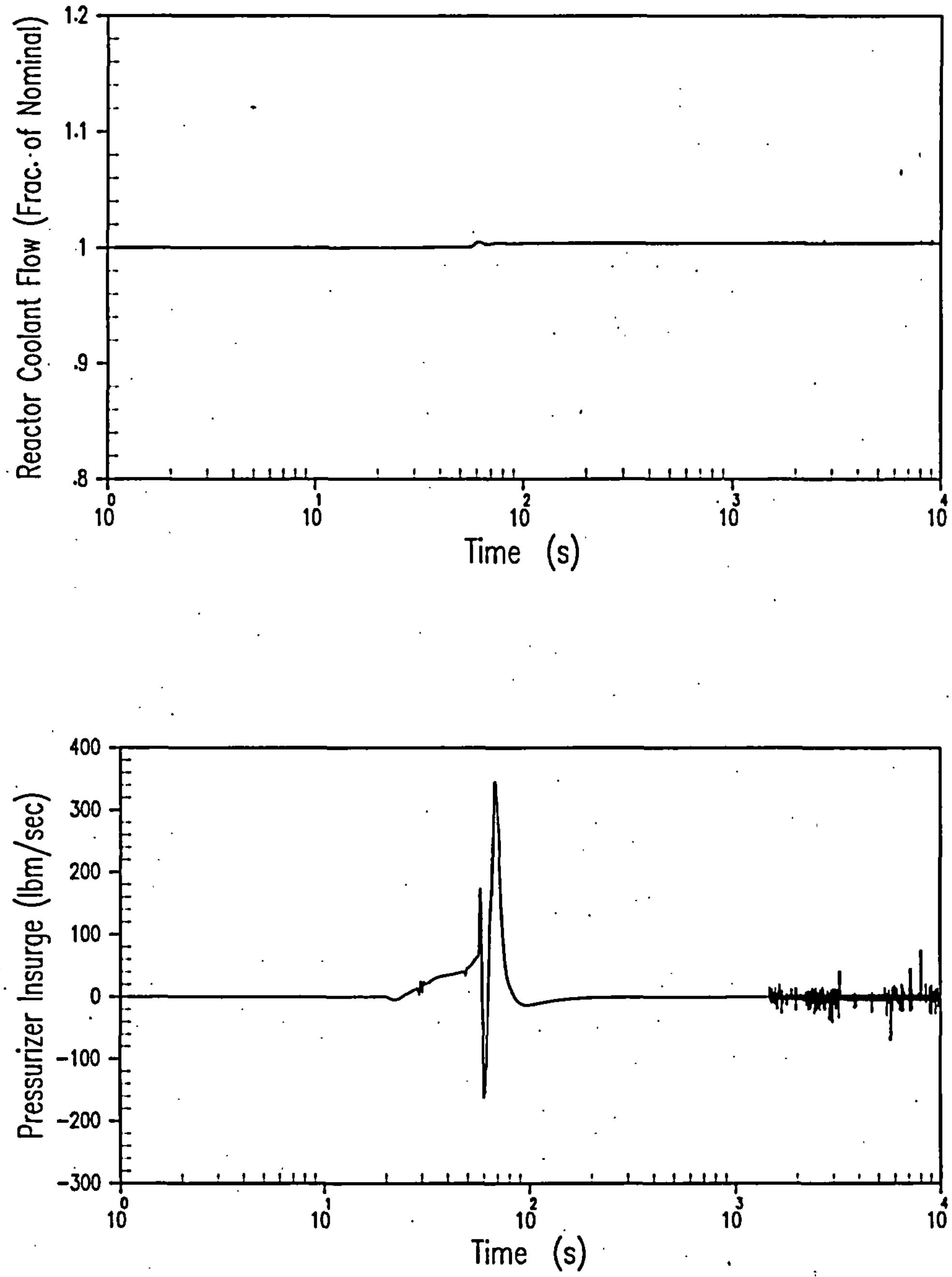


Figure 2.8.5.2.3-4
LONF Reactor Coolant Flow Rate and Pressurizer Insurge vs. Time

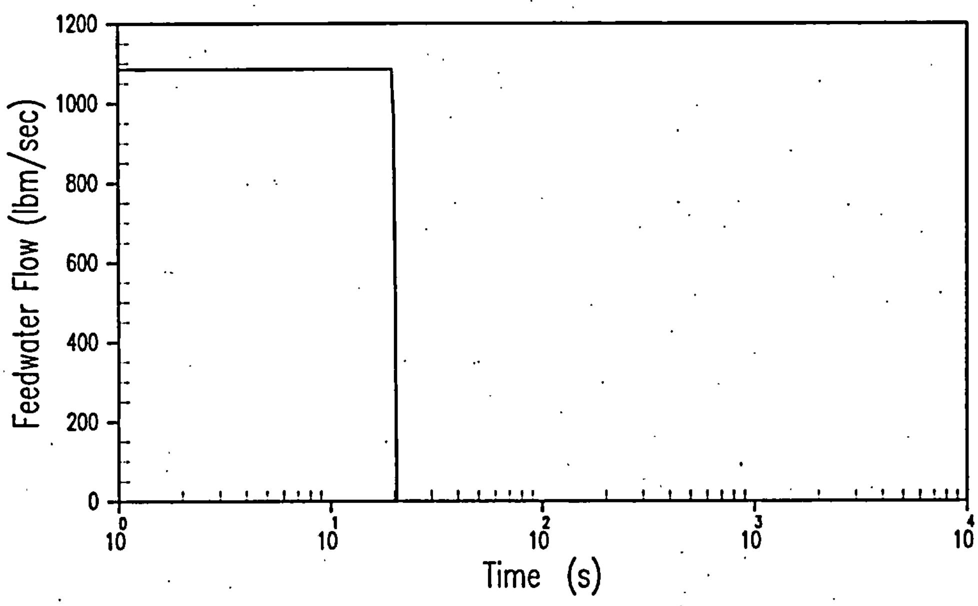


Figure 2.8.5.2.3-5
LONF Feedline Flow vs. Time

2.8.5.2.3.5 Conclusion

The Ginna staff has reviewed the analyses of the LONF event and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of the LONF. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements of GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the LONF event.

2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

2.8.5.2.4.1 Regulatory Evaluation

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system (RCS) cooldown (by excessive energy discharge through the break) or an RCS heatup (by reducing feedwater flow to the affected RCS). In either case, reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The postulated initial core and reactor conditions
- The methods of thermal-hydraulic analyses
- The sequence of events
- The assumed response of the reactor coolant and auxiliary systems
- The functional and operational characteristics of the reactor protection system
- The operator actions
- The results of the transient analyses

The NRC's acceptance criteria are based on:

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, thus ensuring that core cooling capability is maintained

- GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core

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

- GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling

Specific review criteria are contained in the SRP, Section 15.2.8, and other guidance provided in Matrix 8 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), the general design criteria use 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 for 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 analyses of feedwater system breaks inside and outside containment relative to conformance to

- GDC-27 is described in Ginna UFSAR section 3.1.2.3.8, General Design Criterion 27 – Combined Reactivity Control System Capability. As described in this UFSAR section, Ginna Station is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. This is accomplished through combined use of the rod cluster control system for neutron absorption within the core and the chemical and volume control system (CVCS) for chemical (boric acid) absorption of neutrons within the RCS. For various accident sequences, additional concentrated boric acid can be injected into the RCS by the emergency core cooling system (ECCS). The combined capability of these systems ensures core cooling is maintained. The fuel system design basis is further discussed in UFSAR section 4.2.1. The ECCS design and operation are further discussed in UFSAR section 6.3.

- GDC-28 is described in Ginna UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. As described in this UFSAR section, the maximum reactivity limits of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree which could impair the effectiveness of emergency core cooling. Fuel system design is further discussed in UFSAR section 4.2.1. The analysis of feedwater pipe breaks and their effects on core reactivity are discussed in UFSAR section 15.2.7. In addition, feedwater line break analysis was evaluated as part of the SEP review (SEP Topic XV-6) and found acceptable.

- GDC-31 is described in Ginna UFSAR section 3.1.2.4.2, General Design Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary. As described in this UFSAR section, the reactor coolant pressure boundary (RCPB) was fabricated, inspected and tested in accordance with codes applicable at the time. The fracture toughness properties of the RCPB and the reactor vessel are further discussed in UFSAR sections 5.2 and 5.3.1.2, respectively. As part of the SEP review, original design codes were compared to later licensing criteria (SEP Topic III-1) which is discussed in UFSAR section 3.2. Reactor vessel integrity was evaluated as part of SEP Topic V-6 which is further discussed in UFSAR section 5.3.3. Operating limitations are imposed during RCS startup and shutdown to prevent non-ductile failure of the RCPB. An on-going inservice inspection program is maintained for RCPB components in accordance with Ginna Technical Specifications to provide early detection of any fatigue indications.

- GDC-35 is described in Ginna UFSAR section 3.1.2.4.6, General Design Criterion 35 – Emergency Core Cooling. As described in this UFSAR section, Ginna design includes an emergency core cooling system (ECCS) which is capable of providing cooling water to the reactor core in response to various postulated accidents at a rate sufficient to maintain the core in a coolable geometry and to ensure that the clad metal-water reaction is limited. The ECCS is further discussed in UFSAR section 6.3.

In addition to the evaluations described in the UFSAR, the Ginna feedwater system pipe break analysis 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 analysis of feedwater system pipe breaks is not within the scope of license renewal.

2.8.5.2.4.2 Technical Evaluation

The specific acceptance criterion applied by the Ginna staff for this event was that there is no boiling in the hot legs prior to the point in the transient where the heat removal capacity of the auxiliary feedwater (AFW) system exceeds the heat generation. This conservatively ensured that the core remained covered and geometrically intact for the duration of the event. Furthermore, the analysis ensured that appropriate margin for malfunctions, such as stuck rods, were accounted for in the safety analysis assumptions. This conservatively satisfies the Ginna current licensing basis with respect to the requirements of GDC-27, GDC-28, GDC-31 and GDC-35.

The discussion below demonstrates that all applicable acceptance criteria are met for this event at Ginna Station at EPU conditions.

2.8.5.2.4.2.1 Introduction

A major feedwater line break (UFSAR section 15.2.7) is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator can also be discharged through the break. Furthermore, a break in this location could preclude the subsequent addition of AFW to the affected steam generator. A break upstream of the feedline check valve would affect the nuclear steam supply system (NSSS) only as a loss of feedwater. This case is covered by the loss of normal feedwater (LONF) analysis presented in LR section 2.8.5.2.3.

Depending upon the size of the break and the plant operating conditions at the time of the rupture, the break could either cause an RCS heatup or cooldown. The potential RCS cooldown resulting from a secondary pipe break is evaluated in the steamline break analysis presented in LR section 2.8.5.1.2.2.1. Only the RCS heatup effects of a feedline break are presented in this section.

A feedline break reduces the ability to remove heat generated by the core from the RCS. The AFW system is provided to ensure that adequate feedwater is available to provide decay heat removal.

2.8.5.2.4.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The following key assumptions were made in the final cases:

- EPU NSSS power up to 1817 MWt was assumed.
- The initial RCS average temperature was set to 580.0°F; the nominal high T_{avg} value of 576.0°F, plus a T_{avg} uncertainty of 4.0°F.
- The initial RCS pressure was 60 psid below its nominal value of 2250 psia to account for initial condition uncertainties.
- The initial pressurizer level was set to the nominal full power programmed value of 60% span plus 5% span to account for initial condition uncertainties.
- The initial steam generator water level was set to the nominal value (52% narrow range span) plus 8% narrow range span in the faulted steam generator

and the nominal value minus 4% NRS in the intact steam generator to account for initial condition uncertainties.

- The main feedwater flow to all steam generators was assumed to be lost at the time the break occurred (all main feedwater spilled out through the break).

- The full double-ended main feedwater pipe break was assumed. A break size of 1.418 ft^2 was analyzed for Ginna.

- Auxiliary feedwater (AFW) is the only engineered safety feature system assumed to function in this analysis. The flow from the motor-driven auxiliary feedwater (MDAFW) pump aligned to the faulted steam generator and the flow from the turbine-driven auxiliary feedwater (TDAFW) pump are assumed to be directed out the feedline break. Therefore, the worst case failure is the failure of the MDAFW pump aligned to the intact steam generator. The analysis allows for the realignment of this system or the startup of the standby auxiliary feedwater system to provide flow to the intact steam generator.

- Pressurizer PORVs were assumed to be available.

- Reactor trip was assumed to be actuated when the steam generator low-low level trip setpoint was reached in the ruptured steam generator. A conservative mass setpoint corresponding to 0% narrow range span was modeled (reference 3).

- The following AFW assumptions were made: for a feedline break outside the intermediate building, 195 gpm of AFW went to the intact steam generator and was initiated 60 seconds after the steam generator low-low level signal. For a feedline break inside the intermediate building, 235 gpm of AFW went to the intact steam generators and was initiated 870 seconds after the steam generator low-low level signal. In all cases, no AFW went to the faulted steam generator.

- The check valve in the feedline was assumed to close immediately upon turbine trip due to backflow in the feedline. This served to isolate the faulted steam generator.

- Credit was taken for heat energy deposited in portions of the RCS metal during the RCS heatup, as described in the approved methods presented in reference 3.

- No credit was taken for charging or letdown.

- Steam generator heat transfer area was assumed to decrease as the shell-side liquid inventory decreased.

- Conservative feedwater line break discharge quality was assumed. This minimized the heat transfer capability of the faulted steam generator.

- Conservative core decay heat was assumed based upon long-term operation at the initial power level preceding the trip (ANS-5.1-1979 plus 2 σ uncertainty).

- No credit was taken for the following potential protection logic signals to mitigate the consequences of the accident:

– High-pressurizer pressure
– High-pressurizer level
– High-containment pressure
– Overtemperature ΔT

The feedline break accident is an ANS Condition IV occurrence. Condition IV events are faults that are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material.

The specific criteria used in evaluating the consequences of the feedline break were:

- Pressures in the RCS and MSS are maintained below 110% of the design pressures.

- Any fuel damage that can occur during the transient is of a sufficiently limited extent that the core will remain in place and geometrically intact with no loss of core cooling capability.

- Any activity release is such that the calculated doses at the site boundary are within 10CFR50.67 (reference 1).

To conservatively meet these basic criteria, the internal criterion established by Ginna is that no bulk boiling occurs in the primary coolant system following a feedline break prior to the time that the heat removal capability of the steam generator, being fed AFW, exceeds NSSS residual heat generation.

2.8.5.2.4.2.3 Description of Analyses and Evaluations

The transient response following a feedline break event was calculated by a detailed digital simulation of the plant. The analysis modeled a simultaneous loss of main feedwater to both steam generators and subsequent reverse blowdown of the faulted steam generator. The analysis was performed using the RETRAN code (reference 2), which simulated the neutron kinetics, RCS, pressurizer, pressurizer relief valves and PSV, pressurizer spray, steam generators, and steam generator safety valves. The code computed pertinent plant variables including temperatures, pressures, and power level.

The following eight cases were analyzed for Ginna to determine the limiting cases:

Case (1) Maximum reactivity feedback, with offsite power, 1.418 ft^2 break outside intermediate building

Case (2) Maximum reactivity feedback, with offsite power, 1.418 ft^2 break inside intermediate building

Case (3) Maximum reactivity feedback, w/o offsite power, 1.418 ft^2 break outside intermediate building

Case (4) Maximum reactivity feedback, w/o offsite power, 1.418 ft^2 break inside intermediate building

Case (5) Minimum reactivity feedback, with offsite power, 1.418 ft^2 break outside intermediate building

Case (6) Minimum reactivity feedback, with offsite power, 1.418 ft^2 break inside intermediate building

Case (7) Minimum reactivity feedback, w/o offsite power, 1.418 ft^2 break outside intermediate building

Case (8) Minimum reactivity feedback, w/o offsite power, 1.418 ft^2 break inside intermediate building

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

The feedwater line break analyses are not within the scope of license renewal since they are analytical products of postulated events. Systems and system components associated with the analysis that are within the scope of license renewal are addressed in their respective system sections 2.3.1 (Reactor Systems), 2.3.2 (Engineered Safety Features), and 2.3.4 (Steam and Power Conversion Systems). Aging management programs applicable to these systems and components are addressed in SER section 3.1 (Reactor Coolant System), 3.2 (Engineered Safety Features Systems), and 3.4 (Steam and Power Conversion Systems).

No systems or system components are being added or modified as the result of re-evaluation of the spectrum of feedwater line breaks analyzed for EPU conditions. The analyses 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 feedwater line break analyses.

2.8.5.2.4.3 Results

The results of the feedline break cases analyzed showed that all acceptance criteria noted above were met. No bulk boiling occurred in the primary coolant system following a feedline break prior to the time that the heat removal capability of the steam generators, being fed AFW, exceeded NSSS residual heat generation.

For Ginna, Case 6 was the limiting case. This case analyzed a feedline break occurring inside the Intermediate Building, with offsite power available, minimum reactivity feedback, and a break size of 1.418 ft^2. The transient results for this case are presented in Figures 2.8.5.2.4-1 through 2.8.5.2.4-5. The time sequence of events for this case is presented in Table 2.8.5.2.4-1.

The transient results for the similar Case 8, but with offsite power unavailable, are presented in Figures 2.8.5.2.4-6 through 2.8.5.2.4-10. This case models a feedline break inside the Intermediate Building, minimum reactivity feedback, and a break size of 1.418 ft^2 with offsite power unavailable.

Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.2.4-2. In all cases, the EPU analyses are more limiting than the previous analyses.

The results of the analyses performed for Ginna at EPU conditions showed that for the postulated feedwater line rupture, AFW system capacity was adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

2.8.5.2.4.4 References

1. 10CFR50.67, *Accident Source Term.*

2. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, April 1999.

3. WCAP -14882-S1-P, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, Supplement 1 – Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method*, December 2002.

Table 2.8.5.2.4-1 Time Sequence of Events – Major Rupture of a Main Feedwater Pipe		
Case	**Event**	**Time (sec)**
Feedline Rupture Outside Intermediate Building with Offsite Power Available, Minimum Reactivity Feedback, Break Size of 1.418 ft^2	Main feedline rupture occurs	20.0
	Low-low steam generator water level reactor trip setpoint reached in ruptured steam generator	21.8
	Rods begin to drop	23.8
	Steamline check valves close on turbine trip	24.3
	First steam generator safety valve setpoint reached in intact steam generator	36.0
	AFW is started	891.8
	Minimum margin to hot leg saturation	~2600
	Hot and cold leg temperatures begin to decrease	~2600

Table 2.8.5.2.4-2 Major Rupture of a Main Feedwater Pipe - Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Minimum Margin to Boiling in the Hot Leg (°F)	2	13.1	0

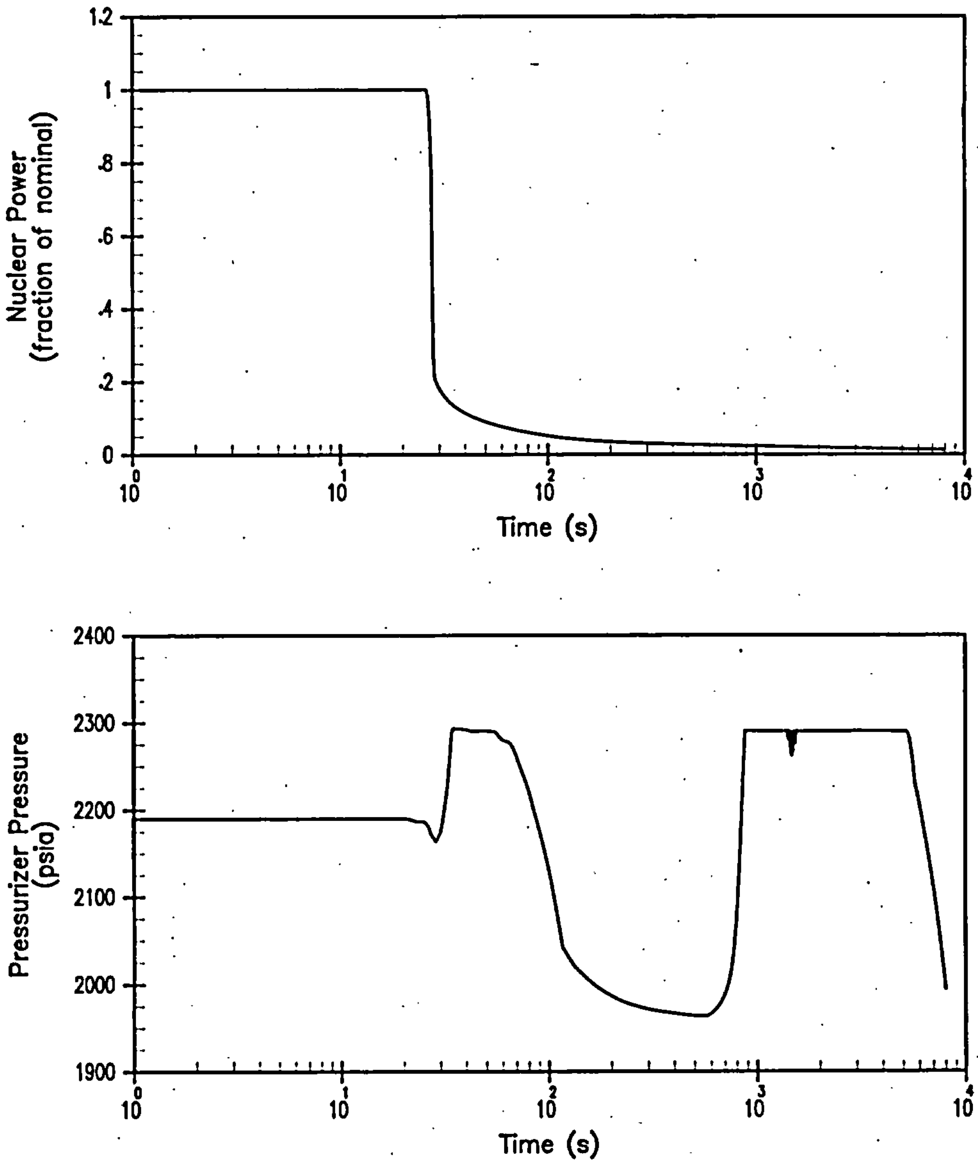


Figure 2.8.5.2.4-1
Feedline Break with Offsite Power, Nuclear Power and Pressurizer Pressure vs. Time

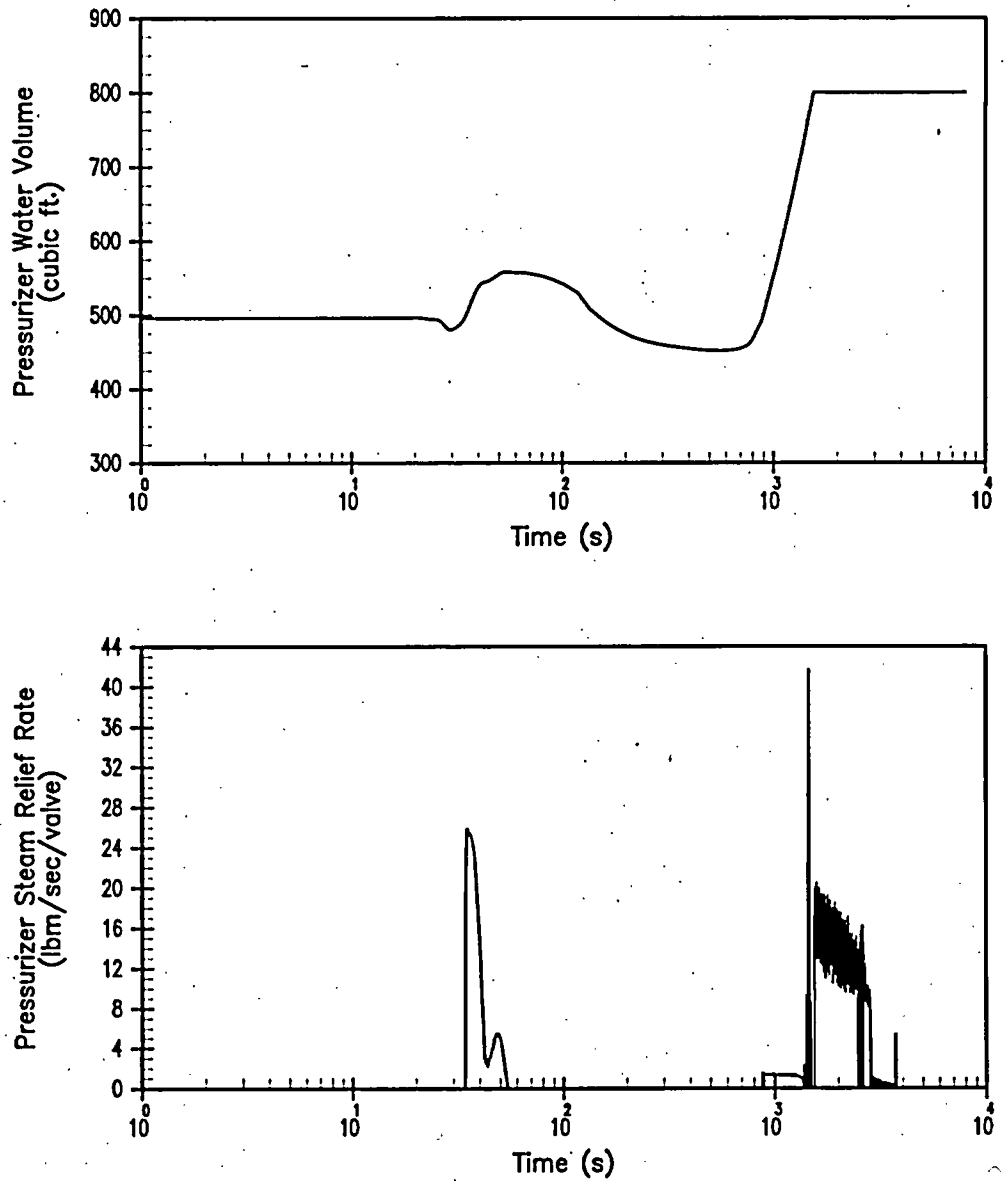


Figure 2.8.5.2.4-2
Feedline Break with Offsite Power, Pressurizer Water Volume and Pressurizer Steam Relief Rate vs. Time

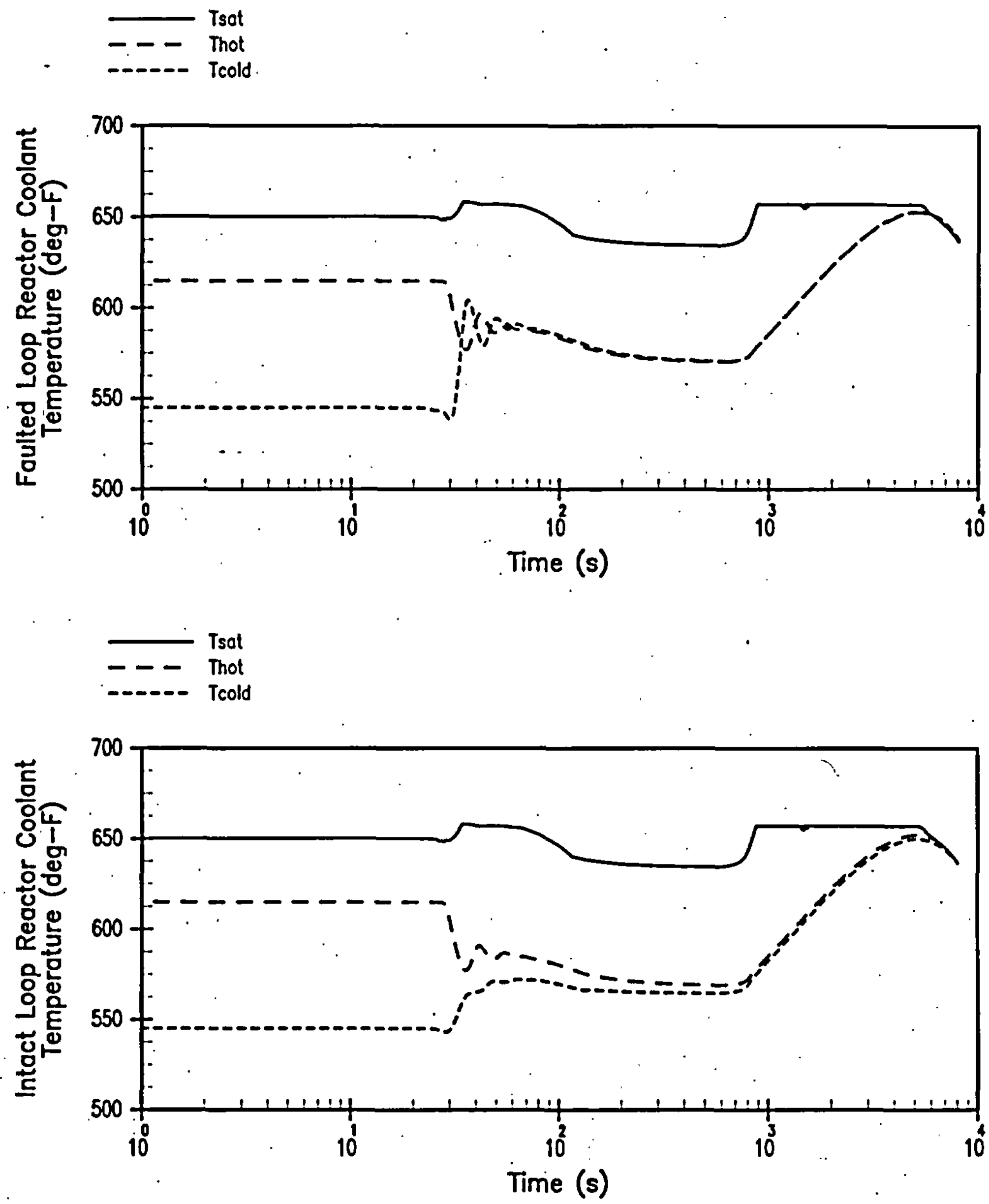


Figure 2.8.5.2.4-3
Feedline Break with Offsite Power, Reactor Coolant Temperature vs. Time for the Faulted and Intact Loops

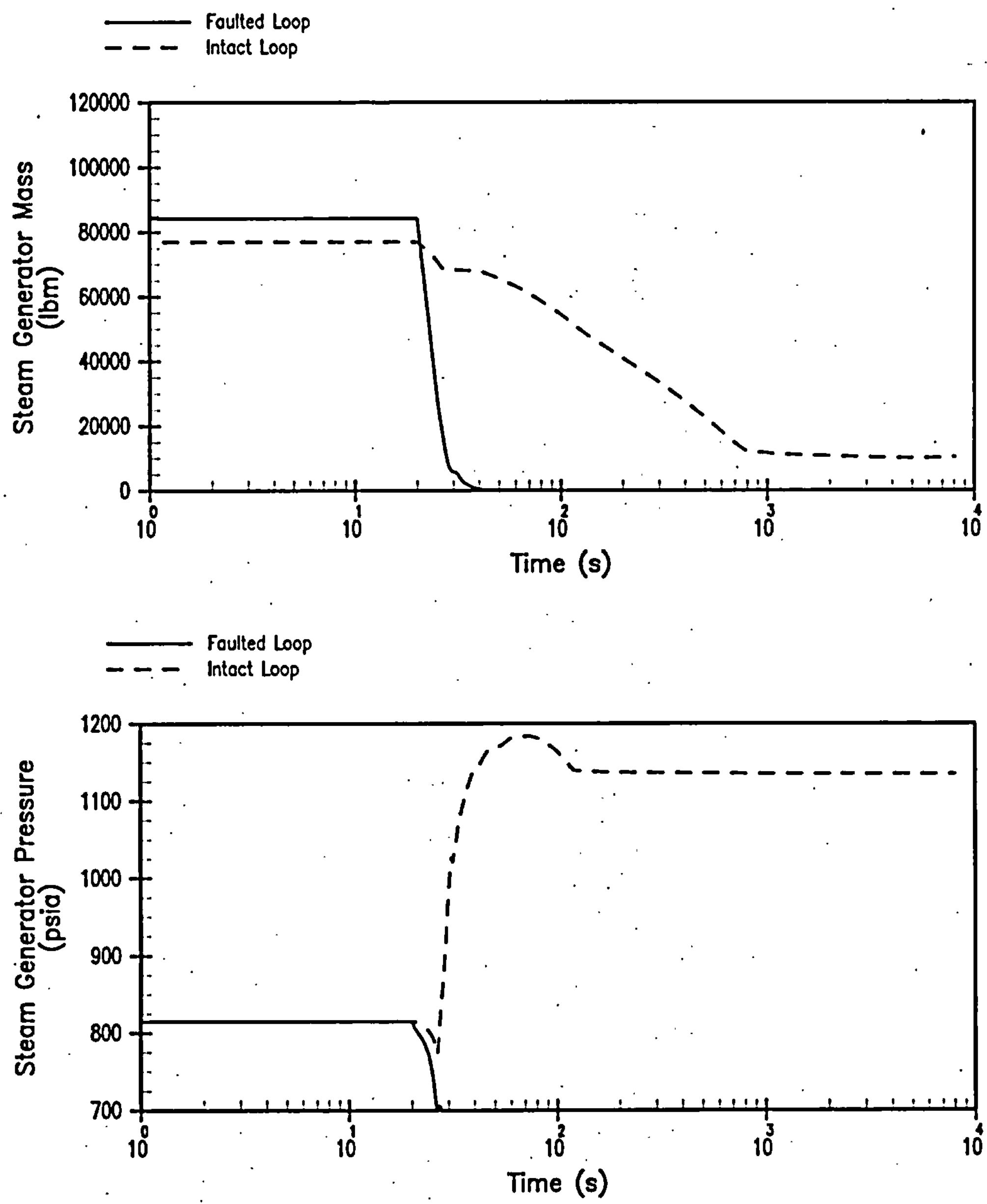


Figure 2.8.5.2.4-4
Feedline Break with Offsite Power, Steam Generator Mass and Steam Generator Pressure vs. Time

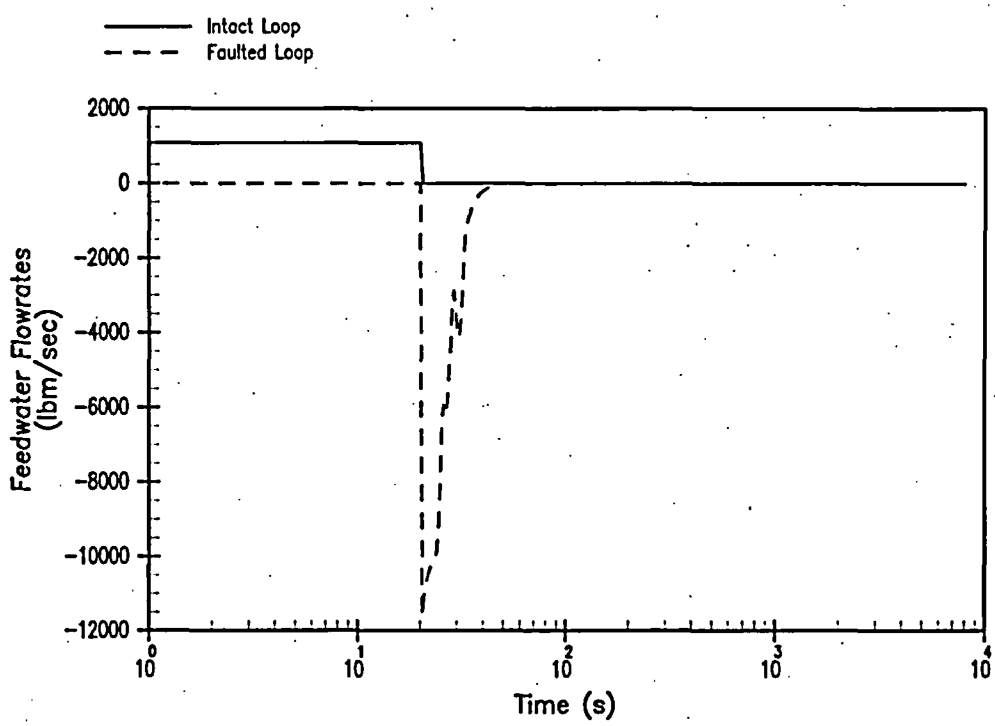


Figure 2.8.5.2.4-5

Feedline Break with Offsite Power, Feedwater Mass Flow Rates vs. Time

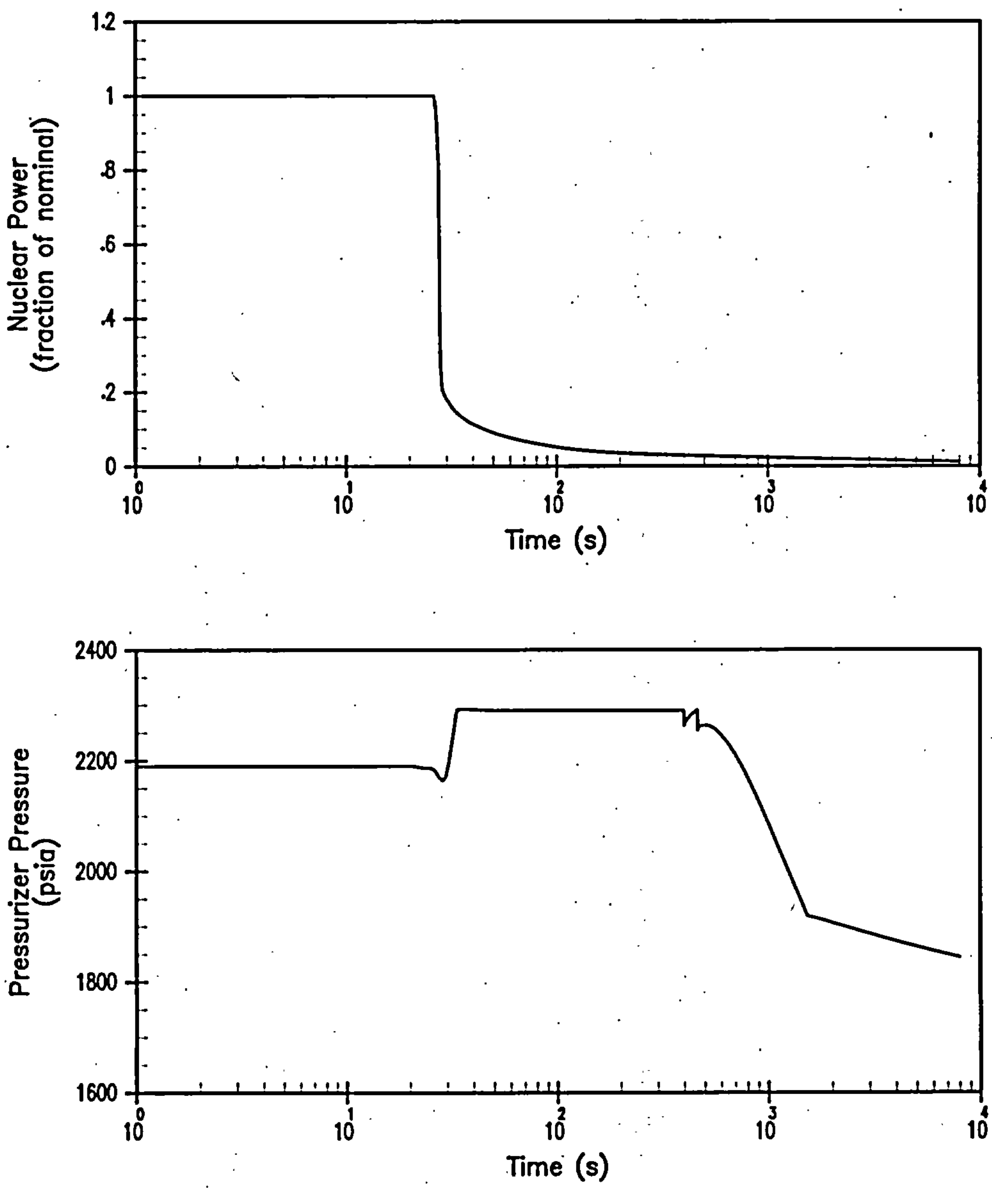
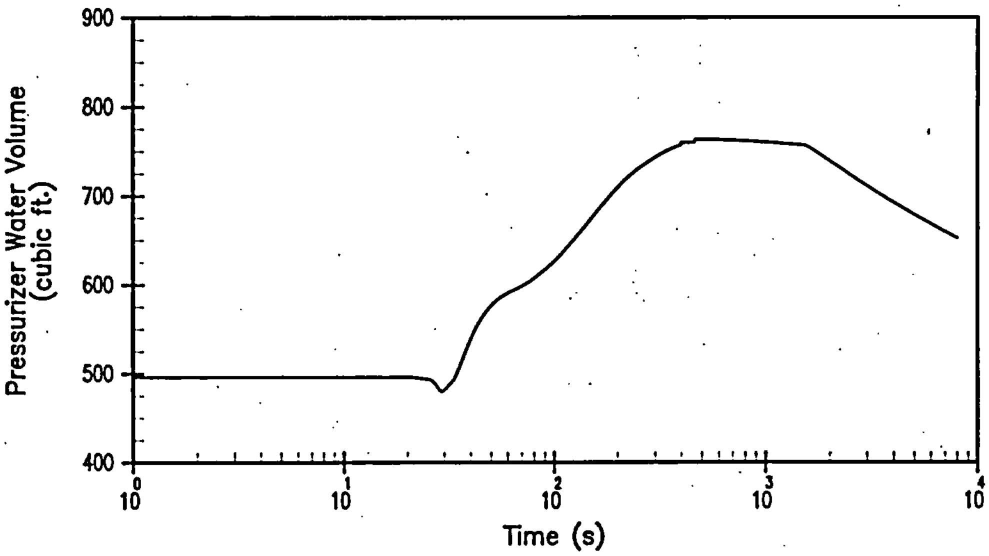


Figure 2.8.5.2.4-6
Feedline Break without Offsite Power, Nuclear Power and Pressurizer Pressure vs. Time



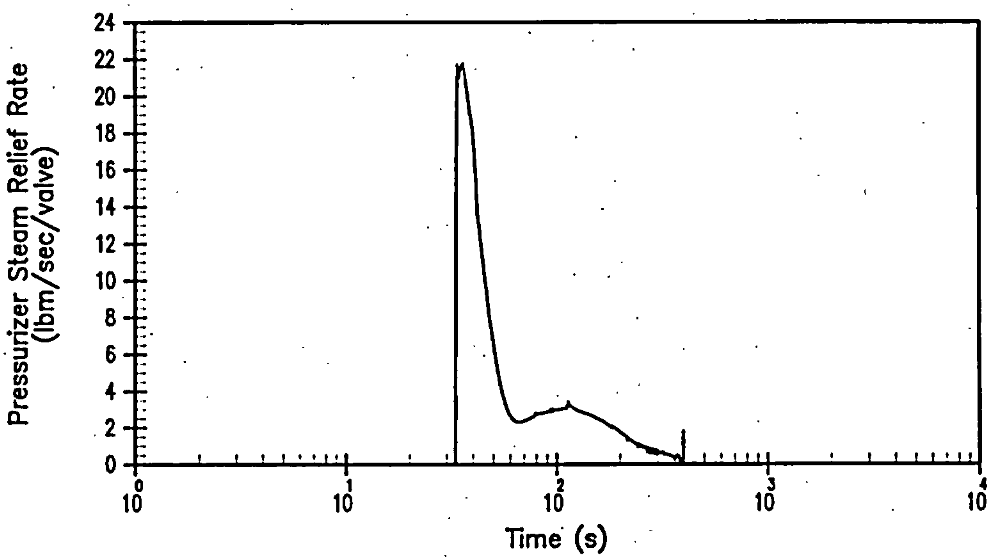


Figure 2.8.5.2.4-7
Feedline Break without Offsite Power, Pressurizer Water Volume and Pressurizer Steam Relief Rate vs. Time

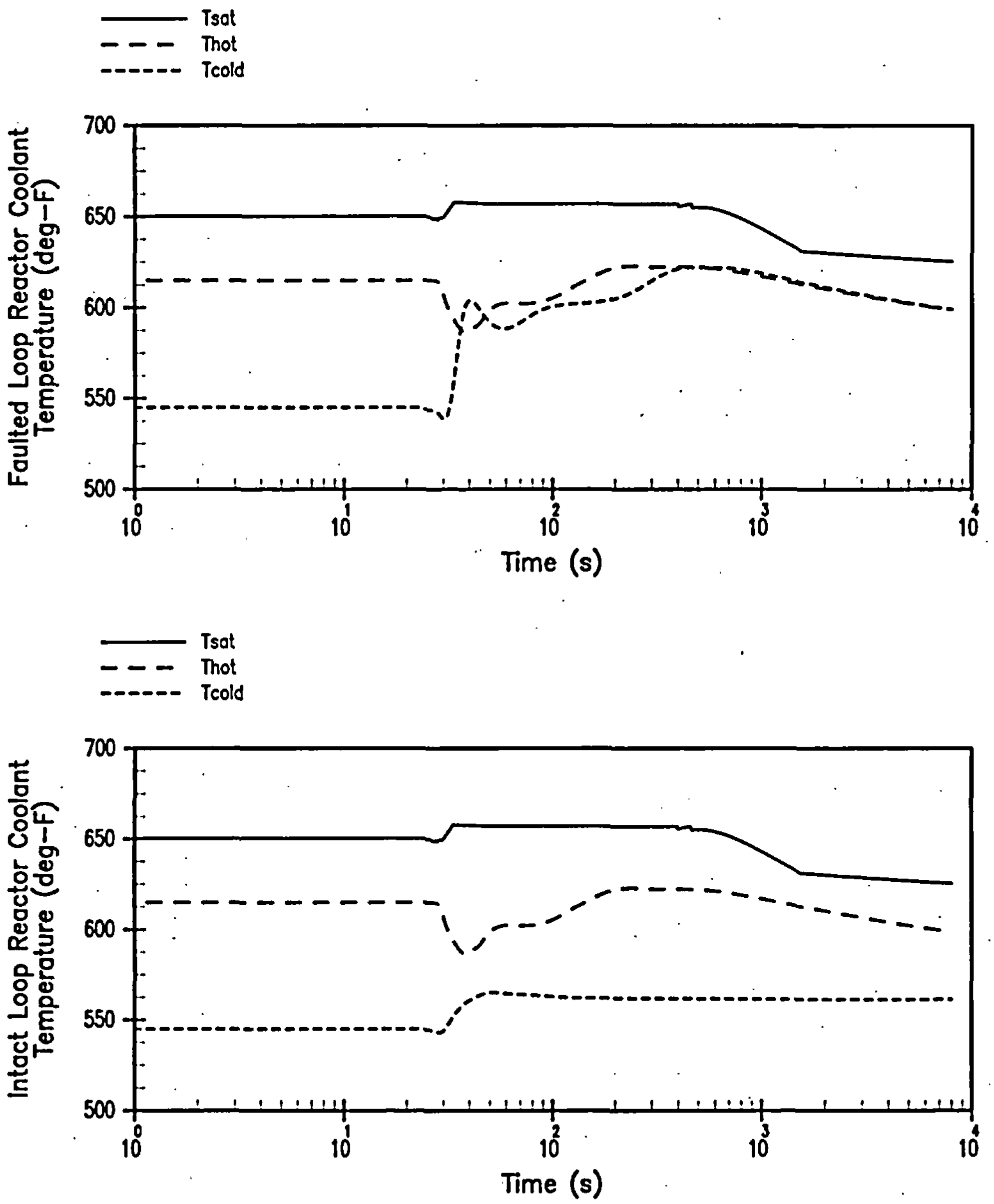


Figure 2.8.5.2.4-8

Feedline Break without Offsite Power, Reactor Coolant Temperature vs. Time for the Faulted and Intact Loops

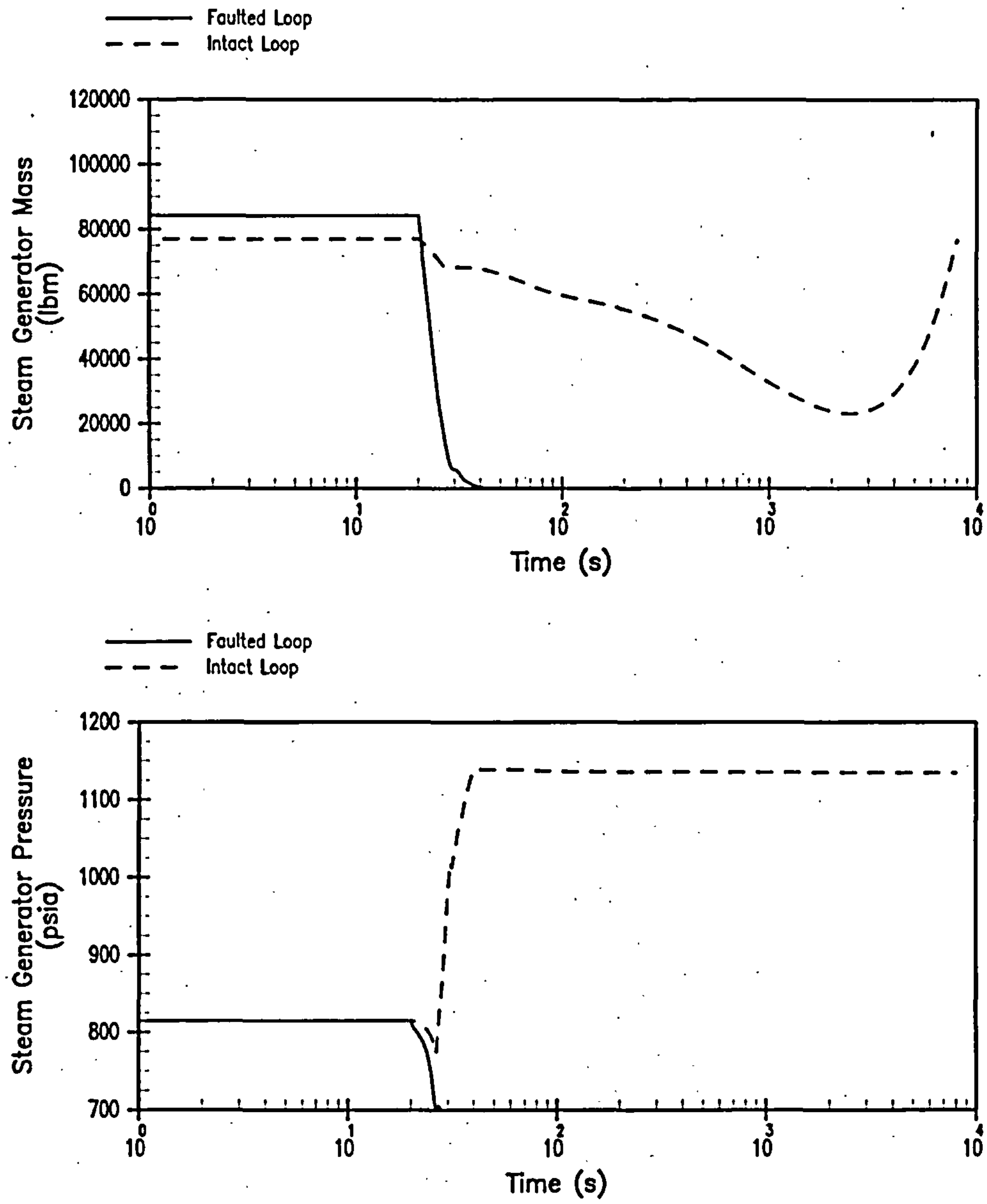


Figure 2.8.5.2.4-9
Feedline Break without Offsite Power, Steam Generator Mass and Steam Generator Pressure vs. Time

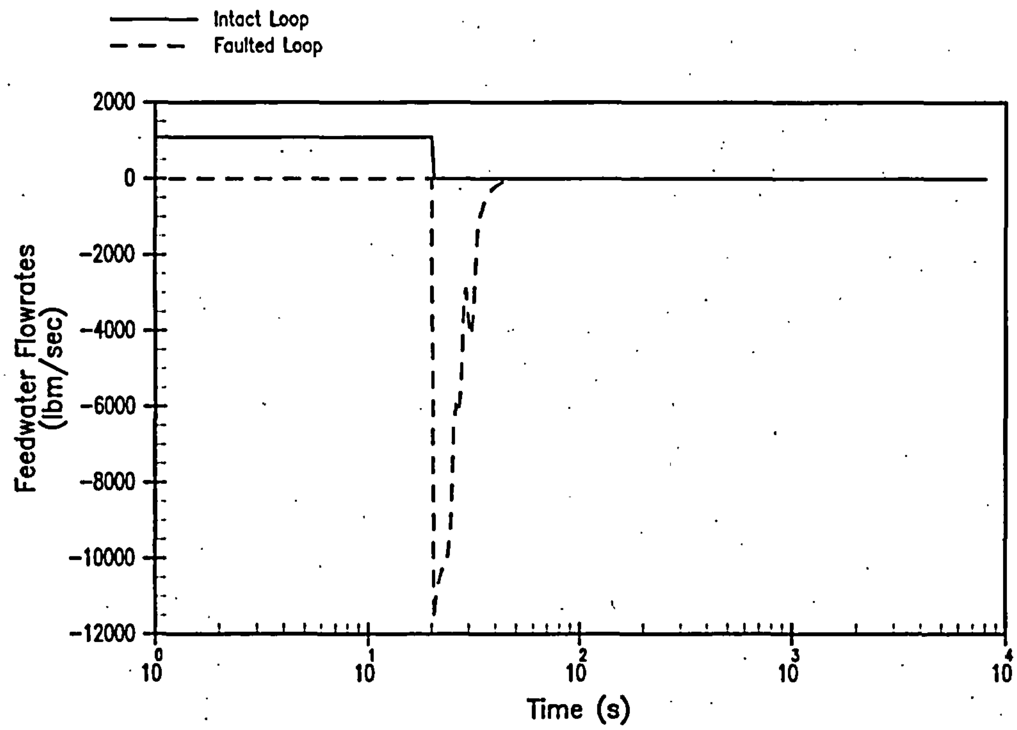


Figure 2.8.5.2.4-10
Feedline Break without Offsite Power, Feedwater Mass Flow Rates vs. Time

2.8.5.2.4.5 Conclusion

The Ginna staff has reviewed the analyses of feedwater system pipe breaks and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a non-brittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis with respect to the requirements of GDC-27, GDC-28, GDC-31, and GDC-35 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to feedwater system pipe breaks.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

2.8.5.3.1.1 Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel design limits are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The postulated initial core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed reactions of reactor systems components
- The functional and operational characteristics of the reactor protection system
- The operator actions
- The results of the transient analyses

NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems are designed with margin sufficient to ensure that the design condition of the reactor coolant pressure boundary is not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system is provided, and is capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded.

Specific review criteria are contained in the SRP, section 15.3.1-2 and other guidance provided in Matrix 8 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 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 loss of forced reactor coolant flow analysis relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criteria 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure the core is not damaged as the result of anticipated operational transients. The reactor control and protective system also prevents the power level or system temperature or pressure from exceeding safety limits. Fuel system design is further discussed in UFSAR section 4.2.

- GDC-15 is described in 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 (RCS) is designed to safely undergo all anticipated transients with pressure peaks below 2485 psig as further discussed in UFSAR chapter 15. The loss-of-coolant flow event is discussed in UFSAR section 15.3.1.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, the Ginna Station has two independent reactivity control systems of different design principles. The rod cluster control assemblies (RCCAs) provide neutron absorption capability within the reactor core. The RCCAs are design to shut down the reactor with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The chemical and volume control system (CVCS) regulates the concentration of neutron absorbing boric acid in the RCS. The CVCS is capable of controlling reactivity changes resulting from planned normal power changes. Reactivity control capability and redundancy are further discussed in UFSAR section 4.2.1.

In addition to the evaluations described in the UFSAR, the Ginna loss of forced reactor coolant flow analysis 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 loss of forced reactor coolant flow analysis is not within the scope of license renewal.

2.8.5.3.1.2 Technical Evaluation

2.8.5.3.1.2.1 Introduction

A loss of forced coolant flow event (UFSAR 15.3.1) can result from a simultaneous loss of electrical power supply or a reduction in power supply frequency to both reactor coolant pumps (RCPs). If the reactor is at power at the time of the event, the immediate effect from the loss of forced coolant flow is a rapid increase in the coolant temperature. This increase in coolant temperature could result in a violation of the DNBR limit, with subsequent fuel damage, if the reactor is not promptly tripped.

The following signals provide protection against a complete loss of forced reactor coolant flow incident:

- Low reactor coolant loop (RCL) flow
- Undervoltage or underfrequency on RCP power supply buses
- Pump circuit breaker opening

The reactor trip on low primary coolant loop flow provides protection against loss-of-flow conditions. This trip function is generated by two-out-of-three low-flow signals per RCL. When the reactor is operating at power levels above Permissive P-8, low flow in either loop will actuate a reactor trip. Between approximately 8% power (Permissive P-7) and the power level corresponding to Permissive P-8, low flow in both loops will actuate a reactor trip that is, low flow in only one loop will not actuate a reactor trip. Reactor trip on low flow is blocked below Permissive P-7.

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs, i.e., loss of offsite power (LOOP). An undervoltage reactor trip serves as an anticipatory backup to the low RCL flow trip. The undervoltage trip function is blocked below approximately 8% power (Permissive P-7).

The RCP underfrequency reactor trip is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. The RCP underfrequency reactor trip function is blocked below Permissive P-7. This trip function also serves as an anticipatory backup to the low RCL flow trip.

A reactor trip from pump breaker-open position is provided as a backup to the low-flow signal. Above Permissive P-8, a breaker-open signal from either pump will actuate a reactor trip. A breaker-open signal from both pumps will actuate a reactor trip above Permissive P-7. Reactor trip on RCP breakers open is blocked below Permissive P-7.

2.8.5.3.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

This event was analyzed using the Revised Thermal Design Procedure (RTDP) (reference 1). Initial core power was assumed to be at its nominal value consistent with steady-state, full-power operation. RCS pressure and RCS vessel average temperature were assumed to be at their nominal values. Uncertainties in the initial conditions were included in the DNBR limit value as described in the RTDP.

A conservatively large absolute value of the Doppler-only power coefficient was used. The analysis also assumed a conservative moderator temperature coefficient (MTC) of zero pcm/°F at hot full power (HFP) conditions. This resulted in the maximum core power and hot spot heat flux during the transient when the minimum DNBR is reached.

Engineered safety systems (e.g., safety injection) were not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event. A complete loss of forced reactor coolant flow event is classified by the American Nuclear Society as a Condition III event; however, for conservatism, the event was analyzed to Condition II criteria. The immediate effect from a complete loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature and subsequent increase in RCS pressure. The specific acceptance criteria for the event are:

- The critical heat flux is not to be exceeded. This criteria is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the RCS and MSS is maintained below 110% of their respective design pressures.
- The peak linear heat generation rate does not exceed a value that would cause fuel centerline melt.

2.8.5.3.1.2.3 Description of Analyses and Evaluations

The following complete loss of forced reactor coolant flow cases were analyzed:

- Complete loss of both RCPs with both loops in operation
- Frequency decay event resulting in a complete loss of forced reactor coolant flow

Note that a partial loss of flow event (i.e., one loop coasting down) was also considered. The results of that case were non-limiting compared to the complete loss of flow events.

A P-8 permissive setpoint evaluation was performed at EPU conditions. The P-8 permissive setpoint defines the highest steady-state power level at which the reactor can operate with one RCS loop inactive without violating the N-1 core thermal limits. The P-8 evaluation was performed with RETRAN by analyzing one loop in operation at a part power steady state condition and demonstrating that the DNBR design basis is satisfied. The RETRAN analysis was performed at 35% power and determined state points that were evaluated and found to satisfy the DNB limit. Therefore, the DNB design basis is satisfied for partial loss of flow at 35% power, demonstrating the acceptability of 35% as the P-8 permissive setpoint for EPU.

The complete loss of flow transients were analyzed with two computer codes. First, the RETRAN computer code (reference 2) was used to calculate the RCS loop and core flow during the transient, the time of reactor trip based on the calculated RCS flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code (reference 3) was then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and core flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel. Additional discussion of the RETRAN and VIPRE codes is contained in <u>LR section 2.8.5.0.9</u>, Computer Codes Utilized.

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

The loss of forced reactor coolant flow analysis is not within the scope of license renewal since it is an analytical product of a postulated accident. Systems and system components associated with this analysis that are within the scope of license renewal are addressed in their respective system sections 2.3.1 (Reactor Coolant Systems) and 2.3.2 (Engineered Safety Features Systems) of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786 sections 3.1 (Reactor Coolant Systems), and 3.2 (Engineered Safety Features Systems).

No systems or system components are being added or modified as the result of re-evaluation of the loss of forced reactor coolant flow analysis for EPU conditions. The analysis does 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 the loss of forced reactor coolant flow analysis.

2.8.5.3.1.3 Results

The complete loss of flow case was assumed to trip on an undervoltage reactor trip signal, and the frequency decay case was assumed to trip on an underfrequency reactor trip signal. The VIPRE (reference 3) analysis for these scenarios confirmed that the minimum DNBR acceptance criterion was met. Fuel clad damage criteria were not challenged in either of the complete loss of forced reactor coolant flow cases since the DNBR criterion was met. The more limiting of these two cases in terms of the minimum calculated DNBR was the frequency decay case. The transient results for this case are presented in Figures 2.8.5.3.1-1 through 2.8.5.3.1-3. The sequence of events for both cases is presented in Table 2.8.5.3.1-1. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.3.1-2. In all cases, the EPU analyses are more limiting than the previous analyses.

The analysis performed for the EPU demonstrates that, for the aforementioned complete loss of flow cases, the DNBR did not decrease below the safety analysis limit value at any time during the transients; thus, no fuel or clad damage is predicted. The peak primary and secondary system pressures remained below their respective limits at all times. All applicable acceptance criteria were therefore met.

The protection features presented in LR section 2.8.5.3.1.2.1, above, provide mitigation for the complete loss of forced reactor coolant flow transients such that the above criteria are satisfied. Furthermore, the results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload process.

2.8.5.3.1.4 References

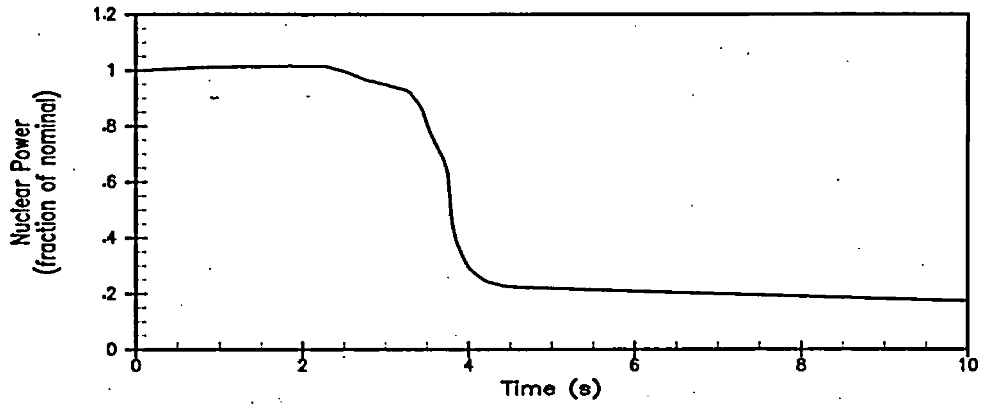
1. WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), Revised Thermal Design Procedure, Friedland, A. J. and Ray, S., April 1989.

2. WCAP-14882-P-A, RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, D. S. Huegel, et al., April 1999.

3. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-Proprietary), VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, Sung, Y. X. et al., October 1999.

Table 2.8.5.3.1-1 Time Sequence of Events – Loss of Forced Reactor Coolant Flow		
Case	**Event**	**Time (sec)**
Frequency Decay	Frequency Decay Begins	0.0
	Underfrequency Reactor Trip Setpoint Reached	0.6
	Rods Begin to Drop	2.0
	Minimum DNBR Occurs	3.4
	Maximum Primary Pressure Occurs	4.6
Complete Loss of Forced Reactor Coolant Flow	Flow Coastdown Begins	0.0
	Undervoltage Reactor Trip Setpoint Reached	0.0
	Rods Begin to Drop	1.5
	Minimum DNBR Occurs	2.9
	Maximum Primary Pressure Occurs	3.4

Table 2.8.5.3.1-2 Loss of Forced Reactor Coolant Flow – Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Minimum DNBR – Single RCP Coasting Down	1.597	1.855	1.38 (EPU)
Minimum DNBR – Both RCPs Coasting Down	1.489	1.723	1.38 (EPU)
Minimum DNBR – Frequency Decay on Both RCPs	1.385	1.604	1.38 (EPU)



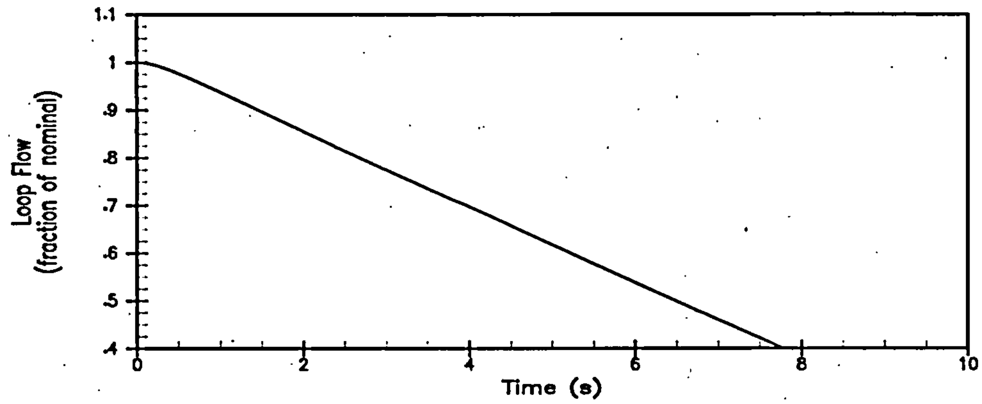


Figure 2.8.5.3.1-1
Loss of Forced Reactor Coolant Flow - Frequency Decay
Nuclear Power and Loop Flow vs. Time

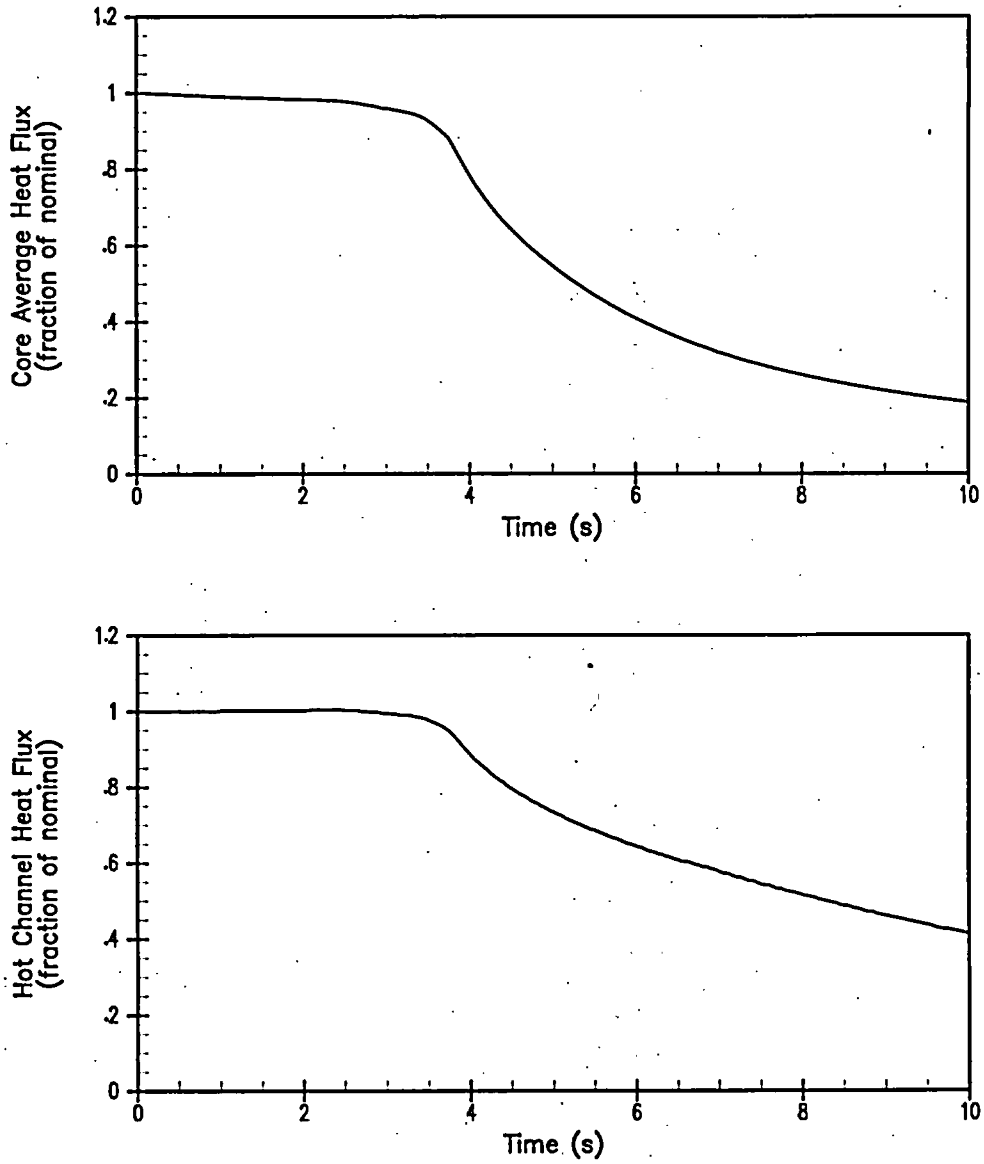


Figure 2.8.5.3.1-2
Loss of Forced Reactor Coolant Flow - Frequency Decay
Core Average and Hot Channel Heat Flux vs. Time

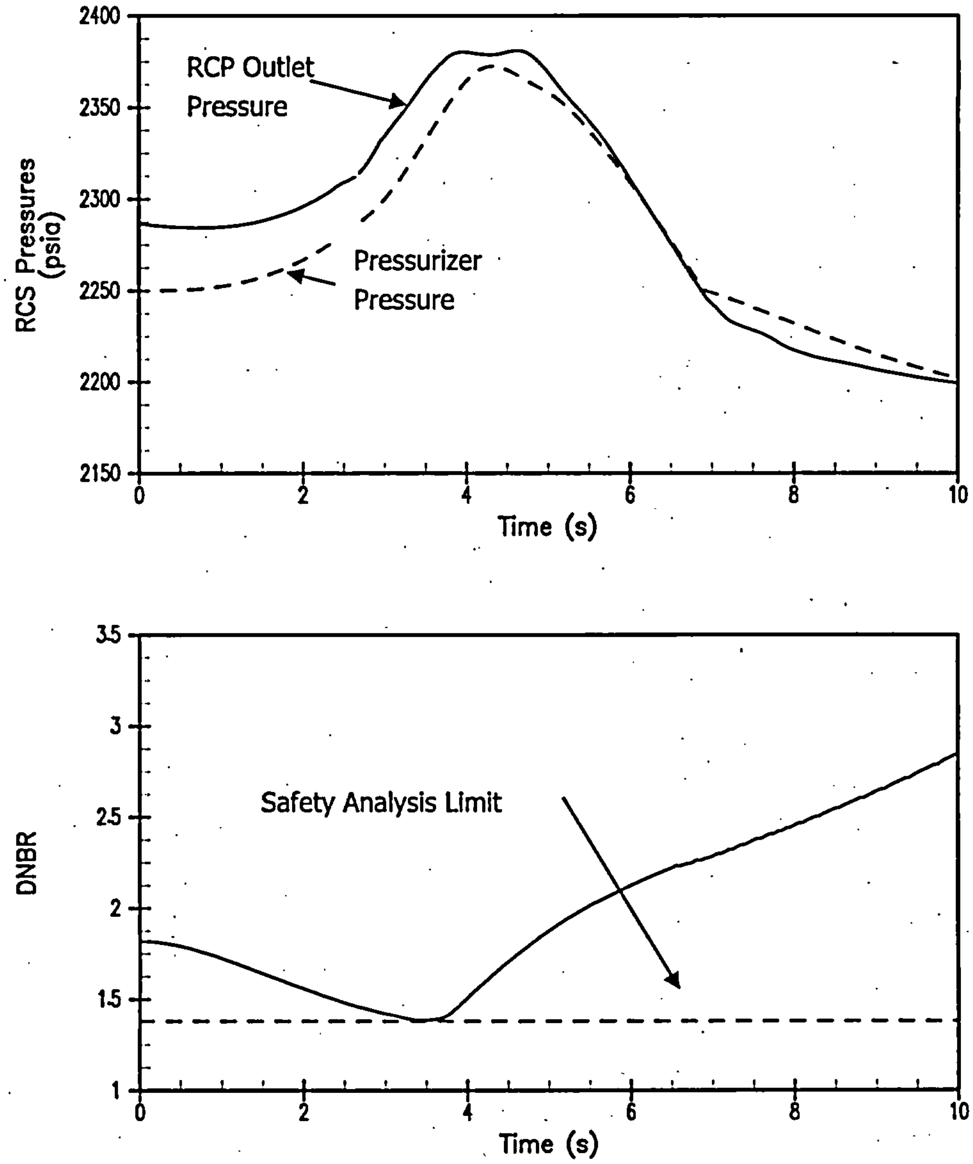


Figure 2.8.5.3.1-3
Loss of Forced Reactor Coolant Flow - Frequency Decay
RCS Pressures and DNBR vs. Time

2.8.5.3.1.5 Conclusion

The Ginna staff has reviewed the analyses of the decrease in reactor coolant-flow event and concludes that the analyses have adequately accounted for plant operations at the power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, The Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

2.8.5.3.2.1 Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (RCP). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The postulated initial and long-term core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor protection system
- The operator actions
- The results of the transient analyses

The NRC's acceptance criteria are based on:

- GDC-27, insofar as it requires that the reactivity control systems are designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained.

- GDC-28, insofar as it requires that the reactivity control systems are designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the core cooling capability.

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

Specific review criteria are contained in the NRC SRP section 15.3.3-4, and other guidance provided in Matrix 8 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 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 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 coolant pump rotor seizure/shaft break analysis relative to conformance to:

- GDC-27 is described in UFSAR section 3.1.2.3.8, General Design Criterion 27 – Combined Reactivity Control System Capability. As described in this UFSAR section, Ginna Station is provided with a means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. In addition, the reactivity control system, in conjunction with the Emergency Core Cooling System (ECCS), has the capability of controlling reactivity changes under postulated accident conditions. The fuel system design is further described in UFSAR section 4.2.1, and the ECCS design is further described in UFSAR sections 6.3.1 and 6.3.2.

- GDC-28 is described in UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. As described in this UFSAR section, the maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary (RCPB) or disruption of the core or vessel internals which could impair the effectiveness of ECCS. UFSAR section 4.2.1 discusses the design basis in meeting this criterion and UFSAR section 15.3.2 discusses the RCP locked rotor analysis.

- GDC-31 is described in UFSAR section 3.1.2.4.2, General Design Criterion 31 – Fracture Prevention in Reactor Coolant Pressure Boundary. As described in this UFSAR section, the RCPB was fabricated, inspected and tested in accordance with codes that were applicable at the time. An inservice inspection (ISI) program for the reactor vessel and RCS piping is maintained in accordance with the Ginna Technical Specifications. Operating limitations during plant startup and shutdown are imposed to protect against non-ductile failure of RCPB components. In addition, reactor vessel integrity has been evaluated as part of SEP Topic V-6 and unresolved safety issues (USI) A-49, Pressurized Thermal Shock, and A-11, Reactor Vessel Materials Toughness, as discussed in UFSAR section 5.3.3.

The SEP review also evaluated the RCP locked rotor/shaft break event (SEP Topic XV-7).

In addition to the evaluations described in the UFSAR, the Ginna RCP locked rotor/shaft break event analysis 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 RCP locked rotor/shaft break event analysis is not within the scope of license renewal.

2.8.5.3.2.2 Technical Evaluation

2.8.5.3.2.2.1 Introduction

The event postulated is an instantaneous seizure of a RCP rotor ("locked rotor") as described in UFSAR section 15.3.2 or the sudden break of the shaft of the RCP. Flow through the affected reactor coolant loop (RCL) is rapidly reduced, leading to initiation of a reactor trip on a low RCL flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell-side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient, and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip due to turbine trip on reactor trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic pressurizer spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves (PSVs), in that sequence. The PORVs are designed for reliable operation and are expected to function properly during the event. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the pressurizer spray, were not included in the analysis.

The consequences of a locked rotor are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenarios.

2.8.5.3.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The transient was analyzed using the Standard Thermal Design Procedure (STDP). The evaluated case assumed one faulted RCP with both RCLs initially in operation. The analysis models the most limiting combination of conditions from the locked rotor and pump shaft break events. This case made assumptions designed to maximize the RCS pressure transient. Initial core power, reactor coolant temperature, and pressure were assumed to be at their maximum values consistent with full-power conditions, including allowances for calibration and instrument errors. This assumption resulted in a conservatively high calculation of the coolant insurge into the pressurizer, which in turn resulted in a maximum calculated peak RCS pressure.

A zero moderator temperature coefficient (MTC) and a conservatively large (absolute value) Doppler-only power coefficient were assumed in the analysis. The negative reactivity from control rod insertion/scram for both cases was based on 3.5% $\Delta k/k$ trip reactivity from HFP.

Normal RCS and engineered safety systems (i.e., safety injection) were not required to function. No single active failure in any system or component required for mitigation adversely affected the consequences of this event.

The RCP locked rotor accident is classified by the ANS as a Condition IV event. An RCP locked rotor results in a rapid reduction in forced RCL flow that increases the reactor coolant temperature and subsequently causes the fuel cladding temperature and RCS pressure to increase.

The specific acceptance criteria applied by Ginna for this event were as follows:

- The peak clad termperature must remain below 2700°F and the maximum zirconium-water reaction must remain below 16%. Appropriate margin for malfunctions, such as stuck rods, were accounted for in the safety analysis assumptions. Demonstrating that these limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-27 and GDC-28.

- Primary and secondary pressures must remain below 120% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-31.

With respect to secondary side overpressurization, this event was bounded by the loss of load/turbine trip event as discussed in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum.

2.8.5.3.2.2.3 Description of Analyses and Evaluations

The following locked rotor/shaft break case was analyzed - Peak RCS pressure resulting from a locked rotor/shaft break in one-of-two loops

The locked rotor event was analyzed with two computer codes. First, the RETRAN computer code (reference 1) was used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (reference 2) was then used to calculate the peak cladding temperature using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. Additional discussion of the RETRAN and VIPRE codes is contained in LR section 2.8.5.0.9, Computer Codes Utilized.

For the peak RCS pressure evaluation, the initial pressure was conservatively estimated to be 60 psi above the nominal pressure of 2250 psia, to allow for initial condition uncertainties in the pressurizer pressure measurement and control channels. This was done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops were added to the calculated pressurizer pressure. The pressure response reported in Table 2.8.5.3.2-2 is at the point in the RCS having the maximum pressure, i.e., at the outlet of the RCP in the faulted loop.

No credit was taken for the pressure-reducing effect of the pressurizer PORVs, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these systems were expected to function and would result in a lower peak pressure, an additional degree of conservatism was provided by not including their effect.

The PSV model included a +3% valve tolerance, plus 5 psi accumulation above the nominal setpoint of 2500 psia. The model includes an additional 1% setpoint shift due to the loop seals and an additional delay of 0.8 seconds to account for the time to purge the loop seals, as discussed in WCAP-12910 (reference 3).

Film Boiling Coefficient

The film boiling coefficient was calculated in the VIPRE code (reference 2) using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties were evaluated at the film temperature. The program calculated the film coefficient at every time-step based upon the actual heat transfer conditions at the corresponding time step. The nuclear power, system pressure, bulk density, and RCS flow rate as a function of time were based on the RETRAN results.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) had a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat was transferred between the pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial maximum fuel temperatures to approximately 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Therefore, a large amount of energy stored in the fuel was released to the cladding at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above a cladding temperature of 1800°F . The Baker-Just parabolic rate equation was used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction was included in the calculation of the hot spot cladding temperature transient.

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

The RCP locked rotor/shaft break analysis is not within the scope of license renewal since it is an analytical product of a postulated event. Systems and system components associated with this analysis are within the scope of license renewal and are addressed in their respective system sections 2.3.1 (Reactor Systems) and 2.3.2 (Engineered Safety Features Systems) of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786 sections 3.1 (Reactor Coolant Systems) and 3.2 (Engineered Safety Features Systems) of NUREG-1786. No systems or system components are being added or modified as the result of re-evaluation of the RCP locked rotor/shaft break analysis for EPU conditions. The analysis does 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 the RCP locked rotor/shaft break analysis.

2.8.5.3.2.3 Results

With respect to the peak RCS pressure, peak clad temperature, and zirconium-steam reaction, the analysis demonstrated that all applicable acceptance criteria were met. The calculated sequence of events is presented in Table 2.8.5.3.2-1 for the locked rotor event. The results of the calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are summarized in Table 2.8.5.3.2-2 along with a comparison to the results from the previous analyses. Note that the previous analyses are more limiting than the EPU analyses because the previous analyses assumed an overly conservative rod drop time. This additional unnecessary conservatism has been removed from the EPU analyses. The transient results for the peak pressure/hot spot case are provided in Figures 2.8.5.3.2-1 through 2.8.5.3.2-3.

The analysis performed for the EPU demonstrated that, for the locked rotor event, the peak clad surface temperature calculated for the hot spot during the worst transient remained considerably less than 2700°F, and the amount of zirconium-water reaction was small. Under such conditions, the core remained in place and intact with no loss of core cooling capability.

The analysis also confirmed that the peak RCS pressure reached during the transient was less than 120% of the design pressure, and thereby, the integrity of the primary coolant system was demonstrated. With respect to secondary overpressurization, the loss of load event described in LR section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum, demonstrates that the secondary pressure limit is met.

The low RCS flow reactor trip function provided mitigation for a locked rotor transient such that the above criteria were satisfied. Furthermore, the results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload process.

2.8.5.3.2.4 References

1. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, D. S. Huegel, et al., April 1999.

2. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X. et al., October 1999.

3. WCAP-12910, *Pressurizer Safety Valve Set Pressure Shift*, Barrett, G. O., et al., March 1991.

Table 2.8.5.3.2-1 Time Sequence of Events – Single RCP Locked Rotor	
Event	**Time (sec)**
Rotor on One Pump Locked or the Shaft Breaks	0.0
Low Flow Reactor Trip Setpoint Reached	0.096
Rods Begin to Drop	1.096
Remaining Pump Loses Power and Begins to Coastdown	1.096
Maximum Clad Average Temperature Occurs	3.08
Maximum RCS Pressure Occurs	3.95
Time of Maximum Clad Oxidation	10.0

Table 2.8.5.3.2-2 Results for Single RCP Locked Rotor and Comparison to Previous Results			
Criteria	EPU Analysis	Previous Analysis	Limit
Maximum Clad Temperature at Core Hot Spot, °F	1924.6 (422V+) 1987.2 (OFA)	2154	2700
Maximum Zirconium-Water Reaction at Core Hot Spot, wt. %	0.53 (422V+) 0.67 (OFA)	1.26	16.0
Maximum RCS Pressure, psia	2782.02	2924	2997

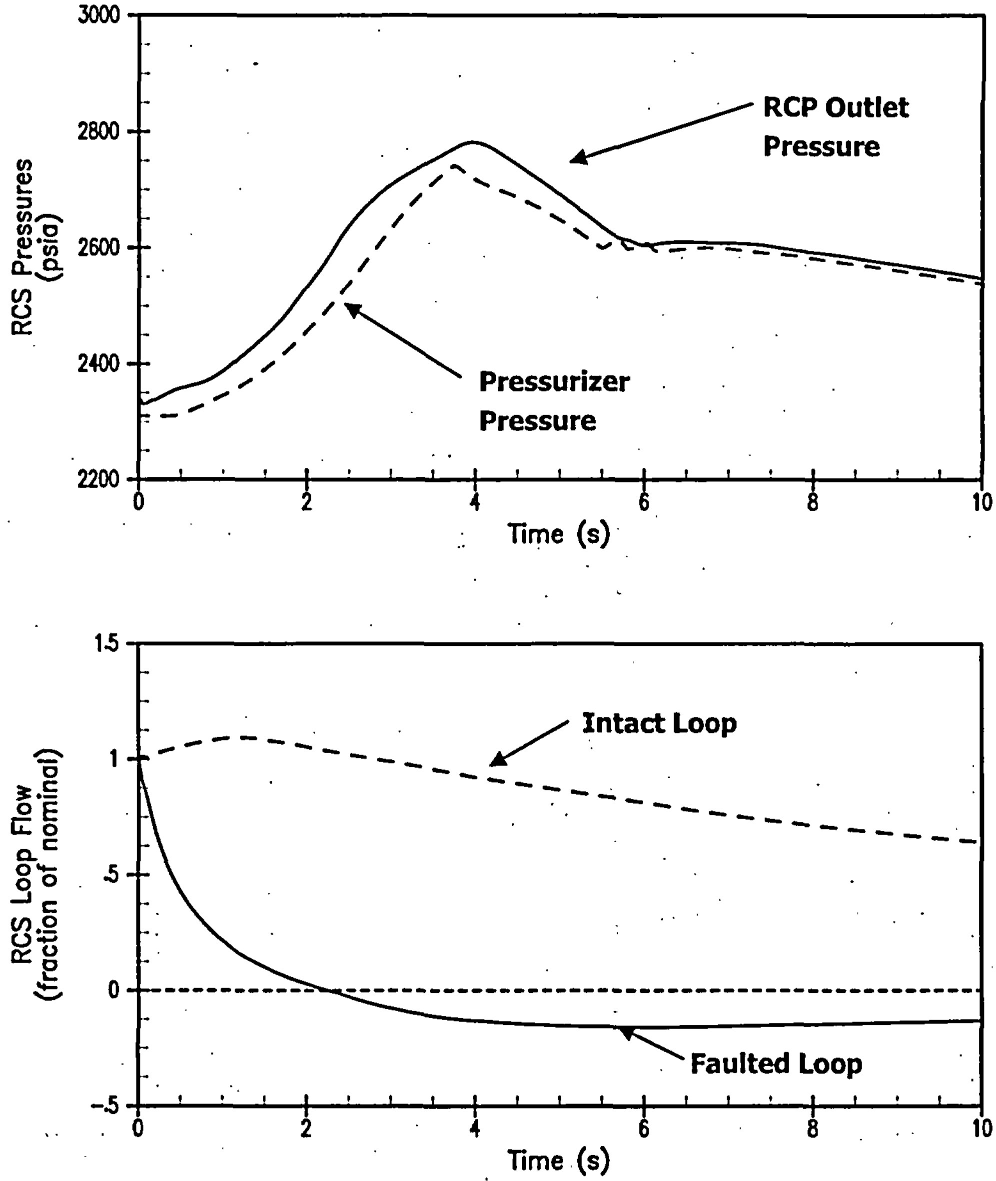


Figure 2.8.5.3.2-1
Single RCP Locked Rotor
RCS Pressures and Loop Flows vs. Time

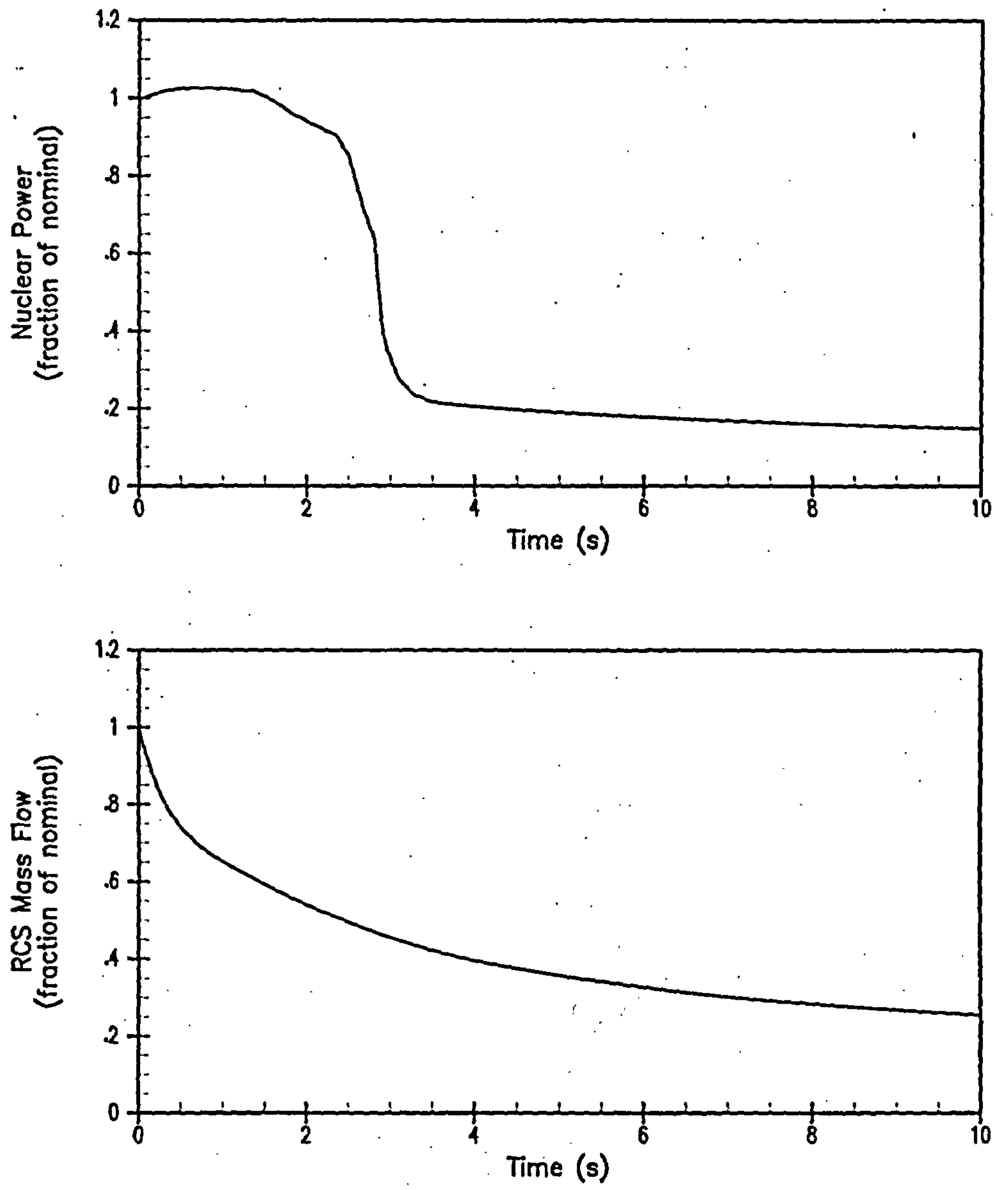


Figure 2.8.5.3.2-2
Single RCP Locked Rotor
Nuclear Power and RCS Mass Flow vs. Time

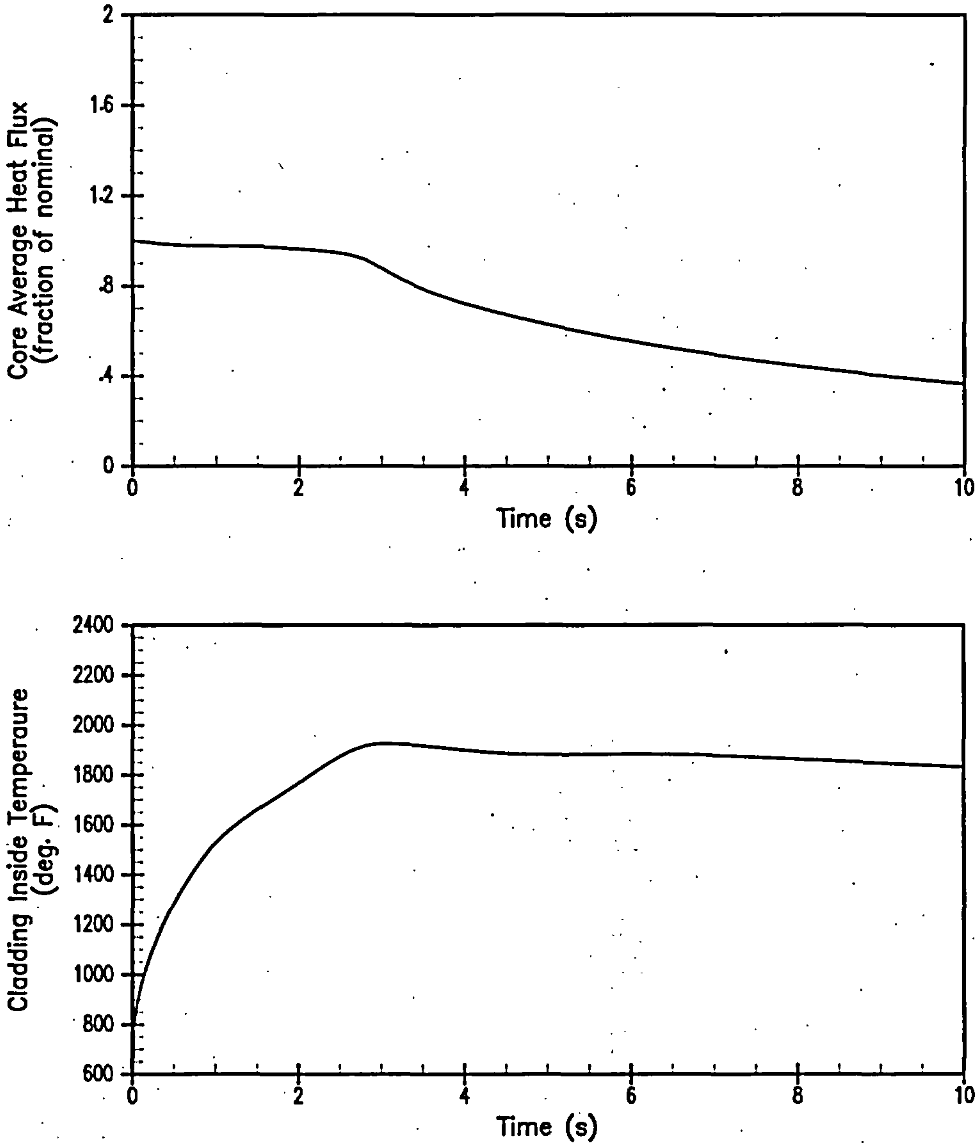


Figure 2.8.5.3.2-3
Single RCP Locked Rotor
Core Average Heat Flux and Cladding Inside Temperature vs. Time

2.8.5.3.2.5 Conclusion

The Ginna staff has reviewed the analyses of the sudden decrease in core coolant flow events and concludes that the analyses have adequately accounted for plant operation at the power level and were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the reactor coolant pressure boundary pressure limits will not be exceeded, the reactor coolant pressure boundary will behave in a non-brittle manner, the probability of propagating fracture of the reactor coolant pressure boundary is minimized, and adequate core cooling will be provided. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-27, GDC-28, and GDC-31 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical or Low- Power Startup Condition

2.8.5.4.1.1 Regulatory Evaluation

An uncontrolled rod control assembly (RCCA) withdrawal from subcritical or low-power startup conditions can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The description of the causes of the transient and the transient itself
- The initial conditions
- The values of reactor parameters used in the analysis
- The analytical methods and computer codes used
- The results of the transient analyses

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences,

- GDC-20, insofar as it requires that the reactor protection system is designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and

- GDC-25, insofar as it requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

Specific review criteria are contained in the SRP, Section 15.4.1 and other guidance provided in Matrix 8 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 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 this positive reactivity addition event relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure that the fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Fuel design and nuclear design are further discussed in LR section 2.8.1 and LR section 2.8.2, respectively. The analysis of an uncontrolled RCCA withdrawal from a subcritical condition is discussed in UFSAR section 15.4.1. The analysis of an uncontrolled RCCA withdrawal while the reactor is at power is discussed in UFSAR section 15.4.2.

- GDC-20 is described in UFSAR section 3.1.2.3.1, General Design Criterion 20 – Protection Systems Functions. As described in this UFSAR section, a plant protection system is provided to automatically initiate appropriate action whenever plant specific conditions reach pre-established limits. These limits ensure that specified fuel design limits are not exceeded when anticipated operational occurrences happen. Fuel design for EPU conditions is evaluated in LR section 2.8.2. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

- GDC-25 is described in UFSAR section 3.1.2.3.6, General Design Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions. As described in this UFSAR section, the Reactor Trip System is designed to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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 (SER) for the R.E. Ginna Nuclear Power Plant, (NUREG-1786), dated May, 2004.

2.8.5.4.1.2 Technical Evaluation

2.8.5.4.1.2.1 Introduction

An uncontrolled rod cluster control assembly (RCCA) withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control rod drive system. This could occur with the reactor either subcritical or at power. The "at power" occurrence is discussed in LR section 2.8.5.4.2. The uncontrolled RCCA withdrawal from a subcritical condition is classified as an ANS Condition II event of moderate frequency.

During startup, reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level by RCCA withdrawal or by reducing the core boron concentration. RCCA motion can cause much faster changes in reactivity than can result from changing boron concentration.

The rods are physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod banks with the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power increase results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup transient since it limits the power to an acceptable level prior to protection system action.

After the initial power increase, the nuclear power is momentarily reduced and then, if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Should a continuous RCCA withdrawal be initiated, the transient will be terminated by one of the following automatic protective functions:

a. **Source range neutron flux reactor trip** – actuated when either of two independent source range channels indicates a flux level above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed when either of the intermediate range neutron flux channels indicates a flux (P-6 permissive) above the source range cutoff power level. It is automatically reinstated when both intermediate channels indicate a flux level below the source range cutoff power level.

b. **Intermediate range neutron flux reactor trip** – actuated when either of two independent intermediate range channels indicates a flux level above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels are reading above approximately 8% power (P-10 permissive) and is automatically reinstated when three of the four channels indicate a power level below this value.

c. **Power range neutron flux reactor trip (low setting)** – actuated when two out of the four power range channels indicate a power level above approximately 25%. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 8% power (P-10 permissive). This trip function is automatically reinstated when three of the four channels indicate a power level below 8% power.

d. **Power range neutron flux reactor trip (high setting)** – actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

This analysis credits the power range neutron flux trip (low setting) to initiate the reactor trip.

2.8.5.4.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The accident analysis uses the Standard Thermal Design Procedure (STDP) methodology since the conditions resulting from the transient are outside the range of applicability of the RTDP methodology (Reference 4). To obtain conservative results for the analysis of the uncontrolled RCCA bank withdrawal from subcritical event, the following input parameters and initial conditions are modeled:

a. The magnitude of the nuclear power peak reached during the initial part of the transient, for any given reactivity insertion rate, is strongly dependent on the Doppler-only power defect. Therefore, a conservatively low absolute value is used (1100 pcm) to maximize the nuclear power transient.

b. A most-positive moderator temperature coefficient (+5 pcm/°F) is used since this yields the maximum rate of power increase. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient.

c. The analysis assumes the reactor to be at hot zero power conditions with a nominal no-load temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature (i.e., shutdown conditions). The higher initial system temperature yields a larger fuel-to-moderator heat transfer coefficient, a larger specific heat of the moderator and fuel, and a less-negative (smaller absolute magnitude) Doppler defect. The less-negative Doppler defect reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux.

d. The analysis assumes the initial effective multiplication factor (K_{eff}) to be 1.0 since it maximizes the peak neutron flux and results in the most severe nuclear power transient.

e. Reactor trip is assumed on power range high neutron flux (low setting). A conservative combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release is modeled. The analysis assumes a 10% uncertainty in the power range flux trip setpoint (low setting), raising it from the nominal value, 25%, to 35%. A delay time of 0.5 seconds is assumed for trip signal actuation and control rod assembly release. No credit is taken for the source range or intermediate range protection. During the transient, the rise in nuclear power is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods release is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

f. The maximum positive reactivity insertion rate assumed (75 pcm/sec) is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum rod withdrawal speed.

g. The DNB analysis assumes the most-limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their highest worth position.

h. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

i. The analysis assumes one of the two RCPs to be in operation. This is conservative with respect to the DNB transient.

j. This accident analysis uses the Standard Thermal Design Procedure (STDP) methodology. The use of STDP stipulates that the RCS flow rates will be based on a fraction of the thermal design flow for one RCP operating. Since the event is analyzed from hot zero power, the steady-state non-RTDP uncertainties are not considered in defining the initial conditions.

k. Two cases were analyzed to consider two different Westinghouse fuel products: one assuming 422V+ fuel and one assuming OFA fuel.

The Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Subcritical event is considered an ANS Condition II event, a fault of moderate frequency, and is analyzed to show that the core and reactor coolant system are not adversely affected by the event. This is demonstrated by showing that the DNB design basis is not violated and consequently that there is little likelihood of core damage. It must also be shown that the peak hot spot fuel centerline temperature remains within the acceptable limit (4800°F), although for this event, the heat up is relatively non-limiting for both cases analyzed (i.e., the 422V+ and OFA fuel cases).

The specific acceptance criteria applied by Ginna for this event were as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met meets the Ginna Station current licensing basis requirements with respect to GDC-10.

- Per GDC-20, the protection system should be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and to sense accident conditions and initiate the operation of important-to-safety systems and components. For this event, protection is provided via the high neutron flux reactor trip.

- GDC-25 requires that the protection system 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. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-25.

2.8.5.4.1.2.3 Description of Analyses and Evaluations

The analysis of the uncontrolled RCCA bank withdrawal from subcritical conditions is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE (reference 1), is used to calculate the core average nuclear power transient, including the various core feedback effects (i.e., Doppler and moderator reactivity). Next, the FACTRAN computer code (reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the core average heat flux and hot spot fuel temperature transients. Finally, the core average heat flux calculated by FACTRAN is used in the VIPRE computer code (reference 3) for transient DNBR calculations.

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.1.3 Results

The analysis shows that all applicable acceptance criteria are met. The minimum DNBR never goes below the limit value and the peak fuel centerline temperature is 2108°F for 422V+ fuel and 2305°F for OFA fuel. The peak temperatures for both cases are well below the minimum temperature where fuel melting would be expected (4800°F).

Figure 2.8.5.4.1-1 shows the nuclear power and core average heat flux transients and Figure 2.8.5.4.1-2 shows the inner clad and fuel average temperature transient at the hot spot for the 422V+ fuel case. Figure 2.8.5.4.1-3 shows the nuclear power and core average heat flux transients and Figure 2.8.5.4.1-4 shows the inner clad and fuel average temperature transient at the hot spot for the OFA fuel case.

The time sequence of events for both cases is presented in Table 2.8.5.4.1-1. Numerical results of the EPU analysis are shown in Table 2.8.5.4.1-2. Note that a comparison to the results from the previous analysis is not shown because the previous licensing basis documentation does not include numerical results – just the conclusion that the DNB design basis is met.

In the event of an RCCA withdrawal event from subcritical conditions, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature results in a minimum DNBR greater than the safety analysis limit values (see Table 2.8.5.4.1-2). Furthermore, since the maximum fuel temperatures predicted to occur during this event are much less than those required for fuel melting to occur, no fuel damage is predicted as a result of this transient. Clad damage is also precluded.

2.8.5.4.1.4 References

1. WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Nonproprietary), *TWINKLE, a Multi-dimensional Neutron Kinetics Computer Code*,. Barry, R. F., Jr. and Risher, D. H.

2. WCAP-7908, *FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod*, Hargrove, H. G., December 1989.

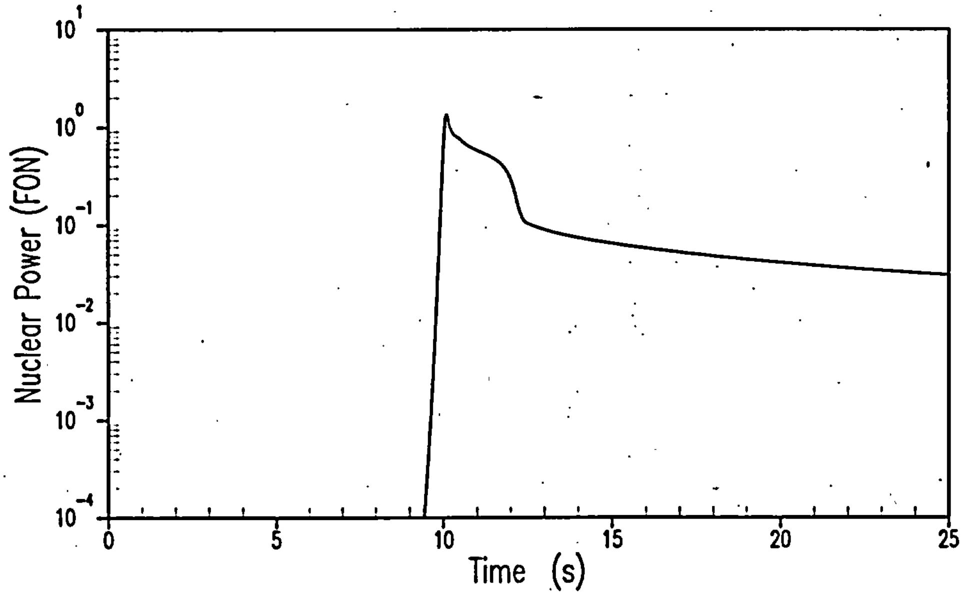
3. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X. et al., October 1999.

4. WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), *Revised Thermal Design Procedure*, April 1989.

Table 2.8.5.4.1-1 Time Sequence of Events – Uncontrolled RCCA Withdrawal from a Subcritical Condition		
Event	**Time (sec)**	
	422V+	**OFA**
Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0	0.0
Power range high-neutron flux (low setting of 0.35) setpoint reached	9.98	9.98
Peak nuclear power occurs	10.11	10.11
Rods begin to fall into the core	10.48	10.48
Peak heat flux (avg. channel) occurs / Minimum DNBR occurs	11.72	11.78
Peak clad temperature occurs	11.98	11.94
Peak average fuel temperature occurs	12.18	12.14
Peak fuel centerline temperature occurs	13.68	13.44

Table 2.8.5.4.1-2 Uncontrolled RCCA Withdrawal from a Subcritical Condition Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Minimum DNBR *below* first mixing vane grid	1.987	N/A *	1.447 (EPU)
Minimum DNBR *above* first mixing vane grid	1.951	N/A *	1.302 (EPU)

* The previous licensing basis analysis concludes that the DNB Design Basis is met but does not contain numerical results.



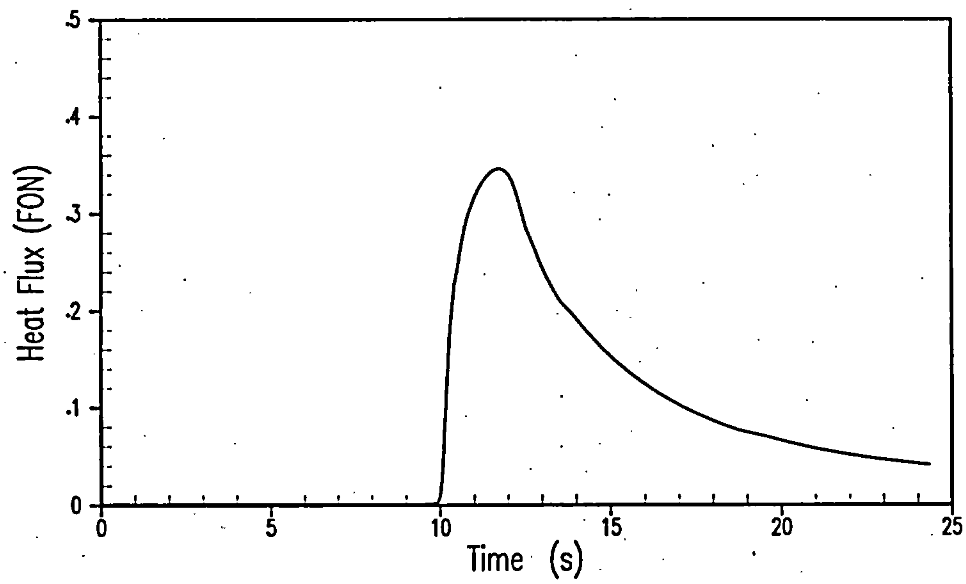


Figure 2.8.5.4.1-1
Rod Withdrawal from Subcritical: 422V+ Fuel
Nuclear Power and Core Average Heat Flux versus Time

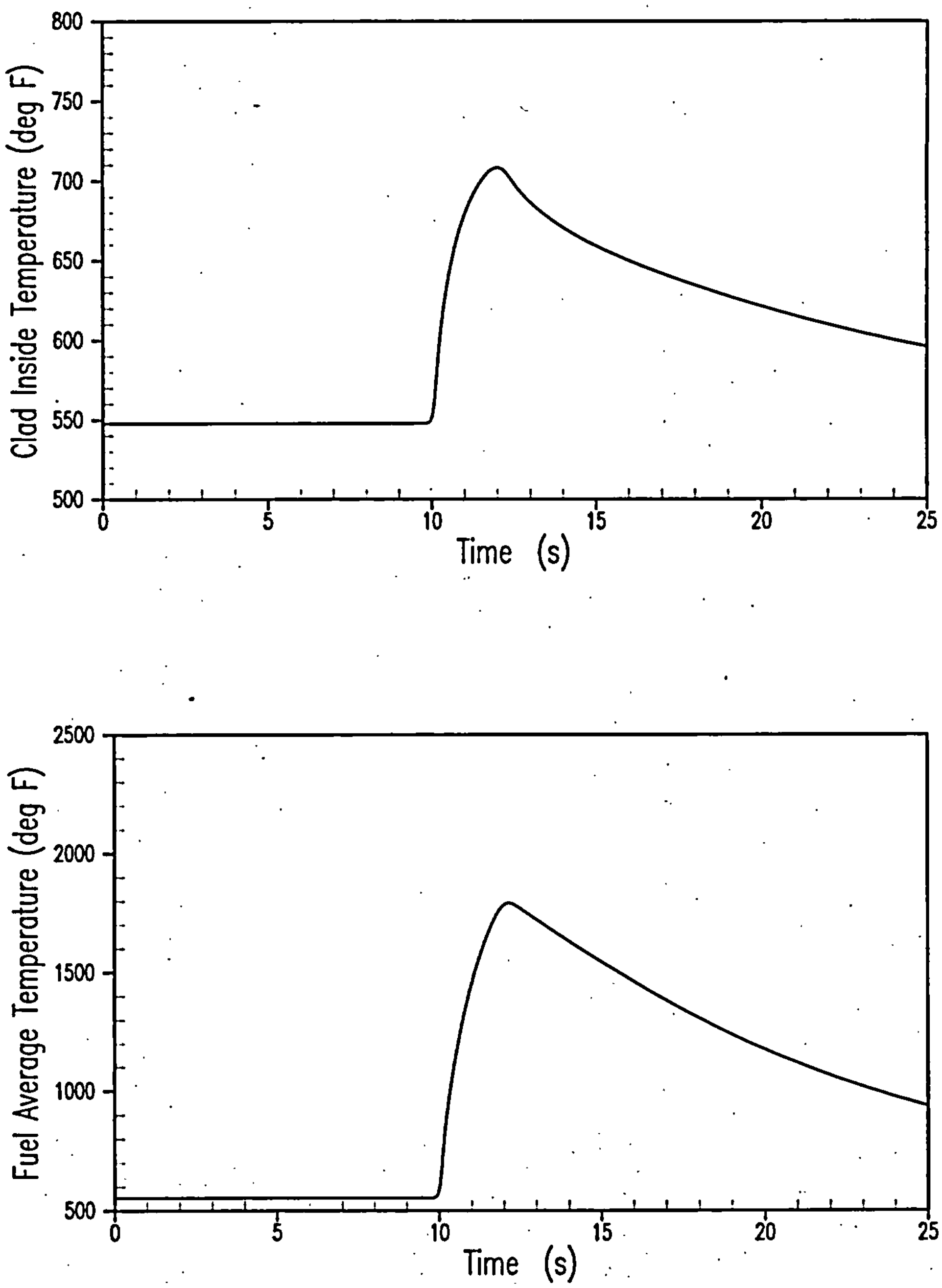
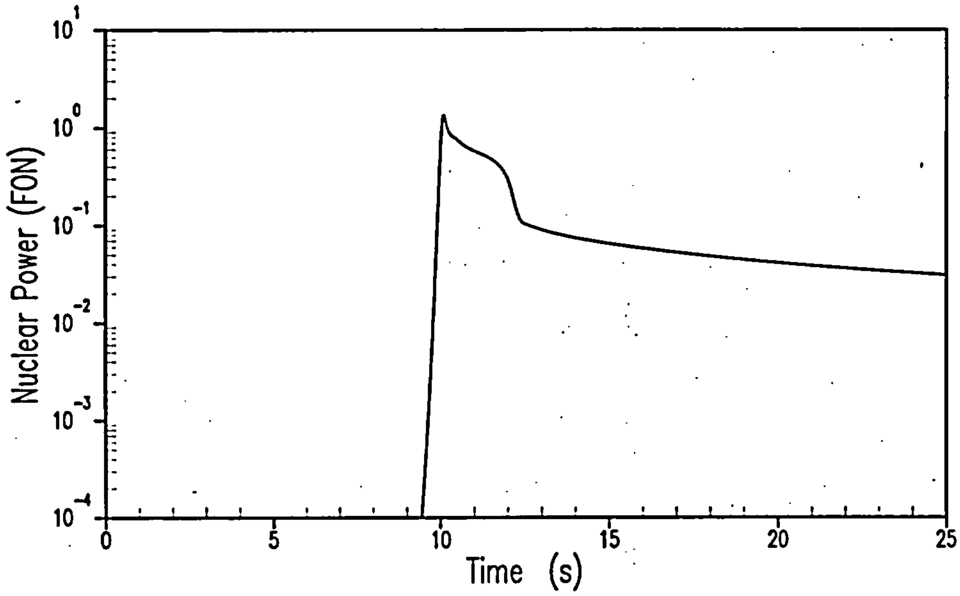


Figure 2.8.5.4.1-2
Rod Withdrawal from Subcritical: 422V+ Fuel
Hot Spot Temperatures versus Time



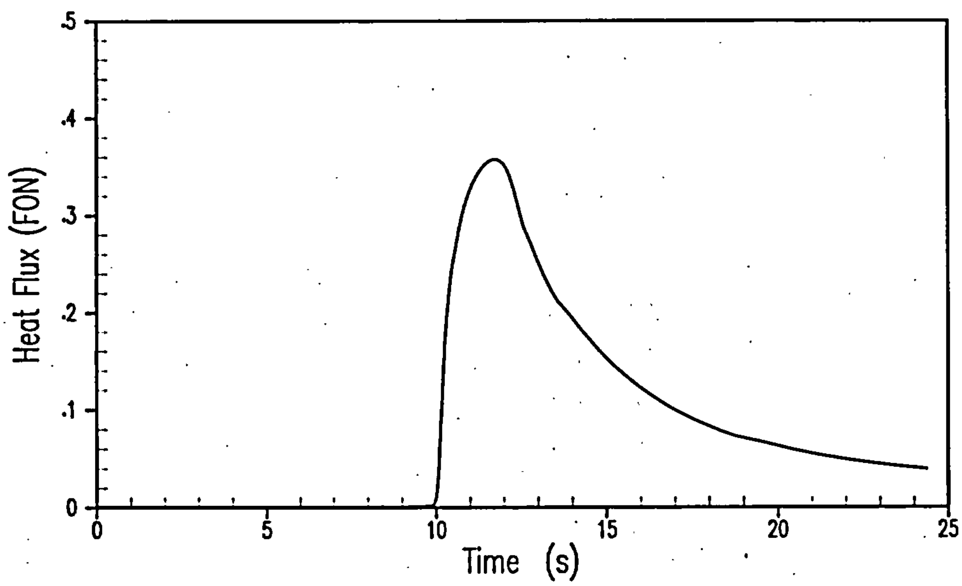


Figure 2.8.5.4.1-3
Rod Withdrawal from Subcritical: OFA Fuel
Nuclear Power and Core Average Heat Flux versus Time

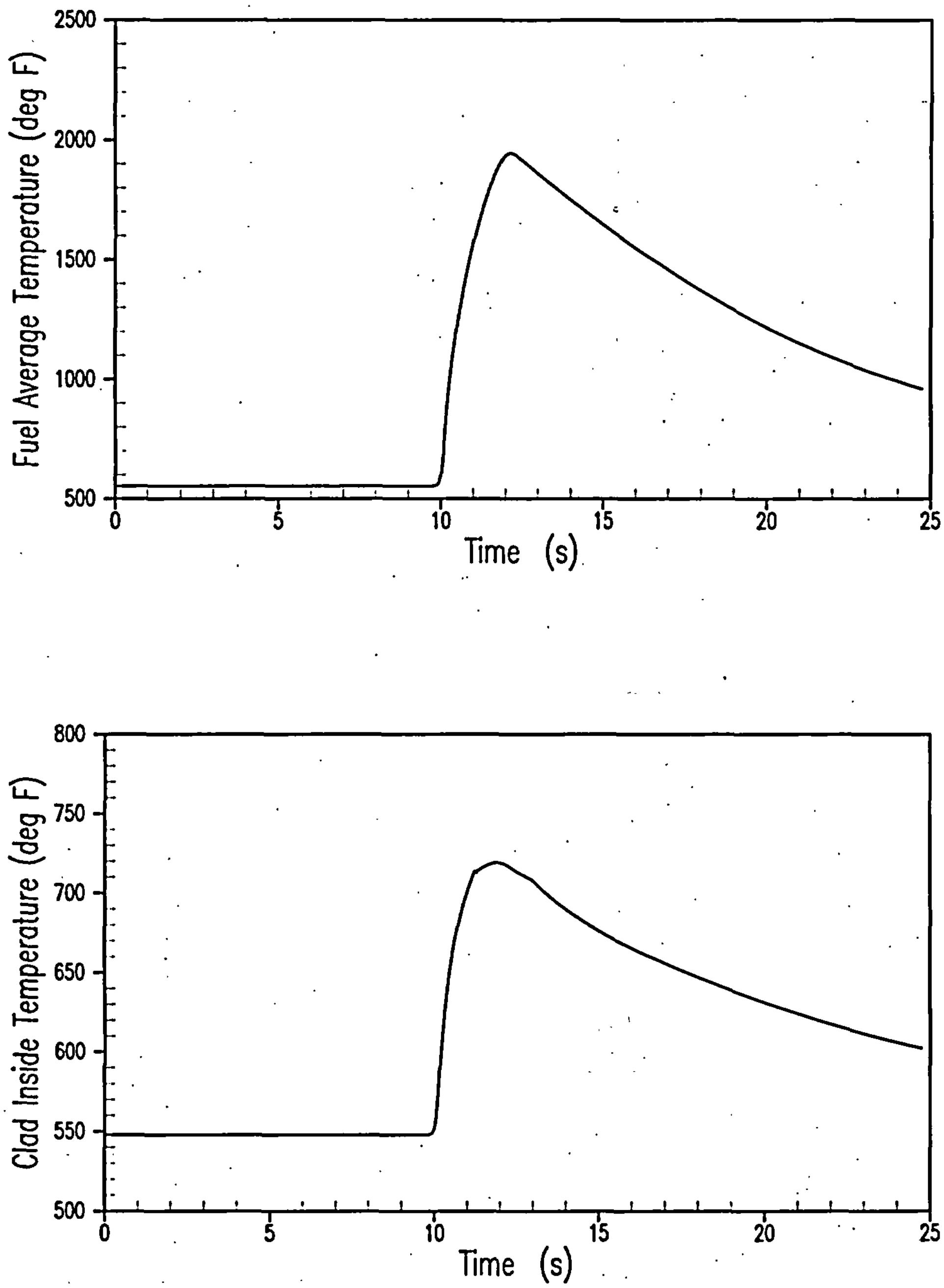


Figure 2.8.5.4.1-4
Rod Withdrawal from Subcritical: OFA Fuel
Hot Spot Temperatures versus Time

2.8.5.4.1.5 Conclusion

The Ginna staff has reviewed the analyses of the uncontrolled RCCA withdrawal from a subcritical or low-power startup condition and concludes that the analyses have adequately accounted for the changes in core design necessary for plant operation at the proposed power level. The Ginna staff also concludes that the analyses were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are not exceeded. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-20, and GDC-25 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the uncontrolled RCCA withdrawal from a subcritical or low-power startup condition.

2.8.5.4.2 Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

2.8.5.4.2.1 Regulatory Evaluation

An uncontrolled rod cluster control assembly (RCCA) withdrawal at power can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The description of the causes of the anticipated operational occurrence and the description of the event itself
- The initial conditions
- The values of reactor parameters used in the analysis
- The analytical methods and computer codes used
- The results of the associated analyses

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences
- GDC-20, insofar as it requires that the reactor protection system is designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences
- GDC-25, insofar as it requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems

Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 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 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 this positive reactivity addition event relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure that the fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Fuel design and nuclear design are further discussed in LR section 2.8.1 and LR section 2.8.2, respectively.

- GDC-20 is described in UFSAR section 3.1.2.3.1, General Design Criterion 20 – Protection Systems Functions. As described in this UFSAR section, a plant protection system is provided to automatically initiate appropriate action whenever plant specific conditions reach pre-established limits. These limits ensure that specified fuel design limits are not exceeded when anticipated operational occurrences happen. Fuel design for EPU conditions is evaluated in LR section 2.8.2. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

- GDC-25 is described in UFSAR section 3.1.2.3.6, General Design Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions. As described in this UFSAR section, the Reactor Trip System is designed to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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.

2.8.5.4.2.2 Technical Evaluation

2.8.5.4.2.2.1 Introduction

An uncontrolled RCCA withdrawal at power that causes an increase in core heat flux can result from incorrect operator action or a malfunction in the rod control system. Immediately following the initiation of the accident, the steam generator heat removal rate lags behind the core power generation rate until the steam generator pressure reaches the setpoint of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNBR limit and/or fuel centerline melt. Therefore, to avoid core damage; the reactor protection system is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value, or the fuel rod linear heat generation rate (kW/ft) limit is exceeded.

The automatic features of the reactor protection system that prevent core damage in an RCCA bank withdrawal incident at power include the following:

- Power range high neutron flux instrumentation actuates a reactor trip on neutron flux if two-out-of-four channels exceed an overpower setpoint.

- Reactor trip actuates if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature, and coolant average pressure to protect against violating the DNBR limit.

- A high-pressurizer pressure reactor trip actuates if any two-out-of-three pressure channels exceed a fixed setpoint.

- A high-pressurizer water level reactor trip actuates if any two-out-of-three level channels exceed a fixed setpoint.

- Main steam safety valves (MSSVs) can open for this event and provide an additional heat sink.

Additional protection is provided by the overpower ΔT signal that is not credited in this analysis.

2.8.5.4.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

A number of cases were analyzed assuming a range of reactivity insertion rates for both minimum and maximum reactivity feedback conditions at various power levels. The cases presented below are representative for this event.

For an uncontrolled RCCA bank withdrawal at power accident, the DNB case analysis assumed the following conservative assumptions:

- This accident was analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 1). Initial reactor power, RCS pressure, and RCS temperature were assumed to be at their nominal values, adjusted to account for any applicable measurement biases, consistent with steady-state full-power operation. Minimum measured flow was modeled. Uncertainties in initial conditions were included in the DNBR limit as described in the RTDP.

- For reactivity coefficients, two cases were analyzed.

– Minimum reactivity feedback: A moderator temperature coefficient (MTC) of +5 pcm/°F is used for power levels less than 70% consistent with the Ginna Technical Specifications. A moderator temperature coefficient of 0 pcm/°F is used for the 100% power cases. A least-negative Doppler-only power coefficient formed the basis for the beginning-of-life (BOL) minimum reactivity feedback assumption.

– Maximum reactivity feedback: A conservatively large, positive moderator density coefficient of 0.45 Δk/g/cc (corresponding to a large negative MTC) and a most-negative Doppler-only power coefficient formed the basis for the end-of-life (EOL) maximum reactivity feedback assumption.

- The reactor trip on high neutron flux was assumed to be actuated at a conservative value of 115% of nominal full power. The ΔT trips included all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation were assumed at their maximum values.

- The RCCA trip insertion characteristic was based on the assumption that the highest-worth RCCA was stuck in its fully withdrawn position.

- A range of reactivity insertion rates was examined. The maximum-positive reactivity insertion rate was greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at a conservative speed (48.125 inches/minute, which corresponds to 77 steps/minute).

- Power levels of 10, 60, and 100% of the NSSS power of 1817 MWt were considered.

For an uncontrolled RCCA bank withdrawal at power accident, the RCS pressure case analysis assumes the following conservative assumptions:

- The Revised Thermal Design Procedure is not required. The initial NSSS power is conservatively set to 8% of the nominal power level (10% indicated power minus 2% uncertainty). Minimum (2190 psia) and maximum (2310 psia) reactor coolant system pressures are analyzed. The initial reactor coolant average temperature is set to a value (553.9°F) corresponding to the indicated power level. Thermal Design Flow is modeled.

- For reactivity coefficients, minimum feedback cases are analyzed.

  A moderator temperature coefficient of +5 pcm/°F is assumed, consistent with the initial power level, in conjunction with a least-negative Doppler-only power coefficient to form the basis for the beginning-of-life (BOL) minimum reactivity feedback assumption.

- The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 115% of nominal full power. The reactor trip on high pressurizer pressure is assumed to be actuated at a conservative value of 2425 psia.

- The RCCA trip insertion characteristic is based on the assumption that the highest-worth rod cluster control assembly is stuck in its fully withdrawn position.

- A range of reactivity insertion rates are examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at a conservative speed (48.125 inches/minute, which corresponds to 77 steps/minute).

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at-power accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the main acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

- Pressure in the RCS and main steam system (MSS) should be maintained below 110% of the design pressures.

The protection features presented in LR section 2.8.5.4.2.2.1 provide mitigation of the uncontrolled RCCA bank withdrawal at-power transient such that the above criteria are satisfied.

The specific acceptance criteria applied by Ginna for this event are as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met satisfies the Ginna Station current licensing basis requirements with respect to GDC-10.

- Per GDC-20, the protection system should be designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and to initiate the operation of systems and components important to safety. For this event, protection is provided via the high neutron flux reactor trip and the overtemperature ΔT trip.

- GDC-25 requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental control rod withdrawal (not ejection or dropout). Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-25.

2.8.5.4.2.2.3 Description of Analyses and Evaluations

The purpose of this analysis was to demonstrate the manner in which the protection functions described above actuate for various combinations of reactivity insertion rates and initial conditions. Insertion rate and initial conditions determined which trip function actuated first.

The uncontrolled rod withdrawal at-power event was analyzed with the RETRAN computer code (reference 2). The program simulated the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves (MSSVs). The program computed pertinent plant variables including temperatures, pressures, power level, and DNBR.

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.2.3 Results

DNB Case

Figures 2.8.5.4.2-1 through 2.8.5.4.2-3 show the transient response for a rapid uncontrolled RCCA bank withdrawal incident (100 pcm/sec) starting from 100% power with minimum feedback. Reactor trip on high neutron flux occurred shortly after the start of the accident. Because of the rapid reactor trip, small changes in T_{avg} and pressure resulted in the margin to the DNBR limit being maintained.

The transient response for a slow uncontrolled RCCA bank withdrawal (5 pcm/sec) from 100% power with minimum feedback is shown in Figures 2.8.5.4.2-4 through 2.8.5.4.2-6. Reactor trip on overtemperature ΔT occurred after a longer period of time, and the rise in temperature was consequently larger than for a rapid RCCA bank withdrawal. Again, the minimum DNBR was greater than the safety analysis limit value.

Figure 2.8.5.4.2-7 shows the minimum DNBR as a function of reactivity insertion rate from 100% power for both minimum and maximum reactivity feedback conditions. It can be seen that the high neutron flux and overtemperature ΔT reactor trip functions provided DNB protection over the range of reactivity insertion rates. The minimum DNBR was never less than the safety analysis limit value.

The RCS and MSS were maintained below 110% of the design pressures by restricting the maximum reactivity insertion rate to less than or equal to 55 pcm/sec. This maximum reactivity insertion rate will become a reload limit which will be reconfirmed for each reload. Note that this is currently being done for each Ginna reload and will continue for the EPU.

Figures 2.8.5.4.2-8 and 2.8.5.4.2-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60 and 10% power, respectively. The results were similar to the 100% power case; however, as the initial power level decreased, the range over which the overtemperature ΔT trip is effective was increased. In all of these cases, the DNBR remained above the safety analysis limit value (1.38).

A calculated sequence of events for two cases is shown in Table 2.8.5.4.2-1. With the reactor tripped, the plant eventually returned to a stable condition. The plant could subsequently be cooled down further by following normal plant shutdown procedures. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.4.2-2. In all cases, the EPU analyses incorporate appropriate conservatisms and are more limiting than the previous analyses, but no limit is reached.

RCS Pressure Case

The plant response to the RCCA withdrawal incident, which is limiting (insertion rate of 55 pcm/sec) with respect to RCS pressure concerns, is shown in Figures 2.8.5.4.2-10 and 2.8.5.4.2-11. Reactor trip on high pressurizer pressure occurs shortly after the start of the accident. After reactor trip, the pressure transient turns around prior to reaching the safety analysis limit.

The high neutron flux and overtemperature ΔT reactor trip functions provided adequate protection over the entire range of possible reactivity insertion rates (i.e., the minimum value of DNBR was always larger than the safety analysis limit value of 1.38). The RCS and MSS were maintained below 110% of the design pressures of 2748.5 psia and 1208.5 psia, respectively. Therefore, the results of the analysis (see Table 2.8.5.4.2-2) show that an uncontrolled RCCA withdrawal at power does not adversely affect the core, the RCS, or the MSS and all applicable criteria were met.

2.8.5.4.2.4 References

1. *WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), Revised Thermal Design Procedure, April 1989.*

2. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, April 1999.

Table 2.8.5.4.2-1 Time Sequence of Events – Uncontrolled RCCA Bank Withdrawal at Power		
Case	**Event**	**Time (sec)**
100% Power, Maximum Feedback, Rapid RCCA Withdrawal (100 pcm/sec)	Initiation of Uncontrolled RCCA Withdrawal	0.0
	Power Range High Neutron Flux – High Setpoint Reached	3.6
	Rods Begin to Fall	4.1
	Minimum DNBR Occurs	4.3
100% Power, Minimum Feedback, Slow RCCA Withdrawal (5 pcm/sec)	Initiation of Uncontrolled RCCA Withdrawal	0.0
	Power Range High Neutron Flux – High Setpoint Reached	26.6
	Rods Begin to Fall	27.1
	Minimum DNBR Occurs	27.5
8% Power, RCS Pressure Case, Minimum Feedback, Limiting RCCA Withdrawal (55 pcm/sec)	Initiation of Uncontrolled RCCA Withdrawal	0.0
	High Pressurizer Pressure Setpoint Reached	13.3
	Rods Begin to Fall	15.3
	Maximum RCS Pressure Occurs	16.7

Table 2.8.5.4.2-2 Uncontrolled RCCA Bank Withdrawal at Power – Results and Comparison to Previous Results			
	EPU Analysis	**Previous Analysis**	**Limit**
Minimum DNBR	1.384	1.727 *	1.38 (EPU)
Peak Primary System Pressure (psia)	2748.1 **	2743 **	2748.5
Peak Secondary System Pressure	1207.7	1127	1208.5

* Based on Exxon fuel and compared to a limit of 1.62. The transition to Westinghouse OFA fuel was qualitatively evaluated. No numerical results are available.

** The maximum reactivity insertion rate is limited to 55 pcm/sec. This is confirmed on a reload specific basis and will continue to be confirmed after the uprating.

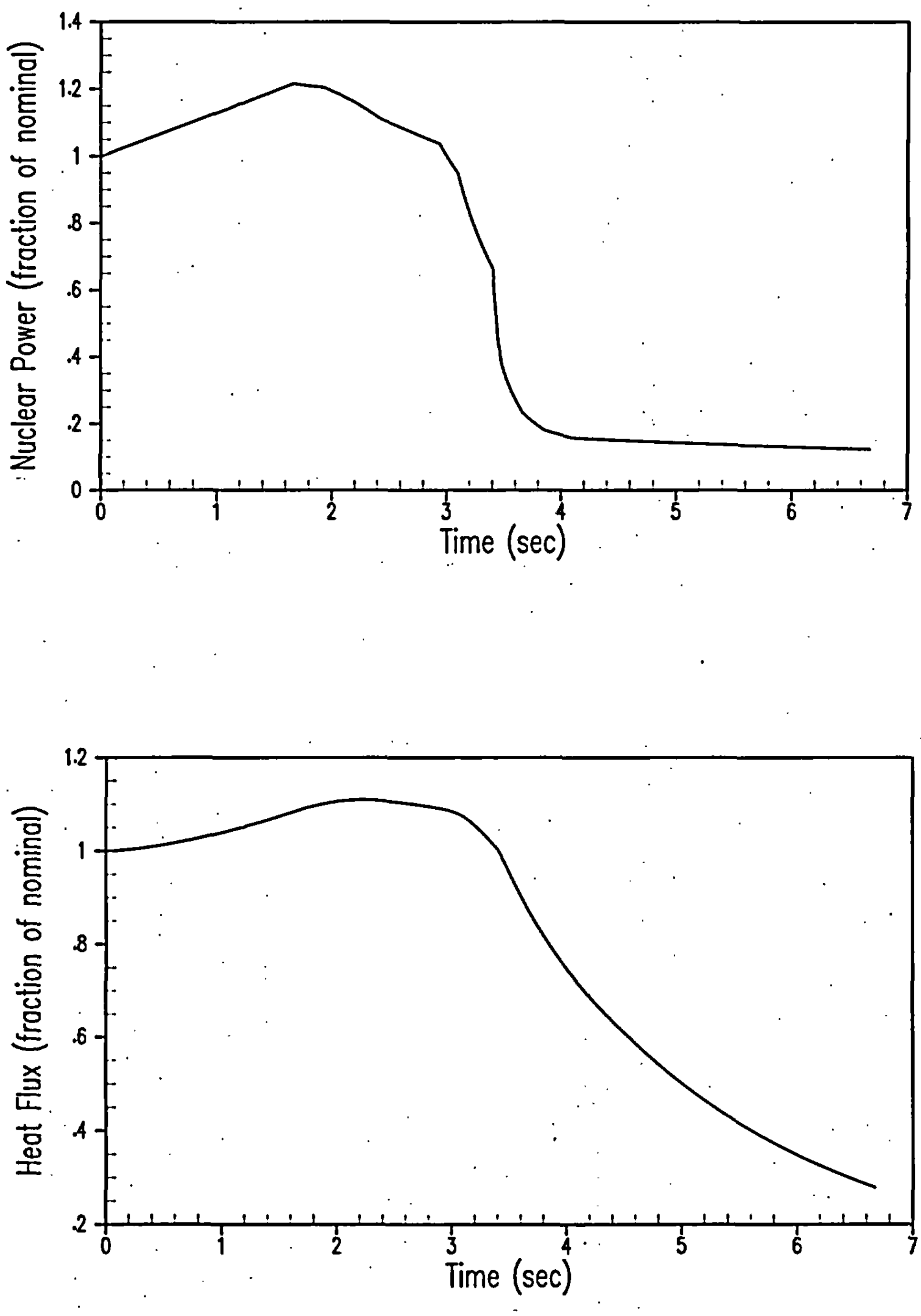
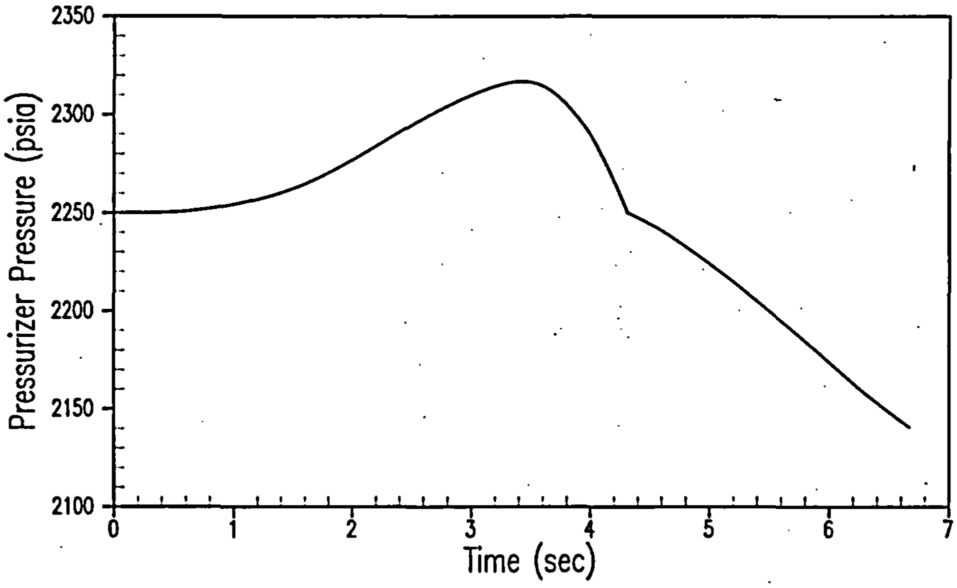


Figure 2.8.5.4.2-1
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 100 pcm/sec
Nuclear Power and Heat Flux vs. Time



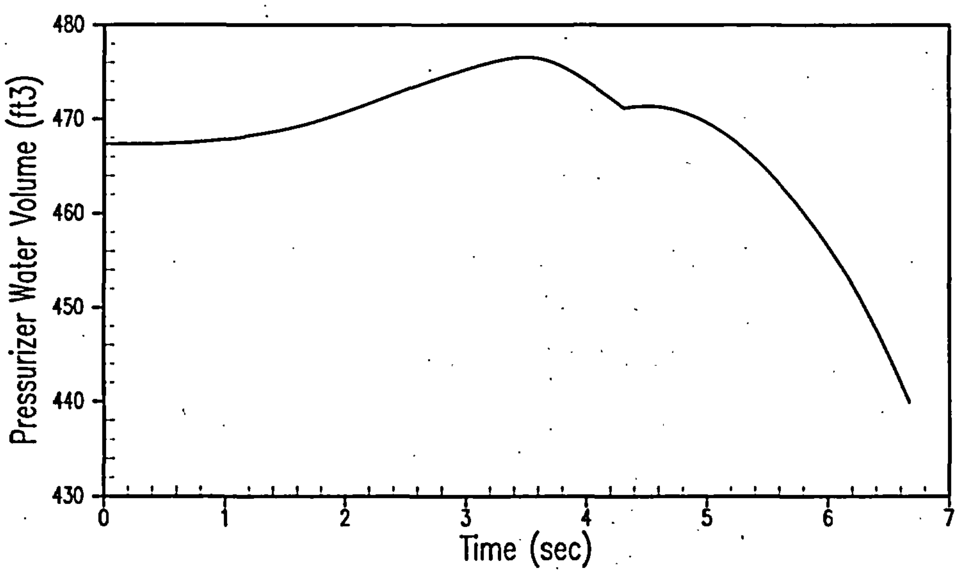


Figure 2.8.5.4.2-2
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 100 pcm/sec
Pressurizer Pressure and Water Volume vs. Time

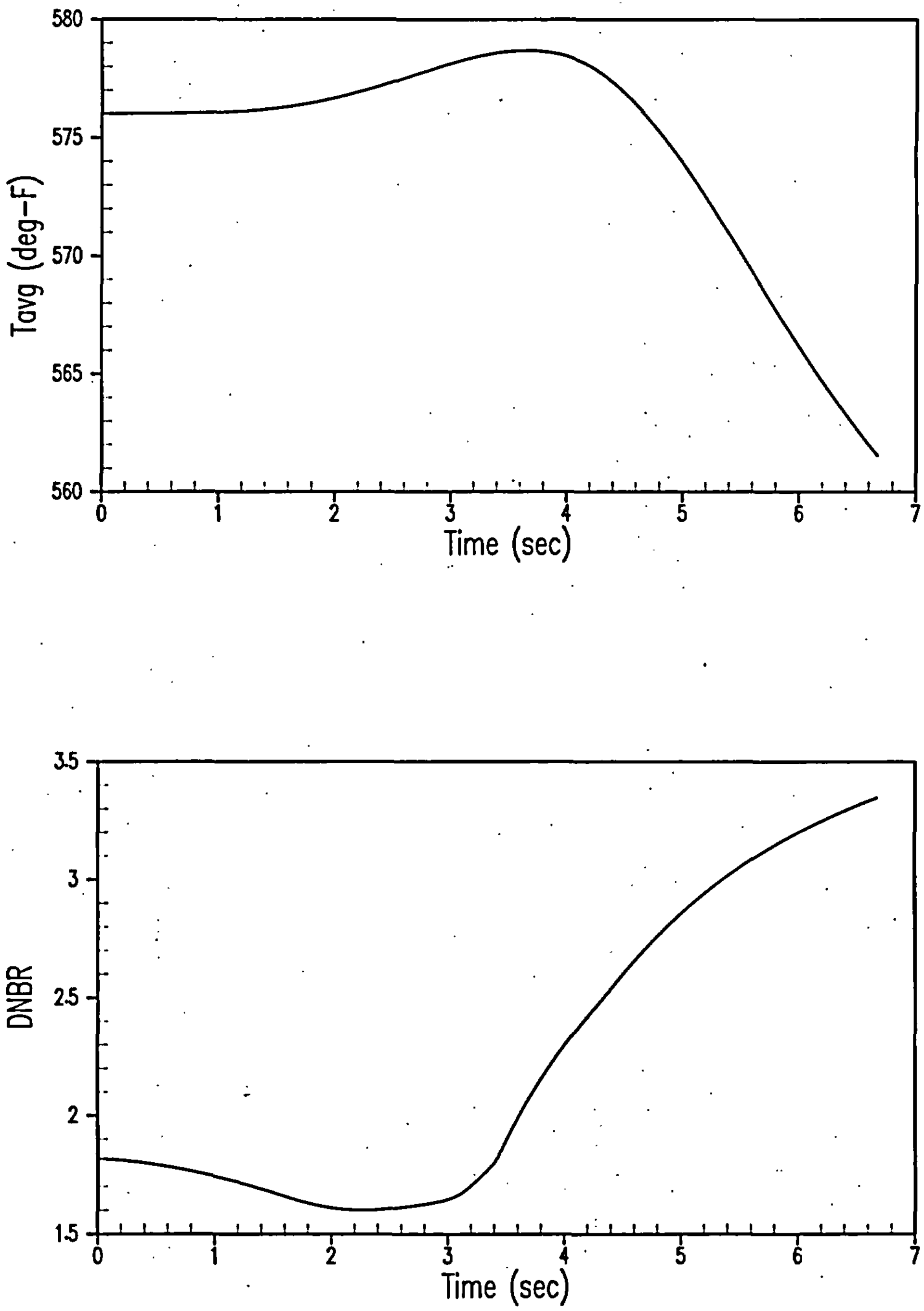
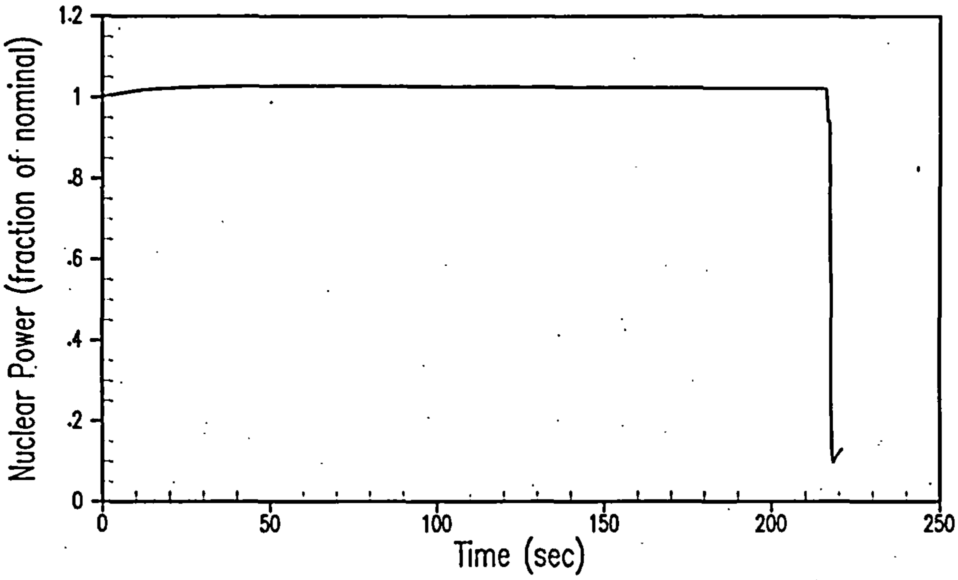


Figure 2.8.5.4.2-3
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 100 pcm/sec
Core Average Temperature and DNBR vs. Time



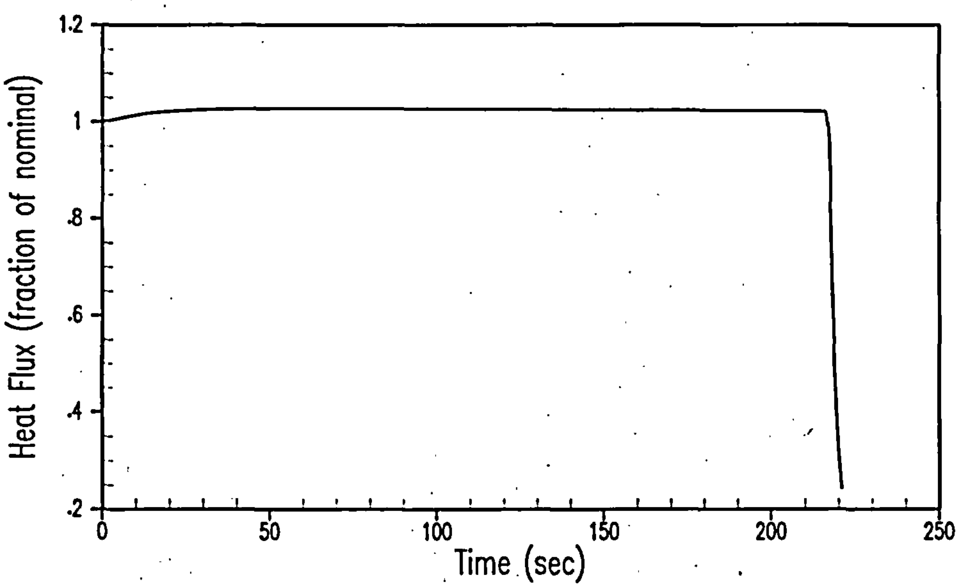


Figure 2.8.5.4.2-4
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 5 pcm/sec
Nuclear Power and Heat Flux vs. Time

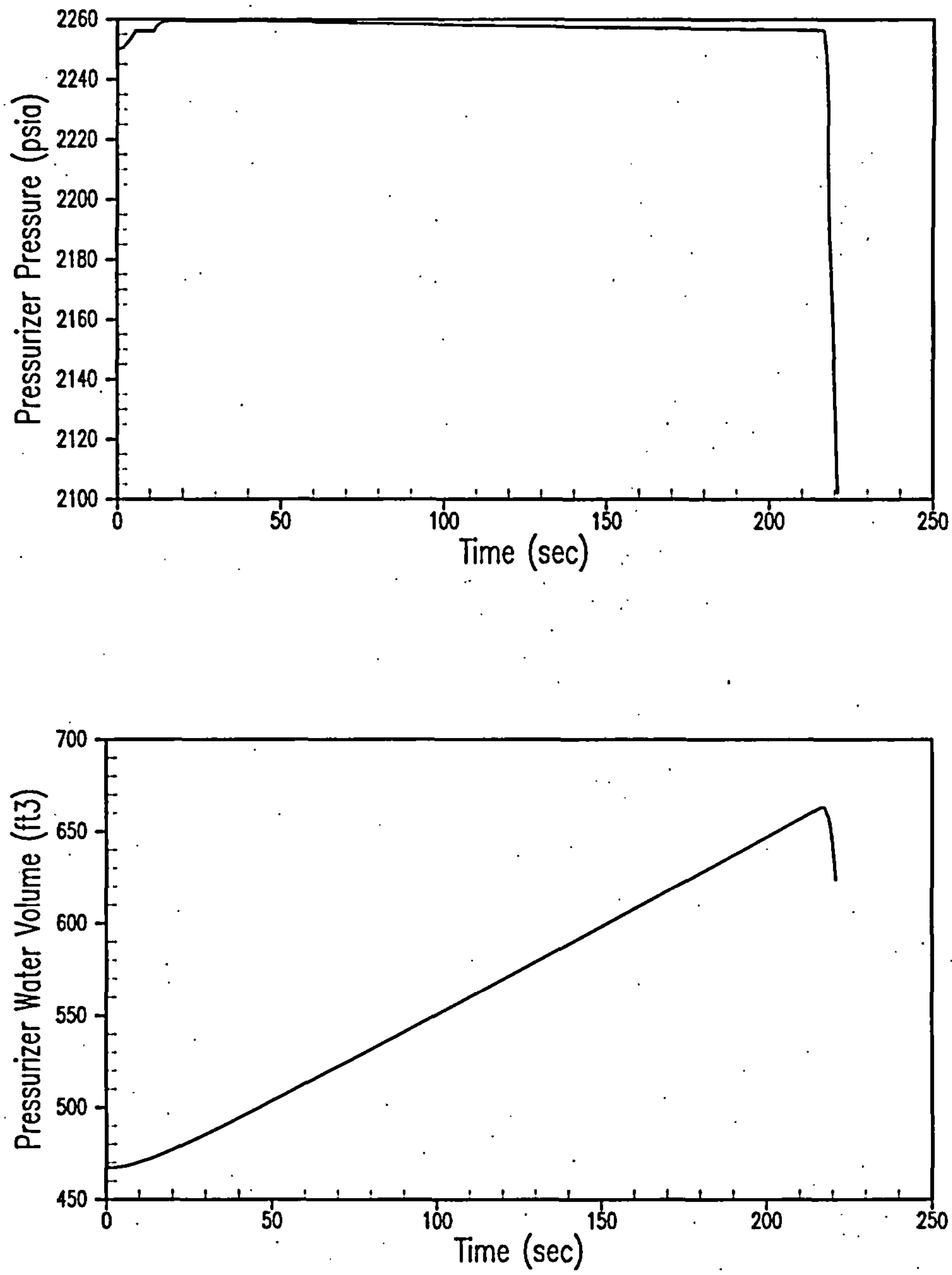
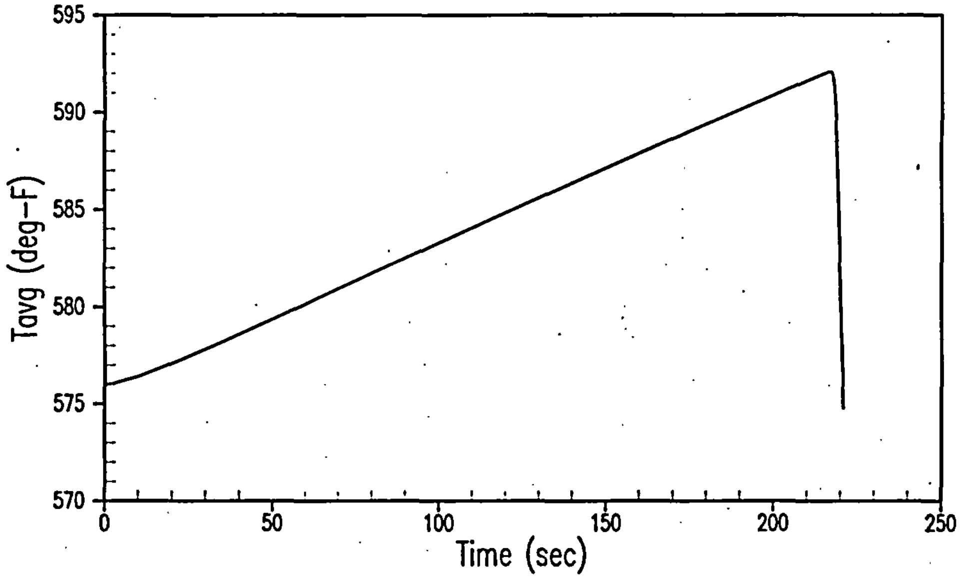


Figure 2.8.5.4.2-5
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 5 pcm/sec
Pressurizer Pressure and Water Volume vs. Time



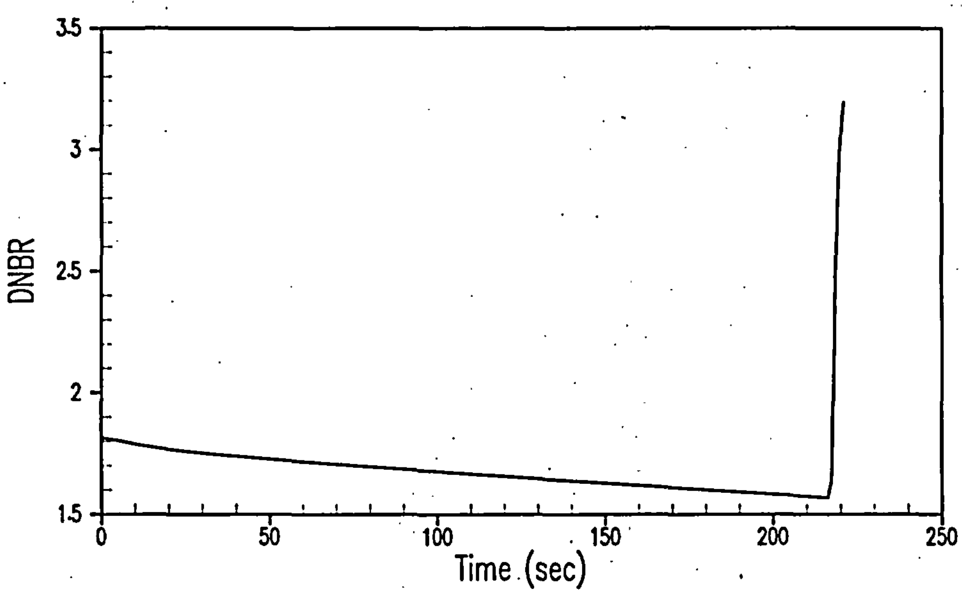


Figure 2.8.5.4.2-6
Rod Withdrawal at Power – DNB Case
Minimum Reactivity Feedback – 100% Power - 5 pcm/sec
Core Average Temperature and DNBR vs. Time

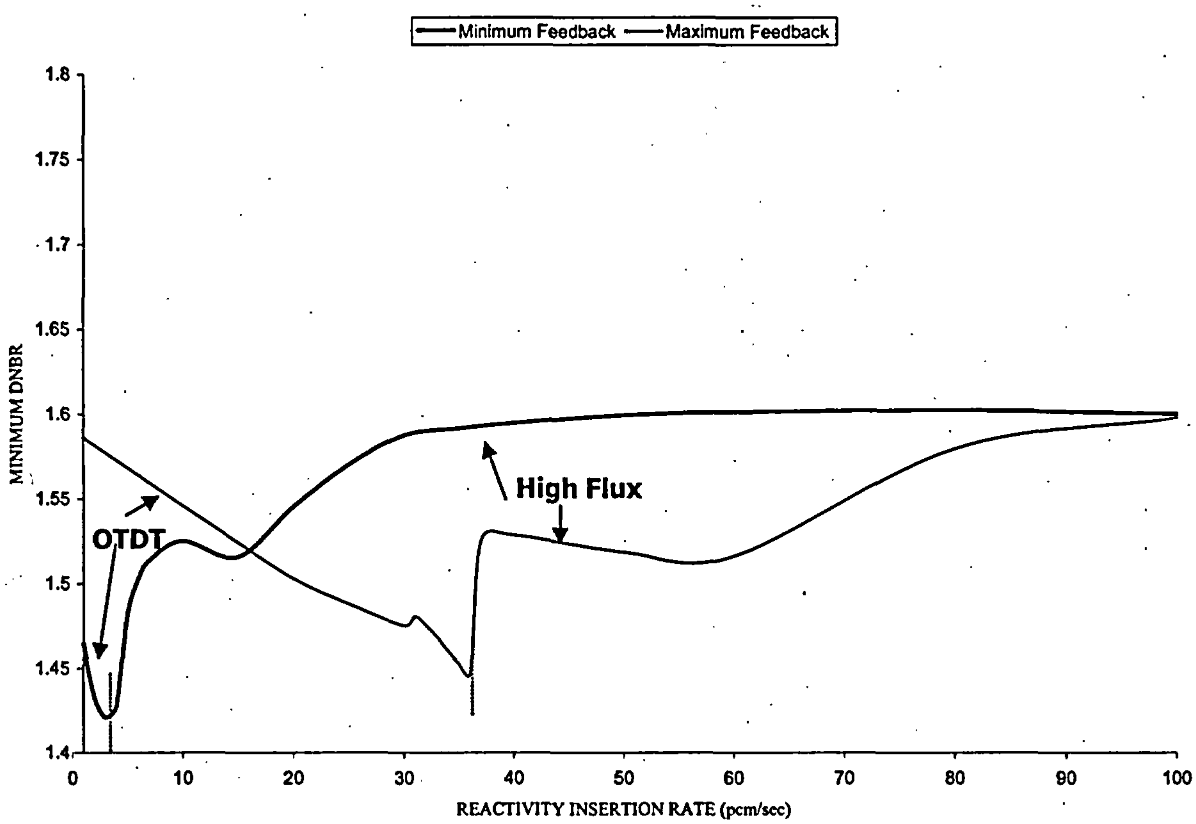


Figure 2.8.5.4.2-7
Rod Withdrawal at Power – DNB Case
100% Power
Minimum DNBR vs. Reactivity Insertion Rate

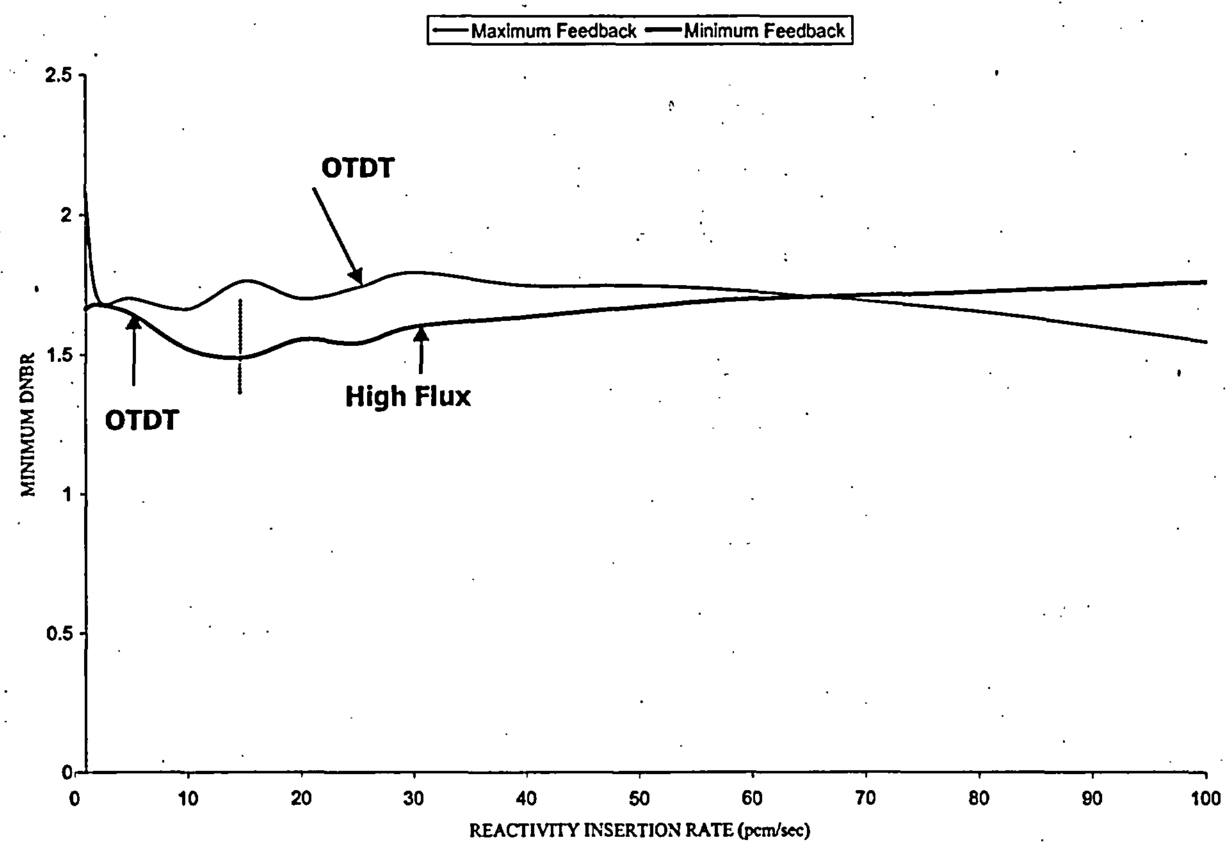


Figure 2.8.5.4.2-8
Rod Withdrawal at Power – DNB Case
60% Power
Minimum DNBR vs. Reactivity Insertion Rate

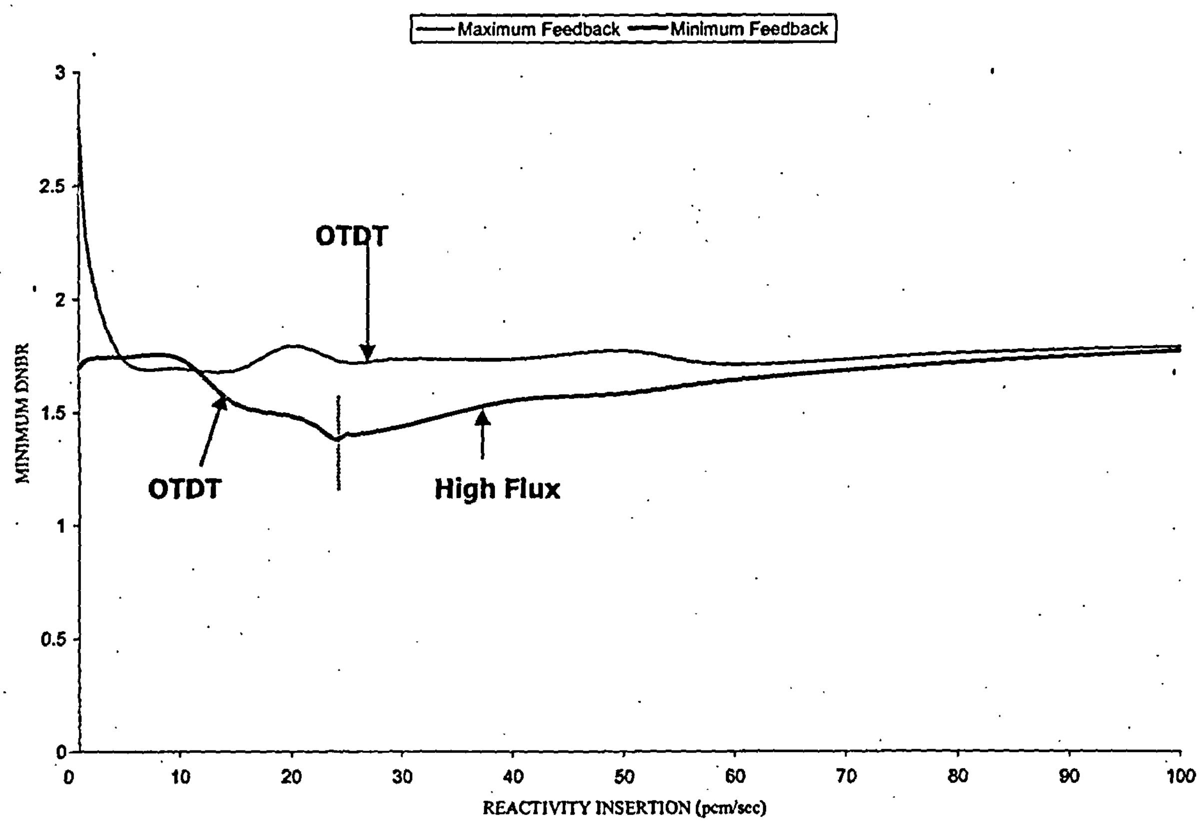


Figure 2.8.5.4.2-9
Rod Withdrawal at Power
10% Power
Minimum DNBR vs. Reactivity Insertion Rate

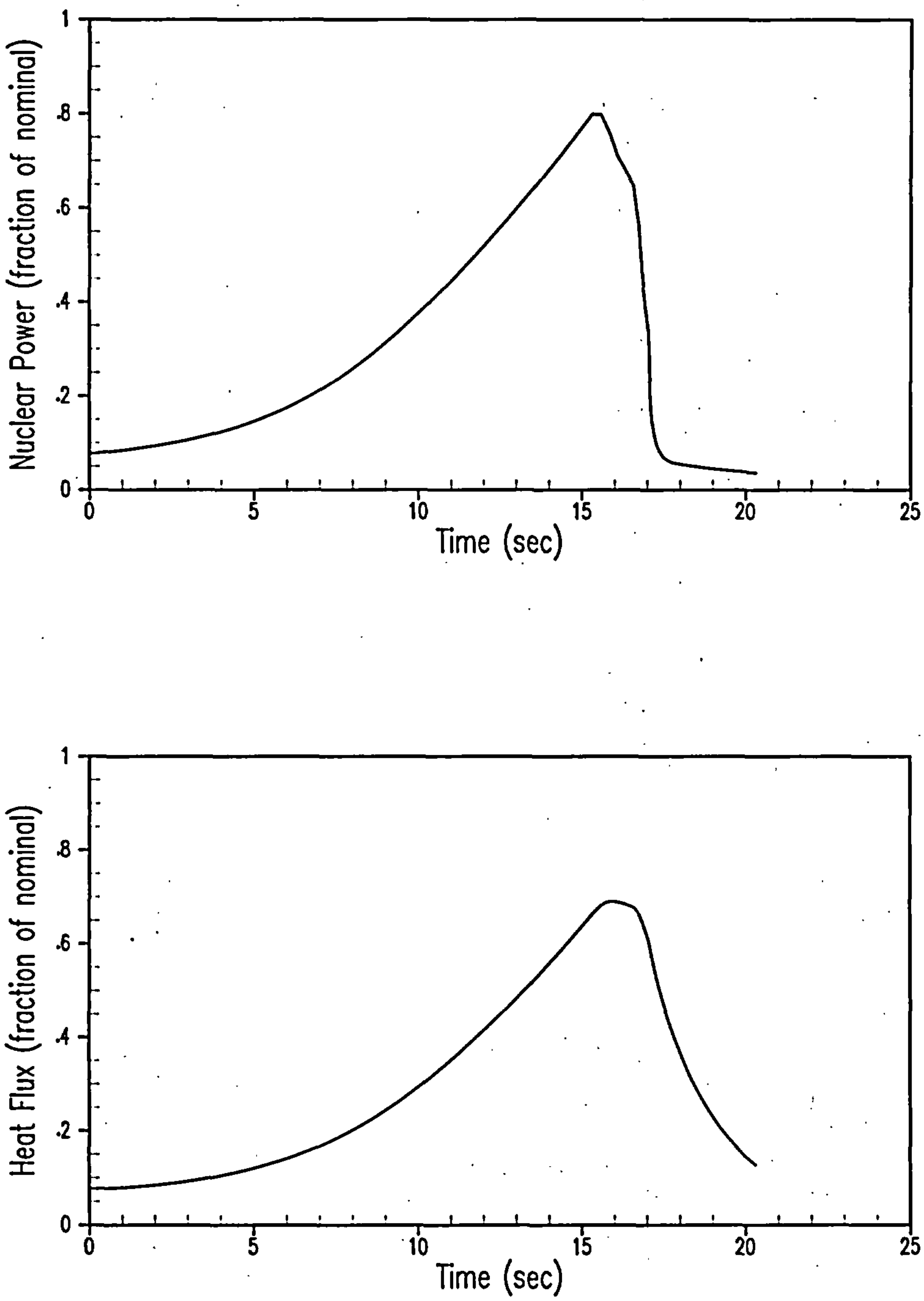
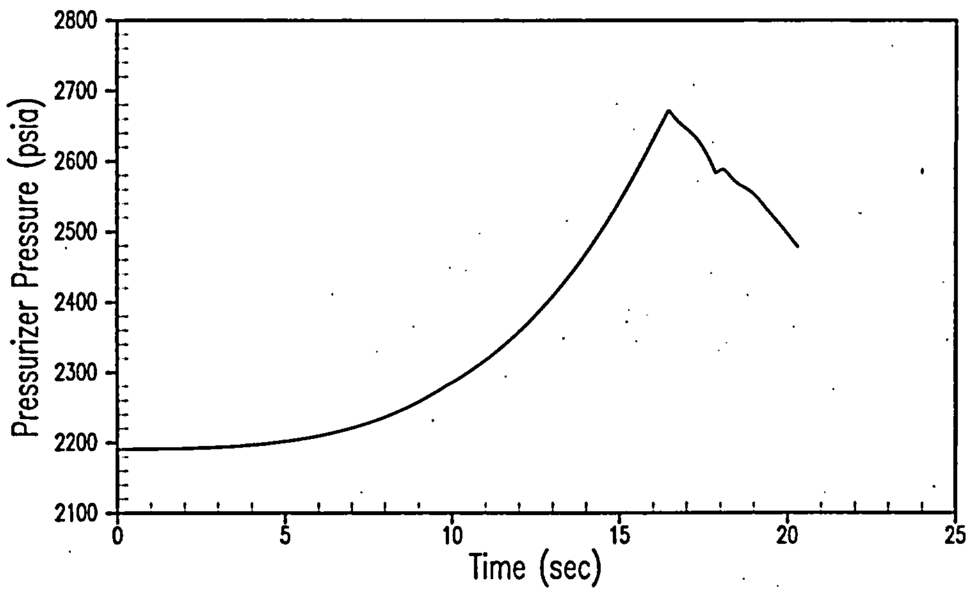


Figure 2.8.5.4.2-10
Rod Withdrawal at Power – RCS Pressure Case
Minimum Reactivity Feedback – 8% Power - 55 pcm/sec
Nuclear Power and Heat Flux vs. Time



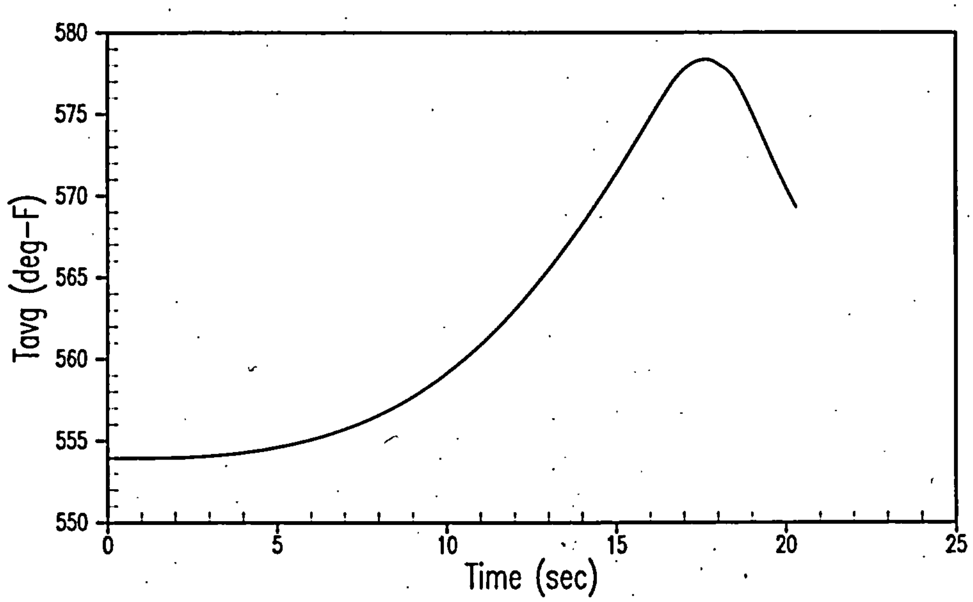


Figure 2.8.5.4.2-11
Rod Withdrawal at Power – RCS Pressure Case
Minimum Reactivity Feedback – 8% Power - 55 pcm/sec
Pressurizer Pressure and Vessel Average Temperature vs. Time

2.8.5.4.2.5 Conclusion

The Ginna staff has reviewed the analyses of the uncontrolled RCCA withdrawal at-power event and concludes that the analyses have adequately accounted for the changes in core design required for plant operation at the proposed power level. The Ginna staff also concludes that the analyses were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-20, and GDC-25 following implementation of the proposed EPU. Therefore, the Ginna staff finds the EPU acceptable with respect to the uncontrolled RCCA withdrawal at power.

2.8.5.4.3 Control Rod Misoperation

2.8.5.4.3.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered:

- The descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) that can mitigate the effects or prevent the occurrence of various misoperations
- The sequence of events
- The analytical model used for analyses
- The important inputs to the calculations
- The results of the analyses

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor core is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any normal operation condition, including the effects of anticipated operational occurrences
- GDC-20, insofar as it requires that the protection system is designed to initiate the reactivity control systems automatically to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to automatically initiate operation of important-to-safety systems and components under accident conditions
- GDC-25, insofar as it requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems

Specific review criteria are contained in the SRP Section 15.4.3 and other guidance provided in Matrix 8 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 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 of this positive reactivity addition event relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure that the fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Fuel design and nuclear design are further discussed in LR section 2.8.1 and LR section 2.8.2, respectively.

- GDC-20 is described in UFSAR section 3.1.2.3.1, General Design Criterion 20 – Protection Systems Functions. As described in this UFSAR section, a plant protection system is provided to automatically initiate appropriate action whenever plant specific conditions reach pre-established limits. These limits ensure that specified fuel design limits are not exceeded when anticipated operational occurrences happen. Fuel design for EPU conditions is evaluated in LR section 2.8.2. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

- GDC-25 is described in UFSAR section 3.1.2.3.6, General Design Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions. As described in this UFSAR section, the Reactor Trip System is designed to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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.

2.8.5.4.3.2 Technical Evaluation

2.8.5.4.3.2.1 Introduction

The rod cluster control assembly (RCCA) misalignment events include the following:

- One or more dropped RCCAs within the same group
- A dropped RCCA bank
- A statically misaligned RCCA

Each RCCA has a position indicator channel that displays the position of the assembly in a display grouping that is convenient to the operator. Fully inserted RCCAs are also indicated by a rod-at-bottom signal that actuates a control room annunciator. Group demand position is also indicated.

RCCAs move in preselected banks that always move in the same preselected sequence. Each bank of RCCAs consists of two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of the control rod drive mechanism (CRDM) withdraws the RCCA held by the mechanism. Mechanical failures are in the direction of insertion or immobility.

A dropped RCCA or RCCA bank is detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system

- Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

- Rod at bottom signal

- Rod deviation alarm

- Rod position indication

Dropping of a full-length RCCA is assumed to be initiated by a single electrical or mechanical failure that causes any number and combination of rods from the same group of a given control bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor can occur due to the skewed power distribution representative of a dropped rod configuration. For this event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following a dropped rod.

Misaligned RCCAs are detected by:

- Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples
- Rod deviation alarm
- Rod position indicators

Per the Ginna Station Technical Specification Bases, the resolution of each rod position indication channel is ±8 steps[1]. Deviation of any RCCA from its group by twice this distance (16 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to group demand position in excess of 12 steps. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indication channels is inoperable, detailed instructions are provided to assure the alignment of the associated RCCAs. The operator is also required to take action as required by the Technical Specifications.

Misaligned or dropped RCCA events cause a localized power depression. Assuming that the power excursion is insufficient to generate a reactor trip, and without crediting

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[1] The 14x14 422V+ fuel that will be implemented with EPU has a higher top-nozzle adapter plate than the current fuel as discussed in LR Section 2.8.1. Ginna is investigating hardware and software solutions to address the impact of this change on the rod position indication system. It is possible that a software solution may impact the rod position indication resolution near the lowest coil which corresponds to rods being near the bottom of the core. This impact will be assessed as part of any software change.

any automatic rod withdrawal block, the rod control system will attempt to return the power to nominal. With instantaneous reactivity feedback and dynamically compensated rod control, the power level may overshoot 100% power. The rod control system will eventually correct the power level to approximately nominal and reach equilibrium. In cases with low reactivity feedback and a high dropped RCCA worth, the reactor is essentially shut down by the dropped RCCA. In these cases, the low pressurizer pressure trip function will complete the shutdown.

2.8.5.4.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

RCCA misalignment events were analyzed generically. The purpose of the generic analysis was to generate transient statepoints that will be evaluated on a plant-specific, cycle-specific basis during the reload process. The initial conditions (except RCS Flow) for these events were analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 3).

The statepoints are in the form of changes in key parameters from the initial values. These changes will be applied to the actual plant-specific conditions for the EPU during each reload evaluation. The effect of a power increase on these generic statepoints has been previously addressed for other Westinghouse-designed PWRs.

Based on the frequency of occurrence, RCCA misalignment events are considered Condition II events. The primary acceptance criterion for these events is that the critical heat flux should not be exceeded and that fuel centerline melt is precluded. This is demonstrated by showing that the DNB design basis is met and that the peak kW/ft is below that which would cause fuel centerline melt.

2.8.5.4.3.2.3 Description of Analyses and Evaluations

The generic statepoints were evaluated using the VIPRE-W computer code (reference 1) to support the DNB and fuel centerline melt criteria and found to be applicable to the EPU. A detailed discussion of the Westinghouse Dropped Rod Methodology is contained in WCAP-11394 and WCAP-11395 (reference 2).

The specific acceptance criteria applied by Ginna for this event are as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met satisfies the Ginna Station current licensing basis requirements with respect to GDC-10.
- Per GDC-20, the protection system should be designed to automatically initiate the operation of appropriate systems, including the reactivity

control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and initiate the operation of safety-related systems and components. For this event, protection is provided via the overtemperature ΔT trip, but only for the most limiting cases. The non-limiting cases considered do not require protection.

- GDC-25 requires that the protection system is designed to ensure 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. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-25.

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.3.3 Results

The results of the evaluation for the RCCA misalignment events show that the DNBR does not fall below the safety analysis limit value and that the peak kW/ft remains below the value which would result in fuel centerline melt.

The DNB design basis criterion for the dropped rod event is met by confirming that the maximum allowable precondition radial peaking factor is greater than the design limit. The dropped rod limit lines are derived such that at any point on the line, for that combination of temperature and pressure, one can determine the power that just meets the DNBR limit. The limiting EPU margin was found to be 0.06% (422V+ fuel) and 1.62% (OFA fuel) at the vessel average temperature corresponding to the upper analyzed limit for RCS average temperature (see LR Table 1.1). Actual reload cycles are expected to operate at a vessel average temperature that is approximately 4°F cooler than this analysis, providing margin to account for reload-by-reload variations. Current nominal power (1520 MWt) dropped rod limit margin (Cycle 32) is approximately 1.8%.

The results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload process.

2.8.5.4.3.4 References

1. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, October 1999.

2. WCAP-11394 (Proprietary) and WCAP-11395 (Nonproprietary), *Methodology for the Analysis of the Dropped Rod Event*, April 1987.

3. WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), *Revised Thermal Design Procedure*, April 1989.

2.8.5.4.3.5 Conclusion

The Ginna staff has reviewed the analyses of control rod misoperation events and concludes that the analyses have adequately accounted for the changes in core design required for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits will not be exceeded during normal or anticipated operational transients. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-20, and GDC-25 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to control rod misoperation events.

2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

2.8.5.4.4.1 Regulatory Evaluation

A startup of an inactive loop transient can result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The sequence of events
- The analytical model
- The values of parameters used in the analytical model
- The results of the transient analyses.

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor coolant system (RCS) is 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

- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems are designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences

- GDC-20, insofar as it requires that the protection system is designed to automatically initiate the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of operational occurrences

- GDC-26, insofar as it requires that a reactivity control system is provided and be capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

- GDC-28, insofar as it requires that the reactivity control systems are designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core

Specific review criteria are contained in SRP section 15.4.4-5 and other guidance provided in Matrix 8 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 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 of this positive reactivity addition event relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure that the fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Fuel design and nuclear design are further discussed in LR section 2.8.1 and LR section 2.8.2, respectively.

- GDC-15 is described in 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, control, and protection systems are designed with sufficient margins so that design conditions are not exceeded during Modes 1 and 2 including anticipated operational occurrences. Overpressurization of the reactor coolant pressure boundary is prevented by a combination of automatic control and pressure relief devices. The analysis of the startup of an inactive reactor coolant loop is discussed in UFSAR section 15.4.3.

- GDC-20 is described in UFSAR section 3.1.2.3.1, General Design Criterion 20 – Protection Systems Functions. As described in this UFSAR section, a plant protection system is provided to automatically initiate appropriate action whenever plant specific conditions reach preestablished limits. These limits ensure that specified fuel design limits are not exceeded when anticipated operational occurrences happen. Fuel design for EPU conditions is evaluated in LR section 2.8.2. The Reactor Trip System, which provides a protective function to prevent fuel limits from being exceeded, is described in UFSAR section 7.2.1.1.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, the control rods are designed to shutdown the reactor with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. The reactivity control provided by the control rods is further discussed in UFSAR section 4.3.2.1.

- GDC-28 is described in UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. As described is this UFSAR section, the maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling. Protection against a positive reactivity excursion from low power conditions such as the event discussed in this LR section is described in UFSAR section 15.4.3.3.1

Specific review criteria are contained in SRP section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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.

2.8.5.4.4.2 Technical Evaluation

2.8.5.4.4.2.1 Introduction

The plant is initially at steady state, low power conditions with one reactor coolant pump running. The second reactor coolant pump is started. Reactor coolant flow instantaneously increases to nominal full flow conditions.

2.8.5.4.4.2.2 Description of Analysis

The Ginna Station Technical Specifications preclude operation with an RCS loop out of service above 8.5% power. As such, this event is not credible for Ginna Station at power levels greater than 8.5% RTP. For power levels ≤ 8.5% RTP, this event is not limiting. The Ginna UFSAR analysis was performed at 8.5% power and assumed a conservatively high temperature difference between the active loop cold leg and the inactive loop hot leg of 20°F.

The specific acceptance criteria applied by Ginna for this event are as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met meets the Ginna Station current licensing basis requirements with respect to GDC-10 and GDC-28.

- Per GDC-20, the protection system should be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and to initiate the operation of important-to-safety systems and components. Due to the low initial power level (8.5% RTP), no protection is needed and no protection is assumed for this event.

- Primary and secondary pressures must remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-15.

- GDC-26 requires reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ensuring that appropriate margin for malfunctions, such as stuck rods, are accounted for in the safety analysis assumptions. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-26.

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.4.3 Results

The EPU does not significantly change the assumption of conservatively high temperature difference between the two loops (i.e., T_{hot}, T_{cold}, T_{avg} at 8.5% power are essentially unchanged between pre-EPU and EPU conditions). Moderator density coefficient limits are essentially unchanged for the EPU. All applicable acceptance criteria are met for this event at EPU conditions. Therefore, the conclusions presented in UFSAR section 15.4.3, Startup of an Inactive Reactor Coolant Loop, remain valid.

2.8.5.4.4.4 Conclusion

The Ginna staff has reviewed the analyses of the inactive-loop-startup event and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, the Ginna staff concludes that the plant will continue to meet the requirements of GDC -10, GDC-15, GDC-20, GDC-26, and GDC-28 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to improper startup of a reactor coolant loop resulting in an increase in core flow event.

2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

2.8.5.4.5.1 Regulatory Evaluation

Unborated water can be added to the reactor coolant system (RCS) via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator should stop this unplanned dilution before the shutdown margin is eliminated. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The conditions at the time of the unplanned dilution
- The causes
- The initiating events
- The sequence of events
- The analytical model used for analyses
- The values of parameters used in the analytical model
- The results of the analyses

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences

- GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems are designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences

- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Specific review criteria are contained in the SRP section 15.4.6 and other guidance provided in Matrix 8 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 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 of this positive reactivity addition event relative to conformance to:

- GDC-10 is described in UFSAR section 3.1.2.2.1, General Design Criterion 10 – Reactor Design. As described in this UFSAR section, the reactor core design, in combination with coolant, control and protection systems, provides margins to ensure that the fuel is not damaged during Modes 1 and 2 or as the result of anticipated operational transients. Fuel design and nuclear design are further discussed in <u>LR section 2.8.1</u> and <u>LR section 2.8.2</u>, respectively.

- GDC-15 is described in 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, control, and protection systems are designed with sufficient margins so that design conditions are not exceeded during Modes 1 and 2 including anticipated operational occurrences. Overpressurization of the reactor coolant pressure boundary is prevented by a combination of automatic control and pressure relief devices.

- GDC-26 is described in UFSAR section 3.1.2.3.7, General Design Criterion 26 – Reactivity Control System Redundancy and Capability. As described in this UFSAR section, the control rods are designed to shutdown the reactor with adequate margin for all anticipated occurrences so that fuel design limits are not exceeded. In addition, the chemical and volume control system (CVCS) provides a reactivity control function by regulating the concentration of boric acid neutron absorber in the reactor coolant system. The reactivity control provided by the CVCS is further discussed in UFSAR section 9.3.4. The analysis of a CVCS malfunction (boron dilution) event is discussed in UFSAR section 15.4.4.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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.

2.8.5.4.5.2 Technical Evaluation

2.8.5.4.5.2.1 Introduction

Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator, during normal charging, to match the boron concentration of makeup water to the existing RCS boron concentration. As discussed below, the CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value that, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

2.8.5.4.5.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The opening of the primary water makeup control valves provides makeup to the CVCS and subsequently to the RCS, which can dilute the boron concentration of the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

There is only a single line connecting the primary grade water header to the CVCS and inadvertent dilution can be readily terminated by isolating this line. The primary grade water header can be supplied by either the primary grade water pumps or from a cross connection to the turbine plant demineralized water system. The maximum dilution flow is 127 gpm (Mode 1) and 120 gpm (Modes 2 and 6), based on operation of both reactor makeup water pumps in addition to two charging pumps if the RCS is at pressure.

Boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required. The operator must switch from the automatic makeup mode to the dilute or alternate dilute mode, and the start switch must be placed in the start position. Omitting either step would prevent dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or makeup water flow rates deviate from preset values as a result of system malfunction.

A CVCS malfunction is classified as an ANS Condition II event, a fault of moderate frequency. Criteria established for Condition II events are as follows:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the RCS and main steam systems (MSS) should be maintained below 110% of the design pressures.
- Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value that would cause fuel centerline melt.

This event is analyzed to show that there is sufficient time for mitigation of an inadvertent boron dilution prior to the complete loss of shutdown margin. A complete loss of plant shutdown margin results in a return of the core to the critical condition causing an increase in the RCS temperature and heat flux. This could violate the safety analysis limit DNBR value and challenge the fuel and fuel cladding integrity. A complete loss of plant shutdown margin could also result in a return of the core to the critical condition causing an increase in RCS pressure. This could challenge the pressure design limit for the RCS.

If the minimum allowable shutdown margin is shown not to be lost, the condition of the plant at any point in the transient is within the bounds of those calculated for other Condition II transients. By showing that the above criteria are met for those Condition II events, it can be concluded that they are also met for the boron dilution event. Operator action is relied upon to preclude a complete loss of plant shutdown margin.

The specific acceptance criterion applied by Ginna for these events is that adequate operator action time is available prior to a complete loss of shutdown margin. For boron dilution events in Modes 1 and 2, there must be at least 15 minutes from the start of the dilution until shutdown margin is lost. For boron dilution events in Mode 6, there must be at least 30 minutes from the start of the dilution until shutdown margin is lost. With shutdown margin maintained, there is no return to critical and no violation of the 95/95

DNBR limit (GDC-10), as well as no violation of the primary and secondary pressures limits (GDC-15). Furthermore, since a return to criticality is precluded and fuel design limits are not exceeded, the Ginna Station requirements with respect to GDC-26 are met.

2.8.5.4.5.2.3 Description of Analyses and Evaluations

Dilution During Mode 6

The analysis of the boron dilution event during Mode 6 assumed a maximum dilution flow rate of 120 gpm. An active RCS volume of 2042 ft^3 was assumed. From the initiation of the event, there were more than 30 minutes available for operator action prior to the complete loss of shutdown margin.

Dilution During Mode 2

In this mode, the plant is being taken from one long-term mode of operation (Mode 3) to another (Mode 1). Typically, the plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. All normal actions required to change power level, either up or down, require operator initiation.

The analysis of the boron dilution event in Mode 2 assumed a maximum dilution flow rate of 120 gpm. An active RCS volume of 5123 ft^3 was assumed.

Mode 2 was a transitory operational mode in which the operator intentionally diluted and withdrew control rods to achieve criticality. During this mode, the rods were in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods. The operator determined the estimated critical position of the control rods prior to approaching criticality, thus ensuring that the reactor did not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip (nominally at 10^5 cps) after receiving P-6 from the intermediate range. Too fast of a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip, and the reactor would immediately shut down.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in Mode 2, the plant status is such that minimal impact resulted. The plant slowly escalated in power until the power range high neutron flux trip setpoint was reached and a reactor trip occurred. From the initiation of the event, there were more than 15 minutes available for operator action prior to return to criticality.

Dilution During Mode 1

In this mode, the plant can be operated in either automatic or manual rod control. The analysis assumed a maximum dilution flow rate of 127 gpm. An active RCS volume of 5123 ft^3 was assumed.

With the reactor in automatic rod control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in available shutdown margin. The rod insertion limit alarms (low and low-low settings) alerted the operator at least 15 minutes prior to a complete loss of shutdown margin. This was sufficient time to determine the cause of dilution and isolate the reactor makeup water source before the available shutdown margin was lost.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise caused the reactor to reach the power range high neutron flux trip setpoint or the overtemperature ΔT trip setpoint, resulting in a reactor trip. The boron dilution transient in this case was essentially equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively 2.4 pcm/sec, which was within the range of insertion rates analyzed for the uncontrolled RCCA bank withdrawal at power. Thus, the effects of dilution prior to reactor trip were bounded by the uncontrolled RCCA bank withdrawal at power analysis (LR section 2.8.5.4.2). Following reactor trip, there were more than 15 minutes prior to criticality. This was sufficient time for the operator to determine the cause of dilution and isolate the reactor makeup water source before the available shutdown margin was lost.

Evaluation of Boron Dilution During Cold Shutdown

A plant-specific evaluation of the boron dilution event during cold shutdown was performed. This evaluation is based upon the operating procedure outlined in reference 1. The operating procedure is based upon a generic boron dilution analysis assuming active RCS and RHR volumes which are consistent with respect to Ginna. Additionally, the operating procedure accommodates drained down operation at cold shutdown conditions. The operating procedure is applicable for maximum dilution flow rates up to 300 gal/min and minimum RHR flow rates of 1000 gal/min. When RHR flow is less than 1000 gpm in Mode 5 and in a reduced inventory condition, administrative measures are invoked to prevent a boron dilution. In this condition, the reference 1 procedure is not applicable. Current plant procedures require one reactor makeup water pump to be secured when no reactor coolant pumps are running, limiting the maximum dilution flow rate to 120 gal/min. In the event of a boron dilution accident during plant shutdown, use of the operating procedure provides the plant operator with sufficient information to maintain an appropriate boron concentration to conservatively assure at least 15 minutes will be available for operator action to terminate the dilution prior to the reactor reaching a critical condition.

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.5.3 Results

If an unintentional dilution of boron in the RCS does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator sufficient time to determine the cause of the addition and take corrective action before shutdown margin is lost.

The boron dilution analysis demonstrated that all applicable acceptance criteria are met at EPU conditions. This means that operator action to terminate the dilution flow within 15 minutes from event initiation from Mode 1 and Mode 2 and within 30 minutes from event initiation from Mode 6 precludes a complete loss of shutdown margin. The results of the boron dilution analysis are provided in Table 2.8.5.4.5-1.

2.8.5.4.5.4 Reference

1. RGE-05-28, *Revised Boron Dilution Interim Operating Procedure for Ginna*, April 2005.

Table 2.8.5.4.5-1 CVCS Malfunction Boron Dilution Event Results and Comparison to Previous Results			
Case *	**EPU Analysis**	**Previous Analysis**	**Limit**
Available Operator Action Time in Mode 1 – Manual Rod Control	**30.3**	**47.2**	**15**
Available Operator Action Time in Mode 1 – Automatic Rod Control	**33.3**	**37.7**	**15**
Available Operator Action Time in Mode 2	**25.1**	**33.9**	**15**
Available Operator Action Time in Mode 6	**32.0**	**30.08**	**30**

* For each case, the initial boron concentration and the critical boron concentration are verified on a cycle specific basis. The initial and critical boron concentrations used for the EPU analyses were optimized to facilitate reload evaluations. As such, a comparison of the EPU results to the previous analysis results is meaningless.

2.8.5.4.5.5 Conclusion

The Ginna staff has reviewed the analyses of the decrease in boron concentration in the reactor coolant due to a CVCS malfunction and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the decrease in boron concentration in the reactor coolant due to a CVCS malfunction.

2.8.5.4.6 Spectrum of Rod Ejection Accidents

2.8.5.4.6.1 Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The Ginna Nuclear Power Plant, LLC (Ginna) evaluated the consequences of a control rod ejection accident to determine the potential damage caused to the reactor coolant pressure boundary and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. Ginna's review covered:

- The initial conditions.
- The rod patterns and worths, scram worth as a function of time, and reactivity coefficients.
- The analytical model.
- The core parameters that affect the peak reactor pressure or the probability of fuel rod failure.
- The results of the transient analyses.

The NRC's acceptance criteria are based on:

- GDC-28, insofar as it requires that the reactivity control systems are designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Specific review criteria are contained in the SRP Section 15.4.8 and other guidance provided in Matrix 8 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 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 this positive reactivity addition event relative to conformance to:

- GDC-28 is described in UFSAR section 3.1.2.3.9, General Design Criterion 28 – Reactivity Limits. As described is this UFSAR section, the maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited by the design of the facility to values which prevent failure of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling. The description and analysis of a rod ejection event are provided in UFSAR section 15.4.5.

In addition to the evaluations described in the Ginna UFSAR, the components of the reactivity control and protection system 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.

2.8.5.4.6.2 Technical Evaluation

2.8.5.4.6.2.1 Introduction

This accident is defined as a mechanical failure of a control rod drive mechanism (CRDM) pressure housing resulting in the ejection of the rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a CRDM housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

- Each full-length CRDM housing is completely assembled and shop tested at 4100 psig.

- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed reactor coolant system (RCS).

- Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress ranges specified in the *ASME Boiler and Pressure Vessel Code*, Section III (reference 2), for Class I components.

- The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that the gross failure of the housing will not occur. The joints between the latch mechanism housing and rod travel housing are threaded joints and reinforced by canopy-type rod welds.

In general, the reactor is operated with the RCCA inserted only far enough to control design neutron flux shape. Reactivity changes caused by the core depletion are compensated by boron changes. Furthermore, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, if a RCCA is ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur. The position of all RCCA is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one control rod assembly deviates from its bank. There are low and low-low level insertion alarm circuits for each bank. The control rod position monitoring and alarm systems are described in WCAP-7588 (reference 1).

2.8.5.4.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Input parameters for the analysis were conservatively selected on the basis of values calculated for this type of core. The most important parameters are discussed below. Table 2.8.5.4.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors were calculated using either three-dimensional (3-D) static methods or a synthesis of one-dimensional (1-D) and two-dimensional (2-D) calculations. Standard nuclear design codes were used in the analysis. No credit was taken for the flux-flattening effects of reactivity feedback. The calculation was performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. The analysis assumed adverse xenon distributions to provide worst-case results.

Appropriate margins were added to the ejected rod worth and hot channel factors to account for any calculational uncertainties.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yielded values of approximately 0.65% at BOL and 0.48% at EOL. The ejected rod accident was sensitive to β if the ejected rod worth was equal to or greater than β_{eff}, as in the zero-power transients. In order to allow for future fuel cycle flexibility, conservative estimates of β of 0.49% at beginning of cycle and 0.43% at end of cycle were used in the analysis.

Reactivity Weighting Factor

The largest temperature rises, and hence the largest reactivity feedbacks, occurred in channels where the power was higher than average. Since the weight of a region was dependent on flux, these regions had high weights. This means that the reactivity feedback was larger than that indicated by a simple single-channel analysis. Physics calculations were performed for temperature changes with a flat temperature distribution and a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors took the form of multipliers that, when applied to single-channel feedbacks, corrected them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, a one-dimensional (axial) spatial kinetics method is employed, thus axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three-dimensional analysis.

Moderator and Doppler Coefficient

The critical boron concentrations at the BOL and EOL were adjusted in the nuclear code in order to obtain moderator density coefficient curves that were conservative when compared to the actual design conditions for the plant. As discussed above, no weighting factor was applied to these results. The resulting MTC was at least +5 pcm/°F at the appropriate zero-power nominal T_{avg} for the BOL cases.

The Doppler reactivity defect was determined as a function of power level using a 1-D steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor increased under accident conditions, as discussed above.

Heat Transfer Data

The FACTRAN (reference 3) code used to determine the hot spot transient contains standard curves of thermal conductivity versus fuel temperature. During a transient, the peak centerline fuel temperature was independent of the gap conductance during the transient. The cladding temperature was, however, strongly dependent on the gap conductance and was highest for high gap conductance. For conservatism, a high gap heat transfer coefficient value of 10,000 Btu/hr-ft^2-°F was used during transients. This value corresponded to a negligible gap resistance and a further increase would have essentially no effect on the rate of heat transfer.

Coolant Mass Flow Rates

When the core is operating at full power, both reactor coolant pumps always operate. For zero power conditions, the system was conservatively assumed to be operating with one pump. The principal effect of operating at reduced flow was to reduce the film boiling heat transfer coefficient. This resulted in higher peak cladding temperatures, but did not affect the peak centerline fuel temperature. Reduced flow also lowered the critical heat flux. However, since DNB was always assumed at the hot spot, and since the heat flux rose very rapidly during the transient, this produced only second order changes in the cladding and centerline fuel temperatures.

Trip Reactivity Insertion

The trip reactivity insertion was assumed to be 3.5% Δk from hot full power (HFP) and 2.0% Δk from hot zero power (HZP), including the effect of one stuck RCCA. These values were also reduced by the ejected rod. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after reaching the power range high neutron flux trip setpoint. It was assumed that insertion to dashpot did not occur until 1.8 seconds after the rods began to fall. The time delay to full insertion, combined with the 0.5 second trip delay, conservatively delayed insertion of shutdown reactivity into the core.

Due to the extremely low probability of a RCCA ejection accident, this event was classified as an ANS Condition IV event. As such, some fuel damage was considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy were carried out as part of the SPERT project by the Idaho Nuclear Corporation (reference 4). Extensive tests of UO_2 zirconium-clad fuel rods representative of those present in pressurized water reactor (PWR) type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design exhibited failure as low as

225 cal/gm. These results differ significantly from the TREAT (reference 5) results that indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreased 10% with fuel burnup. The clad failure mechanism appeared to be melting for unirradiated (zero burnup) rods and brittle fracture for irradiated rods. The conversion ratio of thermal to mechanical energy was also important. This ratio became marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

The real physical limits of this accident were that the rod ejection event and any consequential damage to either the core or the RCS must not prevent long-term core cooling, and any offsite dose consequences must be within the guidelines of 10CFR50.67 (reference 6). More-specific and restrictive criteria were applied to ensure fuel dispersal in the coolant. Gross lattice distortion or severe shock waves did not occur. In view of the above experimental results, and the conclusions of WCAP-7588, Rev. 1-A (reference 1) and reference 7, the limiting criteria were:

- Average fuel pellet enthalpy at the hot spot must be maintained below 225 cal/gm for unirradiated and 200 cal/gm (360 Btu/lbm) for irradiated fuel (the 360 Btu/lbm limit is applied).

- Peak reactor coolant pressure must be less than that which could cause RCS stresses to exceed the faulted-condition stress limits (note: the peak pressure aspects of the rod ejection transient are addressed generically in reference 1).

- Fuel melting is limited to less than 10% of the pellet volume at the hot spot even if the average fuel pellet enthalpy is below the 360 Btu/lbm fuel enthalpy limit.

The criterion applied by Ginna to ensure the core remains in a coolable geometry following a rod ejection incident is that the average fuel pellet enthalpy at the hot spot must remain less than 200 cal/g (360 Btu/lbm). The use of the initial conditions presented in Table 2.8.5.4.6-1 resulted in conservative calculations of the fuel pellet enthalpy. The results of the licensing basis analyses demonstrated that the fuel pellet enthalpy does not exceed 360 Btu/lbm for any of the rod ejection cases analyzed.

Overpressurization of the RCS during a rod ejection event is generically addressed in WCAP-7588, Revision 1-A (reference 1). The RCS pressure limit is 3200 psig.

Another applicable acceptance criterion is that fuel melting must be limited to less than the innermost 10% of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is less than the limit of 360 Btu/lbm. Conservative fuel melt temperatures of 4900° and 4800°F were assumed for the hot spot for the beginning-of-life (BOL) and end-of-life (EOL) cases, respectively. The results of the licensing basis rod ejection analyses demonstrated that the amount of fuel melting was limited to less than 10% of the fuel pellet at the hot spot for each of the rod ejection cases.

2.8.5.4.6.2.3 Description of Analyses and Evaluations

This section describes the models used in the analysis of the rod ejection accident. Only the initial few seconds of the power transient are discussed, since the long-term considerations were the same as for a small loss of coolant accident (LOCA).

The calculation of the RCCA ejection transient was performed in two stages, first an average core channel calculation, and then a hot region calculation. The average core calculation used spatial neutron-kinetics methods to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot were then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback was conservatively assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in reference 1.

Average Core

The spatial-kinetics computer code, TWINKLE (reference 7) was used for the average core transient analysis. This code solved the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. This analysis used the code as a 1-D axial kinetics code since it allows a more-realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension was missing, it was still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The average core energy addition, calculated as described above, was multiplied by the appropriate hot channel factors. The hot spot analysis used the detailed fuel and clad transient heat transfer computer code, FACTRAN (reference 3). This computer code calculated the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using the nuclear power versus time and local coolant conditions as input. The zirconium-water reaction was explicitly represented, and all material properties were represented as functions of temperature. A parabolic radial power distribution was assumed within the fuel rod.

FACTRAN used the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (reference 8) to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation was conservatively used assuming zero bulk fluid quality. The DNB heat flux was not calculated; instead the code was forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient could be calculated by the code; however, it was adjusted to force the full-power, steady-state temperature distribution to agree with fuel heat transfer design codes.

Reactor Protection

The protection for this accident, as explicitly modeled in the analysis, was provided by the power range high neutron flux trip (high and low settings).

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

Components of the reactivity control and protection systems that are within the scope of License Renewal are electrical and instrumentation and control components that are treated as commodity groups in NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786, section 3.6. 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 reactivity control and protection 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.

2.8.5.4.6.3 Results

The results of the analyses performed for the rod ejection event, which cover BOL and EOL conditions at HFP and HZP, are discussed below. The analyses were performed for both the V+ and OFA fuel types.

Beginning of Cycle, Zero Power

The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.75% ΔK, and 11.0, respectively. The peak hot spot average fuel pellet enthalpy reached 273.3 Btu/lbm (151.8 cal/gm). The peak fuel centerline temperature never reached the BOL melt temperature of 4900°F, so no fuel melting is predicted. The uprating analysis included a decrease in the ejected rod worth and increased in the feedback reactivity weighting and Doppler defect. The benefits realized from these changes more than offset the effect of the power uprating, as anticipated. As expected, the optimized fuel assembly (OFA) design was shown to be more limiting than the V+ fuel design, due to the dimensional differences between the two fuel types. The results presented were those of the most limiting case (OFA).

Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.32% ΔK and 5.0, respectively. The peak hot spot average fuel pellet enthalpy reached 320.3 Btu/lbm (177.9 cal/gm). The peak fuel centerline temperature reached the BOL melt temperature of 4900°F; however, fuel melting remained well below the limiting criterion of 10% of total pellet volume at the hot spot. The Beginning-of Cycle (BOC) HFP case was expected to show more limiting results due to the power uprating, and increased ejected rod worth. Increased feedback reactivity weighting, Doppler defect, and use of the Technical Specifications' MTC at HFP conditions compensated for much of the uprating impact, resulting in only a small overall impact on the BOC HFP results. As expected, the OFA design was shown to be more limiting than the V+ fuel design, due to the dimensional differences between the two fuel types. The results presented are those of the most limiting case (OFA).

End of Cycle, Zero Power

The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.90% ΔK and 14.0 (V+) or 12.0 (OFA), respectively. As expected, the OFA design was shown to be more limiting than the V+ fuel design, due to the dimensional differences between the two fuel types. The results presented are those of the most limiting case (OFA). The peak hot spot average fuel pellet enthalpy reached 279.1 Btu/lbm (155.1 cal/gm). The peak fuel centerline temperature never reached the EOL melt temperature of 4800°F, so no fuel melting is predicted. The EOC HZP case was impacted by the uprating, mostly due to the decreased feedback reactivity weighting and change in isothermal temperature coefficient used in the analysis. Considering the input changes, the results were consistent with expectations.

End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.40% ΔK and 5.69, respectively. The peak hot spot average fuel pellet enthalpy reached 319.0 Btu/lbm (177.2 cal/gm). The peak fuel centerline temperature reached melting, conservatively assumed at 4800°F; however, fuel melting remained well below the limiting criterion of 10% of the pellet volume at the hot spot. Based solely on the power uprating, the EOC HFP results would be expected to be more limiting than the previous analysis. An increase in the feedback reactivity weighting and Doppler defect, and a decrease in the ejected rod worth, serve to limit the impact of the power uprating to a slight change, as anticipated. As expected, the OFA design was shown to be more limiting than the V+ fuel design, due to the dimensional differences between the two fuel types. The results presented are those of the most limiting case (OFA).

A summary of the parameters used in the rod ejection analyses, and the analyses results, are presented in Table 2.8.5.4.6-1. The sequence of events for all four cases is presented in Table 2.8.5.4.6-2. Figure 2.8.5.4.6-1 and Figure 2.8.5.4.6-2 show the transient curves for the BOL/HZP cases; Figure 2.8.5.4.6-3 and Figure 2.8.5.4.6-4 show the transient curves for the BOL/HFP cases. Figure 2.8.5.4.6-5 and Figure 2.8.5.4.6-6 show the transient curves for the EOL/HZP cases; Figure 2.8.5.4.6-7 and Figure 2.8.5.4.6-8 show the transient curves for the EOL/HFP cases. Numerical results of the EPU analysis along with a comparison to the previous analysis results are shown in Table 2.8.5.4.6-3. The EPU analyses are more limiting than the previous analyses with the exception of the BOL-HZP cases. The ejected rod worth and ejected F_Q were reduced for the EPU analyses. These parameters are verified on a cycle specific basis as part of the reload evaluation process. Table 2.8.5.4.6-1 shows the ejected rod worths and ejected F_Qs assumed in the analyses.

A detailed calculation of the pressure surge for an ejected rod worth of 1 dollar at BOL, HFP, indicates that the peak pressure did not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits (reference 1). Since the severity of the present analysis did not exceed the "worst-case" analysis, the accident for this plant will not result in an excessive pressure rise or further adverse effects to the RCS.

Despite the conservative assumptions, the analyses indicate that the described fuel and clad limits were not exceeded. It was concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure did not exceed that which would cause stresses to exceed the faulted condition stress limits, it was concluded that there is no danger of further consequential damage to the RCS. Generic analyses demonstrated that the fission product release as a result of fuel rods entering DNB was limited to less than 10% of the fuel rods in the core.

The results and conclusions of the analyses performed for the rupture of a CRDM housing RCCA ejection support operation up to the uprated core power of 1811 MWt, including uncertainties.

Overpressurization of the RCS during a rod ejection event is generically addressed in WCAP-7588, Revision 1-A (reference 1).

2.8.5.4.6.4 References

1. WCAP-7588; Rev. 1-A, *An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Special Kinetics Methods*, January 1975.

2. *American Society of Mechanical Engineers*, Section III, The American Society of Mechanical Engineers, New York.

3. WCAP-7908-A, *FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod*, December 1989.

4. IN-1370, *Annual Report – SPERT Project, October 1968 – September 1969*, Idaho Nuclear Corporation, June 1970.

5. *Studies in TREAT of Zircaloy 2-Clad, UO_2-Core Simulated Fuel Elements, ANL-7225*, p. 177, November 1966.

6. 10CFR50.67, *Accident Source Term.*

7. WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Nonproprietary) *TWINKLE, A Multi-Dimensional Neutron Kinetics Computer Code*

8. ASME 65-HT-31, *Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux*, August 1965.

Table 2.8.5.4.6-1
Parameters and Results of the Limiting RCCA Ejection Analysis (OFA)

	Beginning of Cycle	Beginning of Cycle	End of Cycle	End of Cycle
Core Power Level, MWt	1811	0	1811	0
Ejected Rod Worth, % ΔK	0.32	0.75	0.40	0.90
Delayed Neutron Fraction, %	0.49	0.49	0.43	0.43
Feedback Reactivity Weighting	1.231	2.008	1.316	2.248 (V+) 2.041 (OFA)
Trip Reactivity, % ΔK	3.5	2.0	3.5	2.0
F_Q Before Rod Ejection	2.60	--	2.60	--
F_Q after Rod Ejection	5.0	11.0	5.69	14.0 (V+) 12.0 (OFA)
Number of Operational Pumps	2	1	2	1
Max Fuel Pellet Average Temperature, °F	4069	3564	4055	3627
Max Fuel Centerline Temperature, °F	>4900	3934	>4800	3920
Max Clad Average Temperature, °F	2313	2831	2306	2981
Max Fuel Stored Energy, cal/g	177.9	151.8	177.2	155.1
Fuel Melt at the Hot Spot, %	6.62	0	9.00	0

Table 2.8.5.4.6-2 Time Sequence of Events – RCCA Ejection		
Event	Time (sec)	
	BOL HFP	EOL HFP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.04	0.02
Peak Nuclear Power Occurs	0.14	0.13
Rods Begin to Fall	0.54	0.52
Peak Fuel Average Temperature Occurs	1.74	1.81
Peak Clad Temperature Occurs	1.95	2.00
	BOL HZP	EOL HZP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.22	0.15
Peak Nuclear Power Occurs	0.26	0.18
Rods Begin to Fall	0.72	0.65
Peak Clad Temperature Occurs	1.99	1.51
Peak Fuel Average Temperature Occurs	2.00	1.55

Table 2.8.5.4.6-3 RCCA Ejection Results and Comparison to Previous Licensing Basis Results					
Beginning of Cycle Cases					
	BOL/HFP EPU	**BOL/HFP Previous**	**BOL/HZP EPU**	**BOL/HZP Previous**	**Limit**
Max Fuel Stored Energy, Btu/lbm	320.3	317.4	273.3	287.6	360
Fuel Melt at the Hot Spot, %	6.62	5.10	0.0	0.0	10
Max Clad Average Temperature, °F	2313	2299	2831	2961	3000
Reacted Zirc, %	1.02	1.01	3.64	4.89	16
End of Cycle Cases					
	EOL/HFP EPU	**EOL/HFP Previous**	**EOL/HZP EPU**	**EOL/HZP Previous**	**Limit**
Max Fuel Stored Energy, cal/g	319.0	314.1	279.1	265.2	360
Fuel Melt at the Hot Spot, %	9.00	7.73	0.0	0.0	10
Max Clad Average Temperature, °F	2306	2269	2981	2781	3000
Reacted Zirc, %	1.02	0.97	4.68	3.35	16

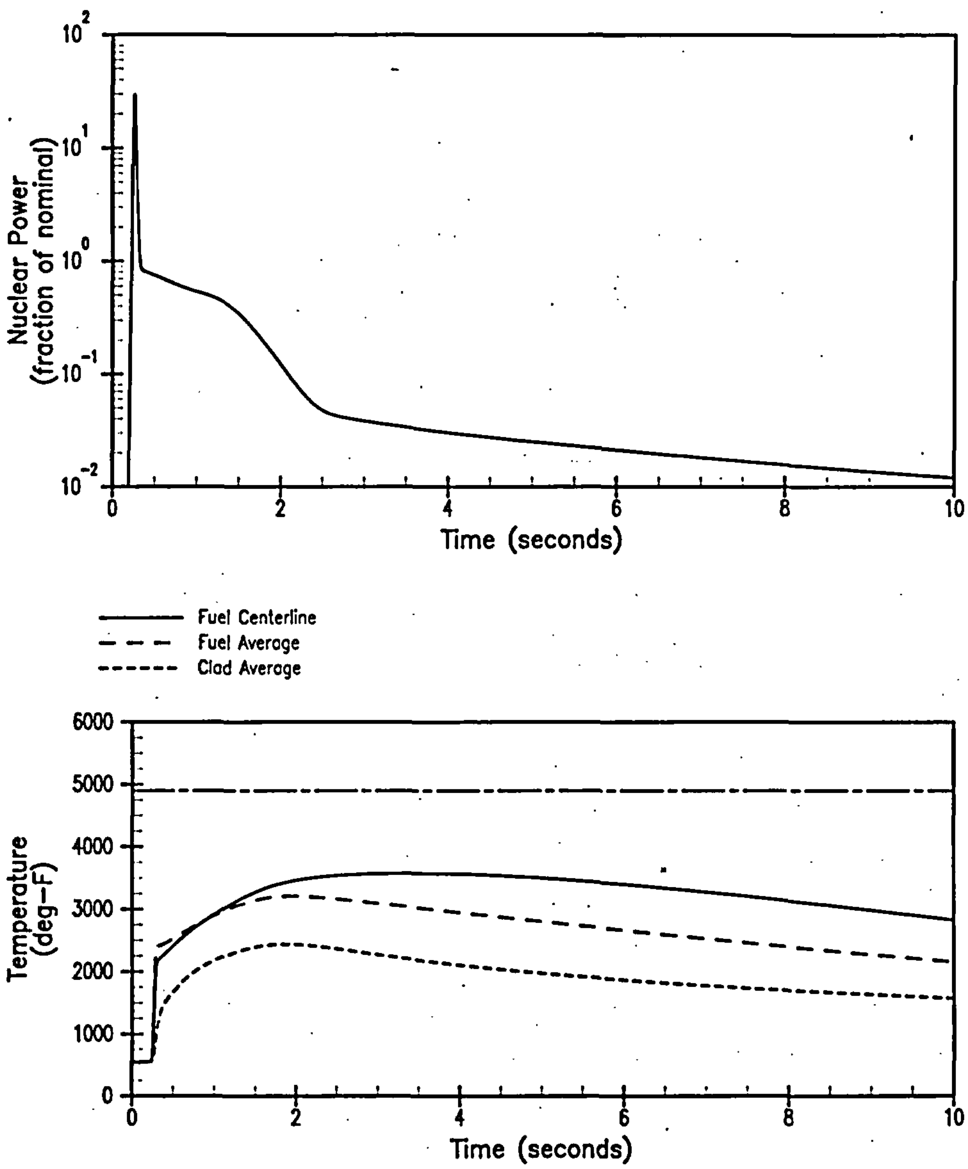


Figure 2.8.5.4.6-1
Rod Ejection – BOL/HZP Case
V+ Design

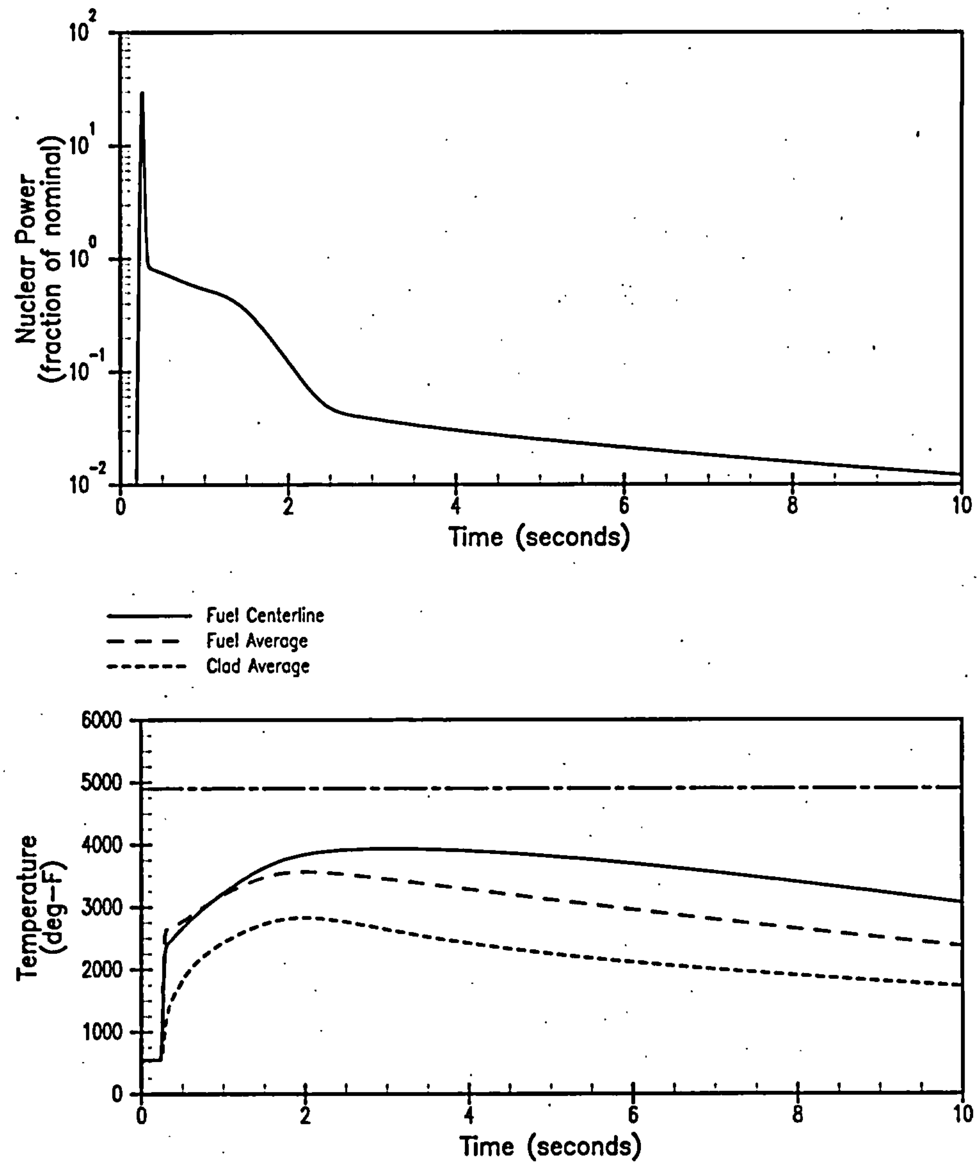


Figure 2.8.5.4.6-2
Rod Ejection – BOL/HZP Case
OFA Design

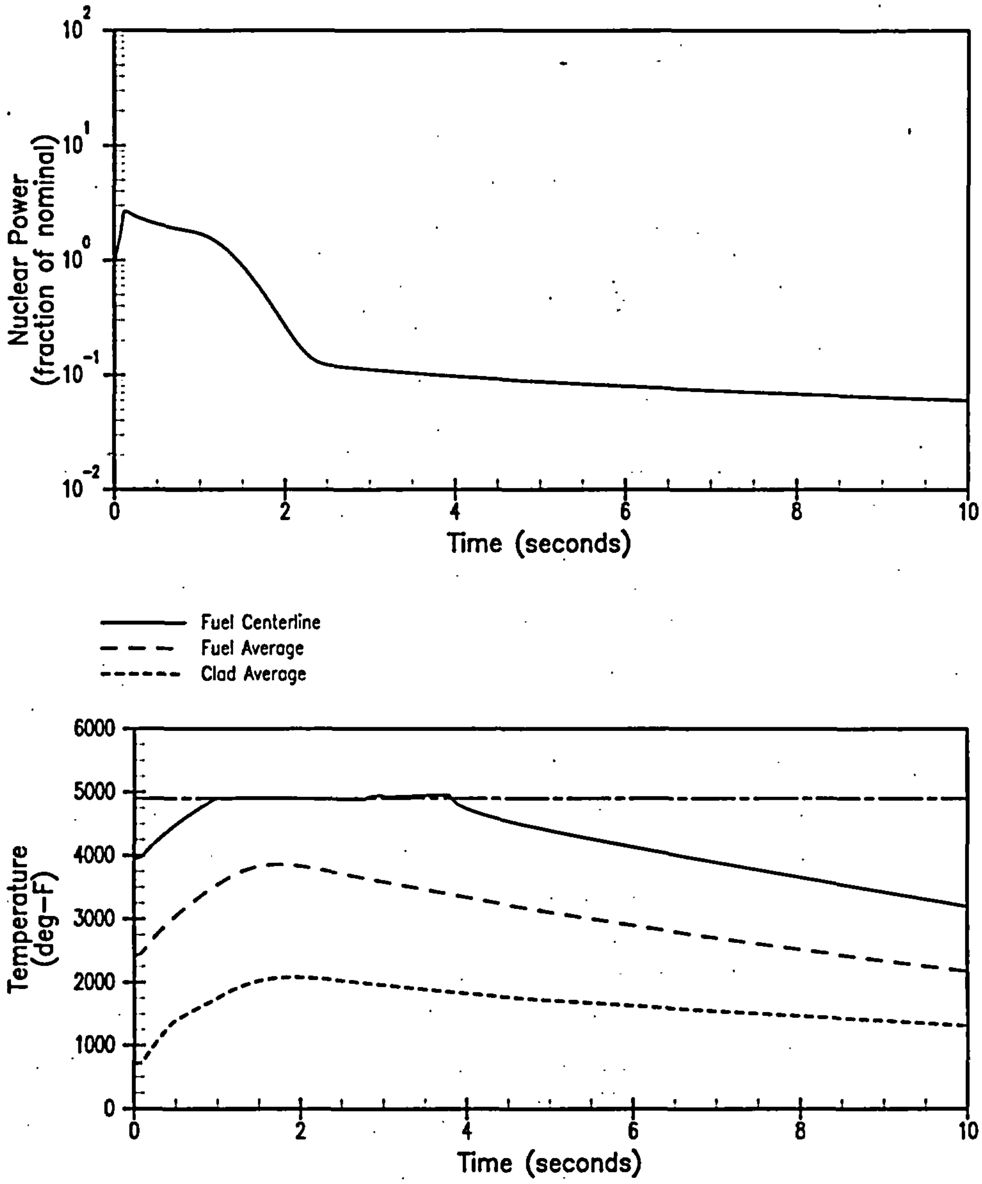


Figure 2.8.5.4.6-3
Rod Ejection – BOL/HFP Case
V+ Design

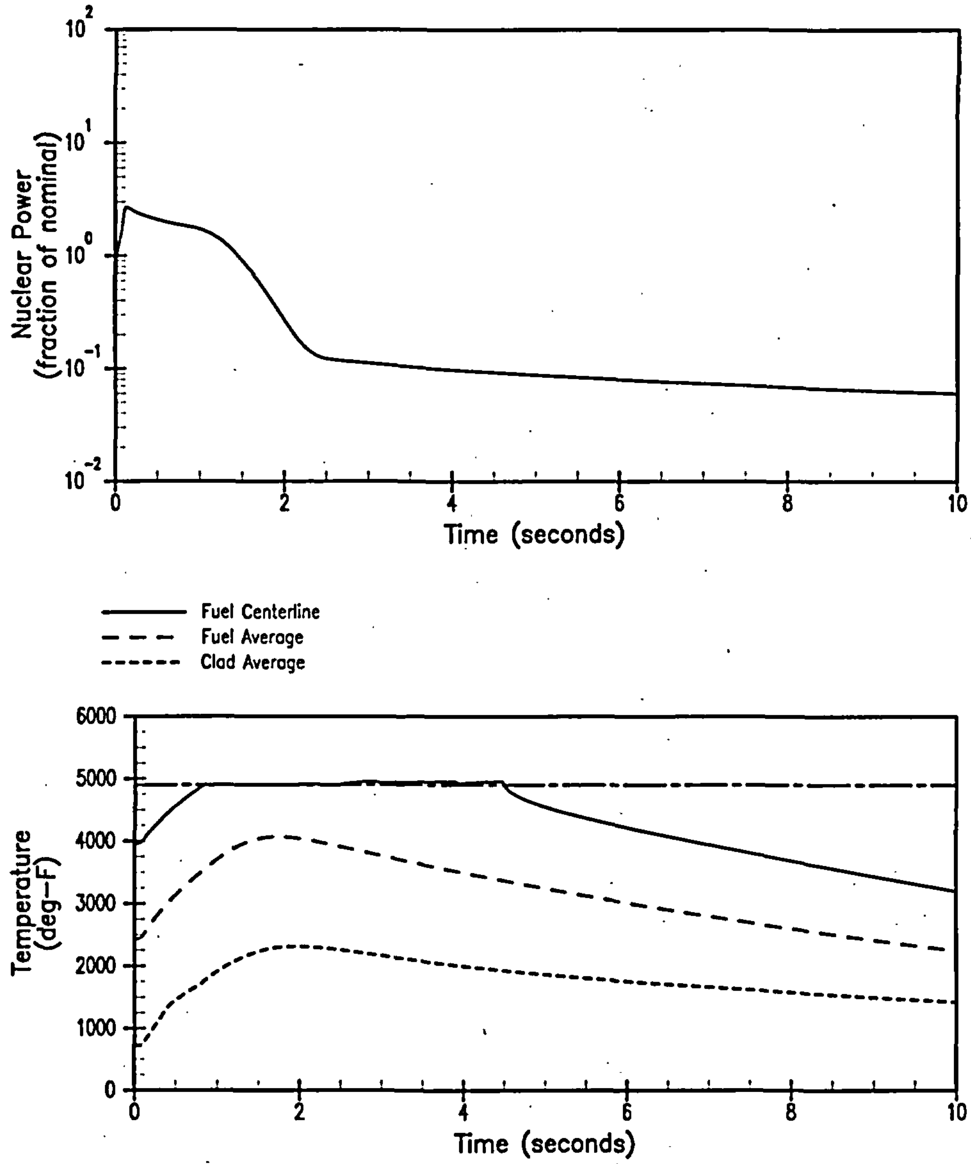
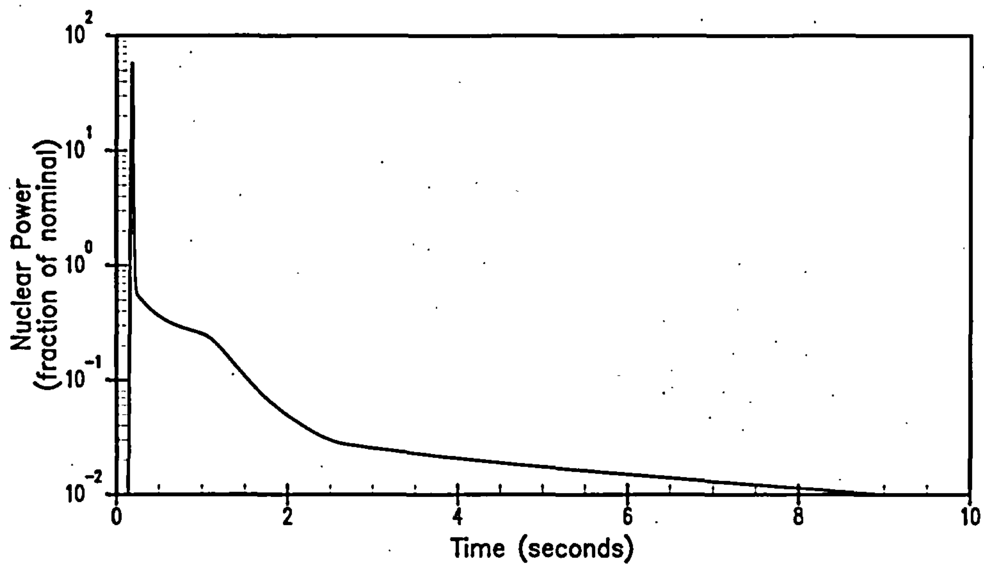


Figure 2.8.5.4.6-4
Rod Ejection – BOL/HFP Case
OFA Design



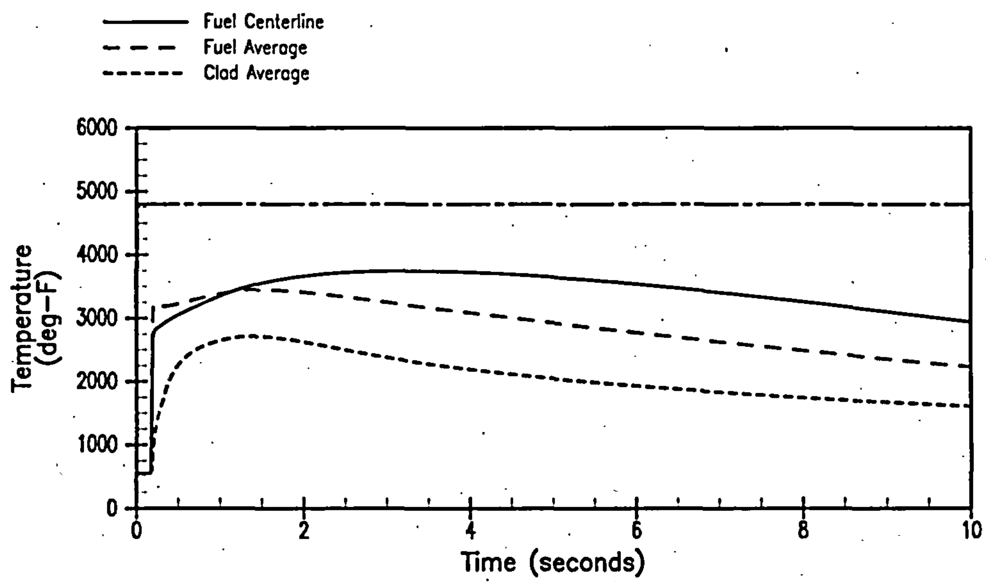


Figure 2.8.5.4.6-5
Rod Ejection – EOL/HZP Case
V+ Design

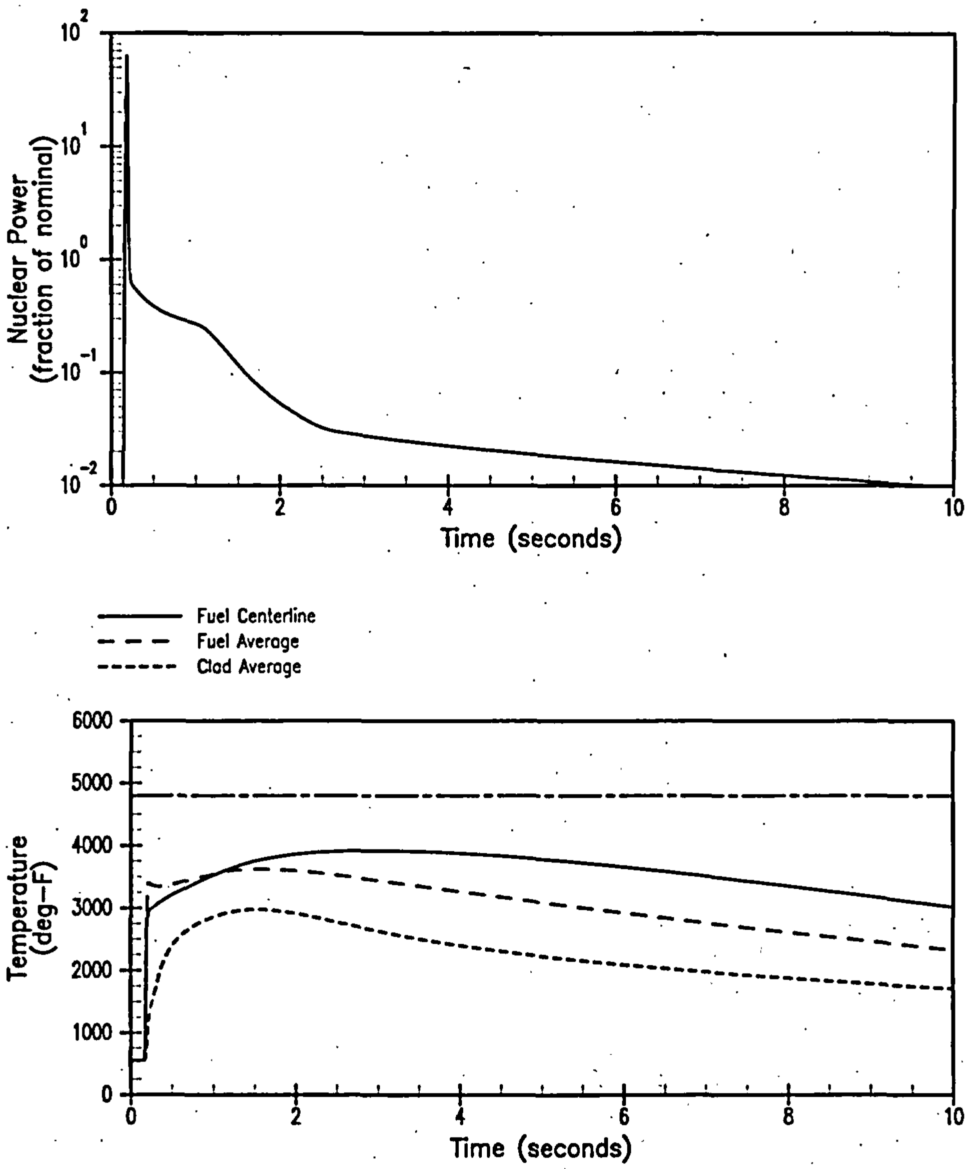


Figure 2.8.5.4.6-6
Rod Ejection – EOL/HZP Case
OFA Design

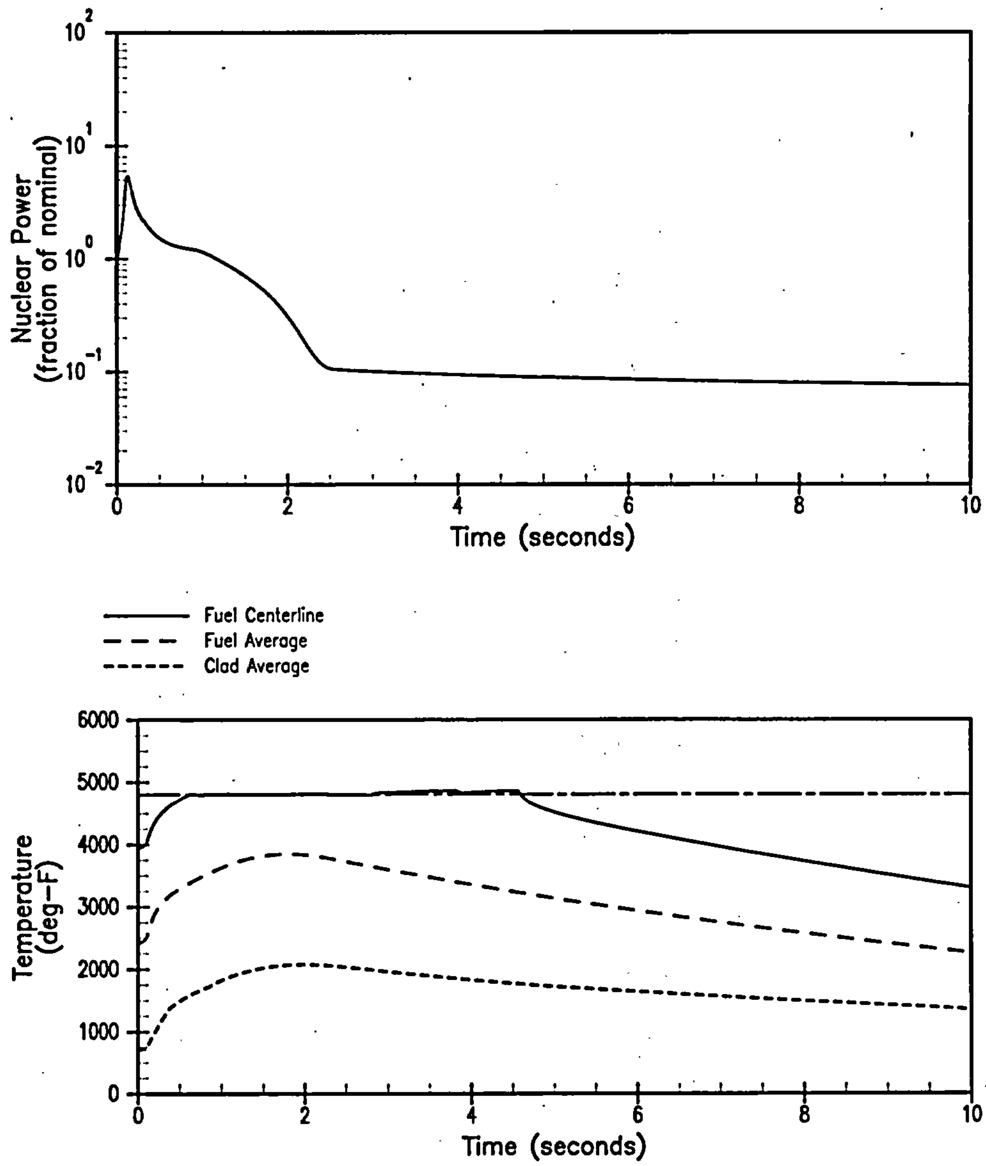


Figure 2.8.5.4.6-7
Rod Ejection – EOL/HFP Case
V+ Design

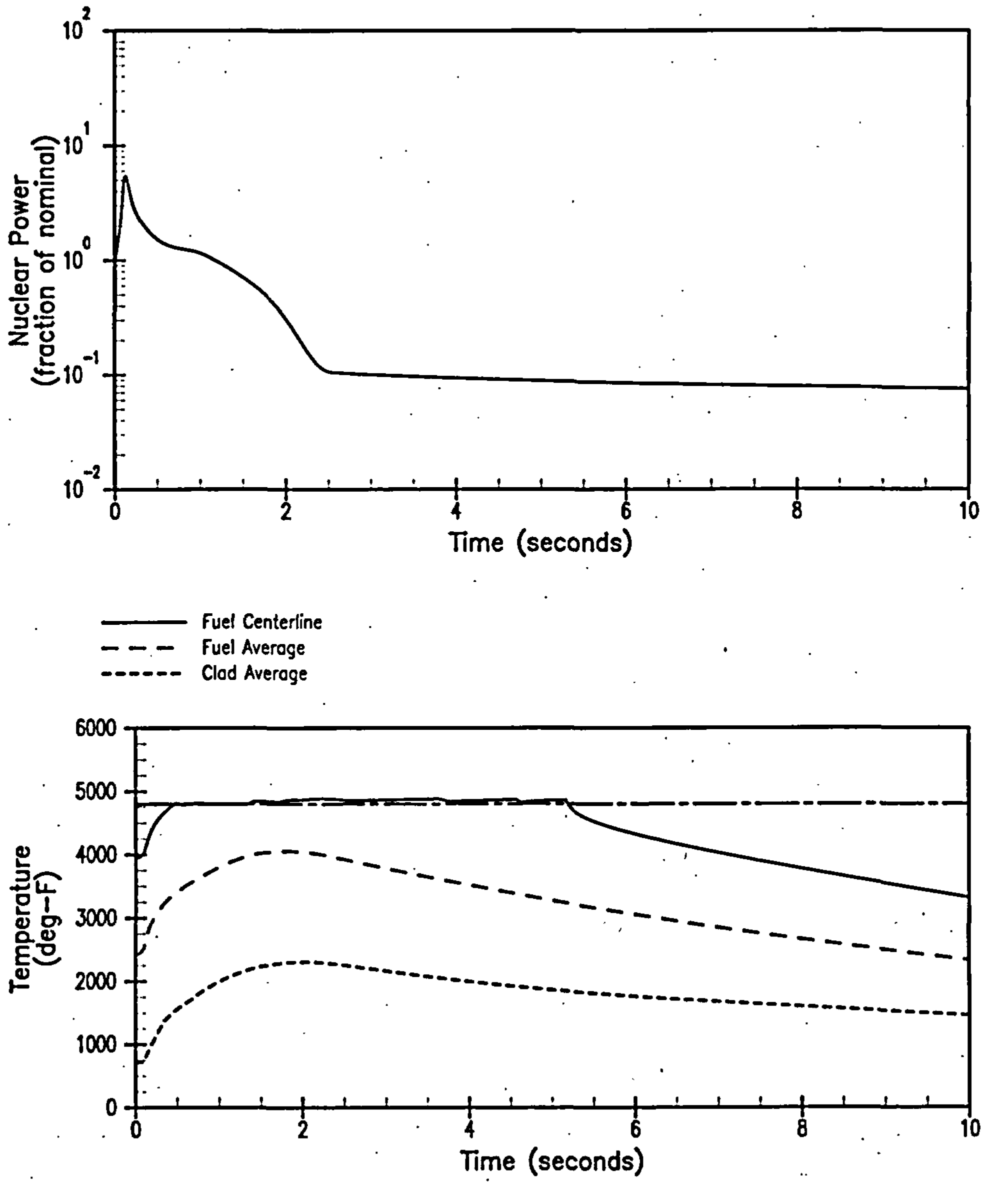


Figure 2.8.5.4.6-8
Rod Ejection – EOL/HFP Case
OFA Design

2.8.5.4.6.5 Conclusion

The Ginna staff has reviewed the analyses of the rod ejection accident and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. The Ginna staff further concludes that the analyses have demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding, or cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the Ginna staff concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-28 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the rod ejection accident.

2.8.5.5 Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

2.8.5.5.1 Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the reactor coolant system (RCS). Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the inadvertent actuation of the emergency core cooling system (ECCS) or chemical and volume control system (CVCS) malfunction for purposes of determining if an increase in reactor coolant inventory that could lead to an increase in RCS pressure and pressurizer level could occur.

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including anticipated operational occurrences.
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences.
- GDC-26, insofar as it requires that a reactivity control system be provided and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, SAFDLs are not exceeded.

Specific review criteria are contained in SRP section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Ginna Current Licensing Basis

Potential events that could increase RCS inventory and lead to RCS overpresurization and increase in pressurizer level have been analyzed as described in UFSAR section 15.5.

ECCS Operation

The emergency core cooling system at Ginna Station is the safety injection system (SIS). During power operations, the high pressure SIS pumps are incapable of delivering flow to the RCS because the pumps' shut-off head is 1500 psi which is considerably less than the nominal 2235 psig operating pressure of the RCS. Therefore, an inadvertent ECCS event which could overpressurize the RCS is not credible for Ginna.

CVCS Malfunction

The CVCS contains three positive displacement charging pumps which can deliver a maximum total flow of 180 gpm (60 gpm per pump). Normal charging flow is maintained at approximately 46 gpm. In the event that charging flow becomes excessive relative to the RCS inventory make-up requirement, there are alarms to alert the operator to high pressurizer level, high pressurizer pressure, and low volume control tank (CVCS make-up inventory source) level. Reactor trip would occur on high pressurizer pressure or level. The nominal steam volume in the pressurizer is 397 ft^3. It would take several minutes to fill this volume at normal charging flow; thus, the operator would have adequate time and indication to terminate the event. In addition, during normal operation automatic plant protection features, such as Pressurizer Power Operated Relief Valves (PORV) and safety valves, are also available to provide overpressure protection and assist in control of inventory as described in UFSAR section 5.2.2.1. Low temperature overpressure protection of the reactor coolant pressure boundary (RCPB) is described in UFSAR section 5.2.2.2.

2.8.5.5.2 Technical Evaluation

The Ginna staff has evaluated the two potential ECCS and CVCS events that could lead to overpressurization of the RCPB:

- The high pressure SIS pumps remain incapable of delivering water to the RCS at sufficient pressure to cause an overpressure event after EPU, therefore, this event initiator is not applicable, and

- The nominal steam volume in the pressurizer is reduced to 333 ft^3 for EPU at the high end of the allowable Tavg range as a result of the required change in pressurizer level program (see LR section 2.4.1). It would take several minutes to fill this volume at normal charging flow. The high pressurizer level, high pressurizer pressure and low volume control tank level alarm setpoints are not affected by EPU. The high pressurizer level and high pressurizer pressure reactor trip setpoints are also not affected by EPU. Therefore, the operator would still have adequate time and indication to terminate the event, and automatic plant protection features are adequate. In addition, automatic plant protection features such as PORVs and safety valves would also be available to control inventory and pressure increases. Overpressure protection during normal operation has been evaluated in LR section 2.8.4.2 and found to be acceptable. Overpressure protection during low temperature operation has been evaluated in LR section 2.8.4.3 and found to be acceptable.

2.8.5.5.3 Conclusion

The Ginna staff has reviewed the evaluation of inadvertent operation of the ECCS and CVCS malfunction events and concludes that the evaluation has adequately accounted for operation of the plant at the proposed power level. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as the result of these events. Based on this, the Ginna staff concludes that the Ginna Station will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS and CVCS events.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Pressurizer Pressure Relief Valve Opening

2.8.5.6.1.1 Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. A reactor trip normally occurs due to low reactor coolant system pressure. Ginna Nuclear Power Plant, LLC's (Ginna) review covered:

- The sequence of events
- The analytical model used for analyses
- The values of parameters used in the analytical model
- The results of the transient analyses

NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences

- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be 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

- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Specific review criteria are contained in SRP, Section 15.6.1, and other guidance provided in Matrix 8 of RS-001.

Ginna Current Licensing Basis

As noted in Ginna Updated Final Safety Analysis Report (UFSAR), the GDC used during the licensing of Ginna Station predate those provided today in 10CFR50, Appendix A. However, for this event, the analyses performed demonstrate that the requirements specified by the GDC in 10CFR50, Appendix A are met.

The analysis of an inadvertent opening of a pressurizer pressure relief valve event is described in UFSAR 15.6.1

The specific acceptance criteria applied by Ginna for this event are as follows:

- The departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met satisfies the Ginna Station current licensing basis requirements with respect to GDC-10.
- Primary and secondary pressures must remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-15.
- GDC-26 requires reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, is accounted for in the safety analysis assumptions. Demonstrating that the fuel design limits (i.e., DNBR) are met satisfies the Ginna Station current licensing basis requirements with respect to GDC-26.

In addition to the evaluations described in the Ginna UFSAR, the pressurizer pressure relief valves 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 analysis of an inadvertent opening of a pressurizer pressure relief valve is not within the scope of license renewal.

2.8.5.6.1.2 Technical Evaluation

2.8.5.6.1.2.1 Introduction

An accidental depressurization of the RCS and a decrease in reactor coolant inventory, could occur as a result of an inadvertent opening of a pressurizer relief valve. To conservatively bound this scenario, the Westinghouse methodology models the failure of a pressurizer safety valve since a safety valve is sized to relieve approximately 40% more steam flow than a relief valve and will allow a much more rapid depressurization upon opening. This yields the most severe core conditions resulting from an accidental depressurization of the RCS. Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot-leg saturation conditions without reactor protection system intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably. However, the pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via the moderator density feedback. However, if the plant is in the automatic mode, the rod control system functions to maintain the power essentially constant throughout the initial stages of the transient. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip because of the decreased reactor coolant density.

The reactor may be tripped by the following reactor protection system signals:

- Low pressurizer pressure
- Overtemperature ΔT

2.8.5.6.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

To produce conservative results in calculating the DNBR during the transient, the following assumptions were made:

- The accident was analyzed using the Revised Thermal Design Procedure (Reference 1). Initial core power, RCS pressure, and RCS temperature were assumed to be at their nominal values, consistent with steady-state full-power operation. Minimum measured flow was modeled. Uncertainties in initial conditions were included in the DNBR limit as described in Reference 1. The initial core power level assumed is 1811 MWt.

- A zero moderator coefficient of reactivity was assumed. This is conservative for beginning-of-life (BOL) operation in order to provide a conservatively low amount of negative reactivity feedback due to changes in moderator temperature.

- A small (absolute value) Doppler coefficient of reactivity is assumed, such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator feedback.

- The spatial effect of voids resulting from local or subcooled boiling was not considered in the analysis with respect to reactivity feedback or core power shape. In fact, it should be noted, the power peaking factors were kept constant at their design values, while the void formation and resulting core feedback effects would result in considerable flattening of the power distribution. Although this would significantly increase the calculated DNBR, conservatively, no credit was taken for this effect.

Based on its frequency of occurrence, the accidental depressurization of the RCS accident was considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This was met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. Note that since this event is a depressurization event, these limits are not challenged. Both primary and secondary pressures decrease for the entire duration of the event.
- Fuel design limits should not be exceeded.

2.8.5.6.1.2.3 Description of Analyses and Evaluations

The purpose of this analysis was to demonstrate that the reactor protection system functions and mitigates the consequences of the RCS depressurization event.

The accident was analyzed by using the detailed digital computer code RETRAN (Reference 2). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

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

The analysis of an inadvertent opening of a pressurizer pressure relief valve is not within the scope of license renewal. LR section 2.2.2.7 provides a description of the impact of pressurizer pressure relief valves on license renewal evaluations. No systems or components are being added or modified as the result of this analysis for EPU conditions. The analysis described in this LR section involves 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 the analysis of an inadvertent opening of a pressurizer pressure relief valve.

2.8.5.6.1.3 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 2.8.5.6.1-1 through 2.8.5.6.1-4. Figure 2.8.5.6.1-1 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly until the reactor trip occurs on overtemperature ΔT (OTΔT). The pressurizer pressure transient is illustrated in Figure 2.8.5.6.1-2. Pressure decreases continuously throughout the transient; however, pressure decreases more rapidly after core heat generation is reduced via the reactor trip. If the saturation temperature is reached in the hot leg, the pressure decrease slows. Also illustrated in Figure 2.8.5.6.1-3 is the loop average temperature transient. The loop average temperature is maintained at approximately the initial value until the reactor trip occurs. The DNBR decreases initially, but increases rapidly following the reactor trip as demonstrated in Figure 2.8.5.6.1-4. The DNBR remains above the limit value of 1.38 throughout the transient.

The calculated sequence of events is shown in Table 2.8.5.6.1-1.

Because different analysis methods were applied, a comparison of results between the current licensing basis analysis and the EPU analysis is not appropriate. The current licensing basis relies on generic analyses performed by Westinghouse (Reference 3) in response to post-Three Mile Island requirements for the inadvertent opening of pressurizer power operated relief valves (PORV). In that generic analysis, two transients with breaks in the pressurizer vapor space were analyzed, a 0.008 ft^2 and a 0.034 ft^2 break. The 0.008 ft^2 break closely represents the flow area of one PORV of a typical Westinghouse plant. The other break area is approximately the flow area of three pressurizer PORVs and would cause the largest insurge of flow to the pressurizer. This envelopes the Ginna design which has two PORVs. The generic analysis showed that in no case did the core uncover. In comparison, as discussed above, a pressurizer safety valve with a full open area of 0.017 ft^2 was conservatively postulated to fail open in the analysis performed in support of the EPU. Because the results of the conservatively large 0.034 ft^2 break analysis are acceptable, the results of the opening of the safety valve (0.017 ft^2) are bounded and also acceptable.

The results of the analysis show that the OTΔT reactor protection system function provides adequate protection against the RCS depressurization event since the minimum DNBR remains above the safety analysis limit throughout the transient. Therefore, no cladding damage or release of fission products to the RCS is predicted for this event.

The results of the analysis performed for the accidental depressurization of the RCS for the NSSS power of 1817 MWt are bounded by the previous analysis at the current licensed power level, and support the implementation of the extended power uprate at the Ginna Station.

2.8.5.6.1.4 References

1. WCAP-11397-P-A, *Revised Thermal Design Procedure*, A. J. Friedland and S. Ray, April 1989.

2. WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor NON-LOCA Safety Analyses*, April 1999.

3. WCAP-9600, *Report on Small Break Accidents for Westinghouse NSSS System,* June 1979.

Table 2.8.5.6.1-1 Time Sequence of Events – Accidental Depressurization of the RCS	
Event	**Time (sec)**
Inadvertent opening of one RCS relief valve	0.0
OTΔT reactor trip setpoint reached	20.9
Rods begin to drop	22.9
Minimum DNBR occurs	23.0

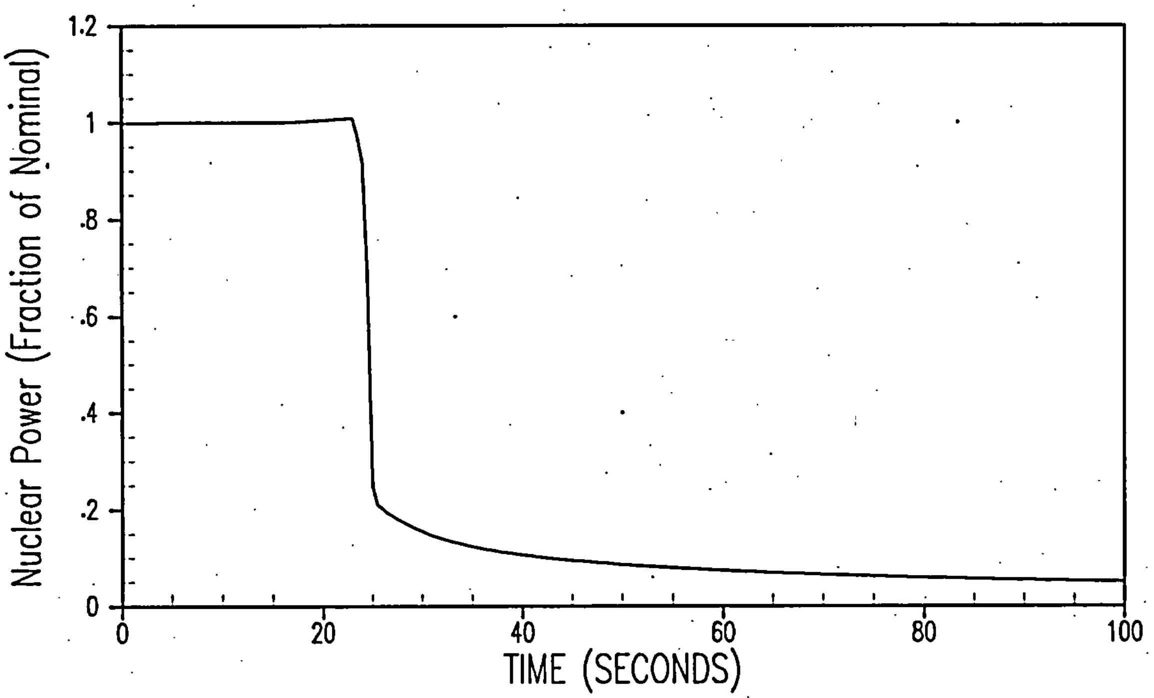


Figure 2.8.5.6.1-1
RCS Depressurization
Nuclear Power vs. Time

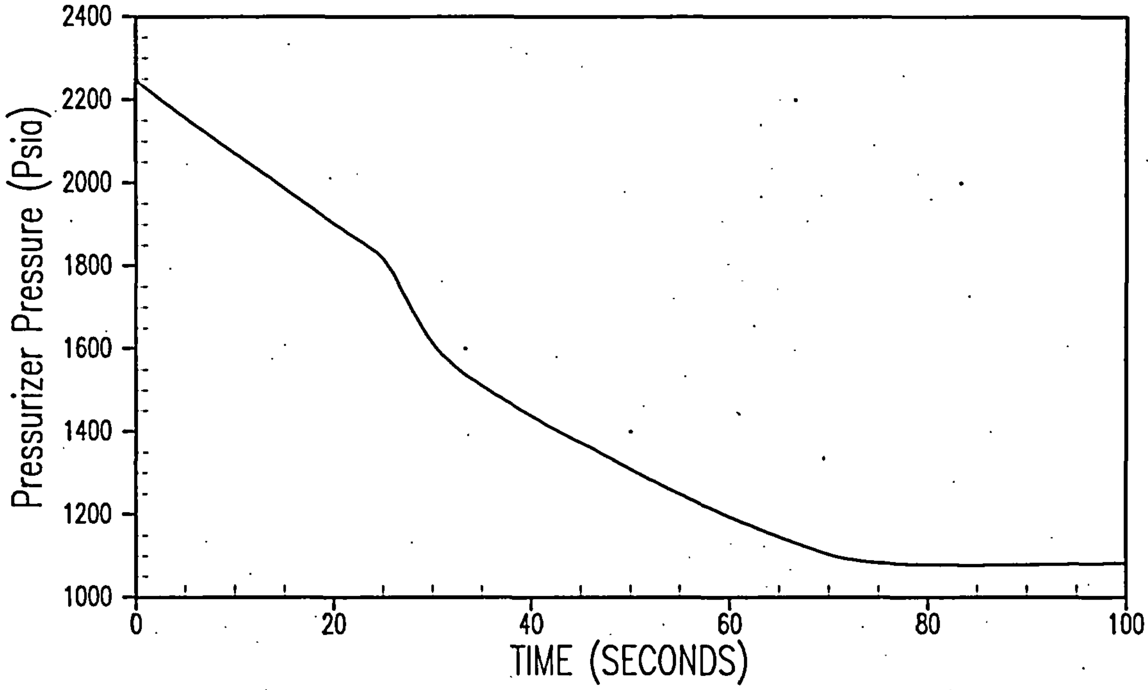


Figure 2.8.5.6.1-2
RCS Depressurization
Pressurizer Pressure vs. Time

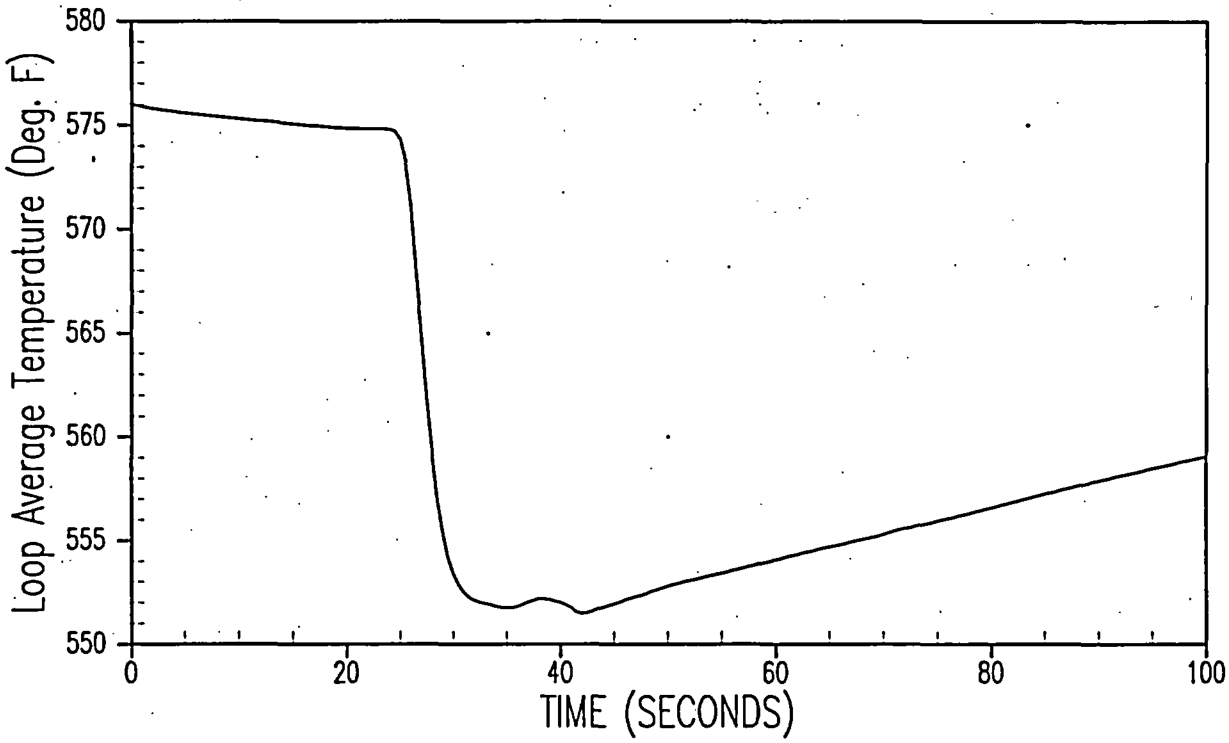


Figure 2.8.5.6.1-3
RCS Depressurization
Indicated Loop Average Temperature vs. Time

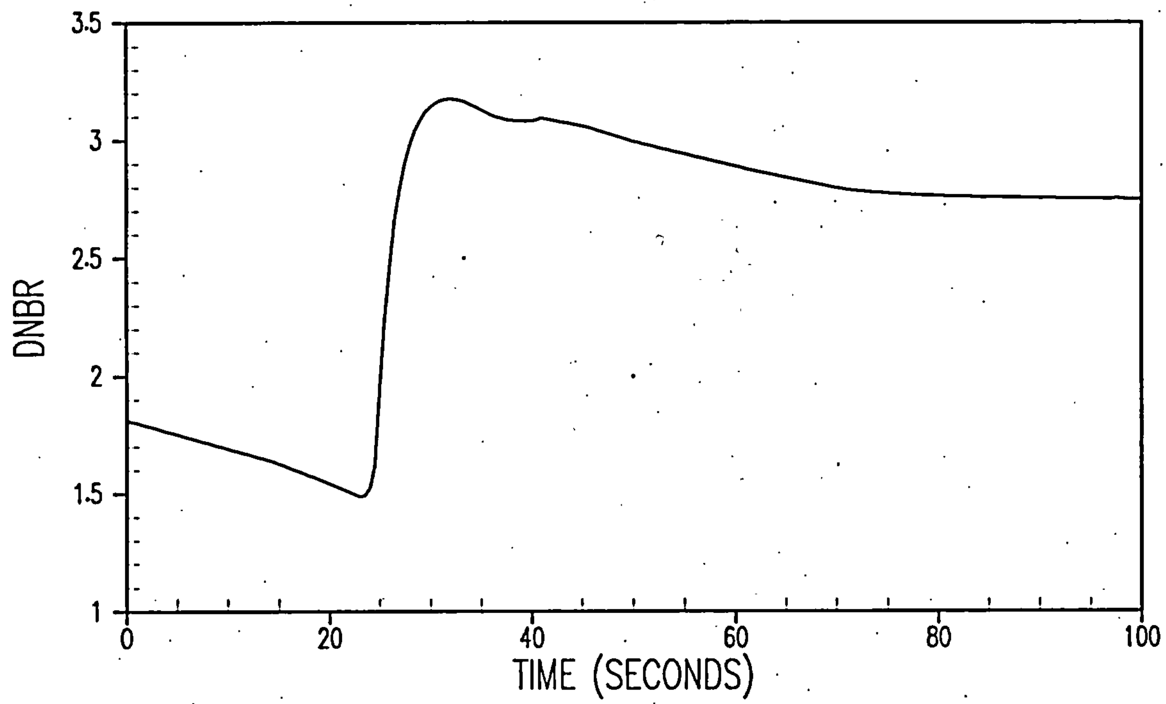


Figure 2.8.5.6.1-4
RCS Depressurization
DNBR vs. Time

2.8.5.6.1.5 Conclusion

The Ginna staff has reviewed the analysis of the inadvertent opening of a pressurizer pressure relief valve event and concludes that the analysis has adequately accounted for plant operation at the uprated power level and was performed using acceptable analytical models. Ginna further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, Ginna concludes that the plant will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-10, GDC-15, and GDC-26 following implementation of the proposed EPU. Therefore, Ginna finds the proposed EPU acceptable with respect to the inadvertent opening of a pressurizer pressure relief valve event.

2.8.5.6.2 Steam Generator Tube Rupture

2.8.5.6.2.1 Regulatory Evaluation

A steam-generator-tube-rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and main steam safety valves (MSSVs) or atmospheric relief valves (ARVs). Reactor protection and engineered safety features (ESFs) are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10CFR100. The Ginna Nuclear Power Plant, LLC (Ginna) staff's review covered:

- The postulated initial core and plant conditions,
- The method of thermal-hydraulic analysis,
- The sequence of events (assuming offsite power either available or unavailable),
- The assumed reactions of reactor system components,
- The functional and operational characteristics of the reactor protection system,
- The operator actions consistent with the plant's emergency operating procedures; and
- The results of the accident analysis

A single failure of a mitigating system is assumed for this event.

The NRC staff's review of the SGTR is focused on the thermal and hydraulic analyses for the SGTR in order to

- Determine whether 10CFR100 is satisfied with respect to radiological consequences, which are discussed in section 2.9.6 of this safety evaluation and
- Confirm that the faulted SG does not experience an overfill. Preventing SG overfill is necessary in order to prevent the failure of main steam lines.

Specific review criteria are contained in SRP section 15.6.3 and other guidance provided in Matrix 8 of RS-001, Revision 0.

Ginna Current Licensing Basis

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR) section 15.6.3, "Steam Generator Tube Rupture," the SGTR accident analysis includes analyses performed to demonstrate margin-to-overfill and analyses to ensure that possible radiological dose consequences are within allowable guidelines. The dose analysis requires thermal-hydraulic calculations be performed to determine the amount of reactor coolant discharged to the ruptured steam generator and the amounts of steam released from the steam generators. The UFSAR analyses were analyzed following the methodology of WCAP-10698 and Supplement 1 to WCAP-10698, (References 1 and 2), using the LOFTTR2 code to calculate the margin-to-overfill and mass-release data. The methodology and the code were developed by the Westinghouse Owners Group (WOG) SGTR subcommittee that was formed to address NRC questions regarding assumptions used in SGTR safety analyses that arose as a result of the January 1982 SGTR event at the Ginna Station. The LOFTTR2 code is described in WCAP-11002 (Reference 3). LOFTTR2 is identical to the LOFTTR1 code described in WCAP-10698 and Supplement to WCAP-10698 (References 1 and 2), with the exception that it has the additional capability to model overfill conditions.

Ginna Station has received approval to implement the alternate source term dose calculation methodology. With this approval, the dose criteria in 10CFR50.67 became the licensing basis for all subsequent radiological consequences analyses. Ginna's staff review of the SGTR is focused on the thermal-hydraulic analysis for the SGTR in order to determine whether 10CFR50.67 is satisfied with respect to radiological consequences, which are discussed in LR section 2.9.6, "Radiological Consequences of a Steam Generator Tube Rupture," and confirm that the ruptured steam generator does not experience an overfill. Preventing steam generator overfill is necessary to prevent the release of water to the environment through the MSSVs or ARVs and to preclude the possibility of failure of main steam lines.

In addition to the evaluations described in the Ginna UFSAR, the components associated with the SGTR analysis 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.

2.8.5.6.2.2 Technical Evaluation

The evaluation of the design basis SGTR event demonstrated that the current design is acceptable to support the EPU operation.

2.8.5.6.2.2.1 Introduction

The SGTR analysis is described in Ginna UFSAR, Section 15.6.3, Steam Generator Tube Rupture." The SGTR accident analysis included analyses performed to demonstrate margin-to-overfill and analyses to ensure that possible radiological dose consequences are within allowable guidelines. The dose analysis required thermal-hydraulic calculations be performed to determine the amount of reactor coolant discharged to the ruptured steam generator, and the amounts of steam released from the steam generators. The effects of limiting single failures and the times for required operator actions were explicitly included in the analyses. Typically, it is not known beforehand which end of the average temperature (T_{avg}) window and steam generator tube plugging conditions give the bounding result for each type of analysis considered. Therefore, four cases were analyzed separately for the margin-to-overfill and mass-release analyses. All cases were analyzed with a loss-of-offsite power. Only the results of the limiting margin-to-overfill and mass-release cases are presented in the UFSAR.

The EPU analyses were performed using the methodology employed in the Ginna UFSAR but with the RETRAN-02 computer code. As documented in WCAP-14882 (Reference 4) the NRC has approved the use of the Westinghouse RETRAN-02 models to replace the use of LOFTTR2.

The analysis included an analyzed core power level of 1811 MWt, and a full-power T_{avg} operating range from 564.6° to 576.0°F and up to 10% steam generator tube plugging as well as a main feedwater temperature range from 390° to 435°F. Consistent with Ginna UFSAR, Section 15.6.3, four cases were analyzed separately for the margin-to-overfill and mass-release analyses, to consider the range of T_{avg} and tube plugging. All margin-to-overfill cases considered the low main feedwater temperature to maximize the initial secondary side water mass, and all mass-release cases considered the high-feedwater temperature to minimize the initial secondary side water mass. All cases were analyzed with a loss-of-offsite power.

The margin-to-overfill transient was analyzed until the ruptured steam generator secondary side and reactor coolant system (RCS) pressures equalized, at which time the ruptured tube flow was considered isolated. The four cases to determine the minimum margin-to-overfill were:

Case 1	Margin-to-overfill with 10% steam generator tube plugging level and T_{avg} at 576.0°F
Case 2	Margin-to-overfill with 0% steam generator tube plugging level and T_{avg} at 576.0°F
Case 3	Margin-to-overfill with 10% steam generator tube plugging level and T_{avg} at 564.6°F
Case 4	Margin-to-overfill with 0% steam generator tube plugging level and T_{avg} at 564.6°F

The mass-release cases determine the primary-to-secondary break flows and steam releases for the SGTR radiological consequences analysis. These cases are analyzed through tube rupture flow isolation and cooldown to residual heat removal system (RHRS) in-service conditions to obtain the total steam releases from the intact and ruptured steam generators. (At this point the plant proceeds to MODE 5 [cold shutdown] conditions without additional steam release using the RHRS.) The four mass-release cases are:

Case 5	Mass release with 10% steam generator tube plugging level and T_{avg} at 576.0°F
Case 6	Mass release with 0% steam generator tube plugging level and T_{avg} at 576.0°F
Case 7	Mass release with 10% steam generator tube plugging level and T_{avg} at 564.6°F
Case 8	Mass release with 0% steam generator tube plugging level and T_{avg} at 564.6°F

For the EPU, as with the analysis presented in Ginna UFSAR section 15.6.3, higher tube plugging and low T_{avg} are limiting for the margin-to-overfill calculation. For the mass-release calculation the higher tube plugging and higher Tavg are limiting for the EPU. It is noted that the differences in mass-release data between the four cases considered in the Ginna UFSAR are small and does require a major change in the transients for the limiting case to change. Only the results of the limiting margin-to-overfill and mass-release cases (Cases 3 and 5, respectively) are presented.

The radiological consequences analysis is presented in LR section 2.9.6, "Radiological Consequences of a Steam Generator Tube Rupture."

2.8.5.6.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet-cold-leg-side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary-to-secondary leakage than a break on the hot side of the steam generator. However, the break flow flashing fraction was conservatively calculated for use in the radiological consequences analysis assuming that all of the break flow came from the hot-leg side of the steam generator. The combination of these conservative assumptions regarding the break location results in a conservative calculation of the radiological consequences. It was also assumed that loss-of-offsite power occurred at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. Due to the assumed loss-of-offsite power, the condenser was not available for steam releases once the reactor was tripped. Consequently, after reactor trip, steam was released to the atmosphere through the steam generator ARVs.

Single Failure Considerations

The effects of single failures in margin-to-overfill and mass-release analyses were investigated in WCAP-10698 and its Supplement 1 (references 1 and 2). The limiting single failures for the Ginna Station SGTR analyses are described below.

The limiting single failure for margin-to-overfill considerations is the ARV failing closed on the intact steam generator (reference 1). The ARV on the intact steam generator must be locally opened before the RCS cooldown can begin. The additional time to open the ARV delays the depressurization of the RCS, causing an increase in the amount of reactor coolant discharged to the secondary side of the ruptured steam generator.

The limiting single failure for the mass-release analysis for the Ginna Station is the ARV failing open on the ruptured steam generator (reference 2). Failure of this ARV causes an uncontrolled depressurization of the ruptured steam generator resulting in increased primary-to-secondary flow. Pressure in the ruptured steam generator remains less than the RCS until the failed ARV is isolated and recovery actions are completed.

Operator Actions Assumed

Important operator actions in the WOG Emergency Response E-3 Guidelines were explicitly modeled in the analysis. These actions were intended to terminate flow though the SGTR before proceeding to long-term cooldown. The operator actions modeled in the EPU analysis and the associated times were consistent with those currently incorporated in the analyses presented in Ginna UFSAR section 15.6.3.

The times required to perform the major recovery actions modeled in the SGTR analyses performed for the EPU were unchanged from those included in Ginna UFSAR section 15.6.3. These action times consisted of two components: initiation times (for the operator to start actions) and plant/system response times (for the plant conditions to reach performance objectives such as temperature, pressure, flow, etc., required by the recovery action). The latter times were determined from the thermal-hydraulic transient analyses of the SGTR accident. The operator action times are summarized in Table 2.8.5.6.2-1.

The operator actions that were modeled include:

Identifying the ruptured steam generator.

> Several means are available to the operator. The predominant indications are an unexpected rapid increase in the ruptured steam generator's narrow range level following the reactor trip, high radiation from a steam generator blowdown radiation monitor, or high radiation from a steam line radiation monitor.

Isolating the steam flow from the ruptured steam generator and throttling auxiliary feedwater flow to the ruptured steam generator.

Isolating the ruptured steam generator minimizes radiological releases and reduces the possibility of overfilling by minimizing the accumulation of feedwater. This action also enables the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary-to-secondary flow. It was assumed that the ruptured steam generator would be isolated when the level in the steam generator reached between being just on span and 50% on the narrow range instrument (modeled as 33% narrow range level), or after an operator action time of 10 minutes, whichever was longer.

Cooling down the RCS by dumping steam from the intact steam generator.

The RCS is cooled down as rapidly as possible to a temperature less than the saturation temperature corresponding to the ruptured steam generator's pressure. The cooldown is performed using the intact steam generator's ARV since neither the steam dump valves nor the condenser were available following the assumed loss-of-offsite power. The cooldown continues until RCS subcooling at the ruptured steam generator pressure is 20°F, plus an allowance of 18°F for instrument uncertainty.

Depressurizing the RCS after cooldown to minimize break flow and restore pressurizer level.

After the RCS cooldown, safety injection is terminated since it is the principal contributor to tube rupture flow. Depressurizing the RCS is required to ensure an adequate RCS inventory and reliable pressurizer level indication prior to stopping injection. Since offsite power was assumed to be lost at the time of reactor trip, the reactor coolant pumps were not running, and thus normal pressurizer spray was not available. It was assumed that the operator depressurized the RCS using a pressurizer power-operated relief valve (PORV). The operator continues to depressurize until any of the following is satisfied:

- RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 5% (0% plus 5% allowance for level uncertainty), or
- Pressurizer level is greater than 75% (80% minus 5% allowance for level uncertainty), or
- RCS subcooling is less than the 18°F allowance for subcooling instrument uncertainty.

Terminating safety injection to prevent re-pressurization of the RCS and terminate primary to secondary flow.

Safety injection is terminated when all of the following are satisfied:

- The RCS pressure stabilizes or started to increase.
- The RCS subcooling is greater than the 18°F allowance for subcooling instrument uncertainty.
- The minimum auxiliary feedwater flow is available or the intact steam generator level is in the narrow range.
- The pressurizer level is greater than the 5% allowance for level uncertainty.

Additional operator actions are required to recover from the single failures postulated for the margin-to-overfill and mass-release analyses. These operator actions that occur outside the control room include locally opening the intact steam generator ARV and locally closing the intact steam generator ARV block valve (for the margin-to-overfill analysis), as well as locally closing the ruptured steam generator ARV block valve (for the mass-release analysis). The times associated with performing these operator actions are listed in Table 2.8.5.6.2-2. It is noted that the 20 minutes required to open the intact steam generator ARV (in the margin-to-overfill case) consists of 10 minutes to identify and locate the valve, and 10 minutes to open the valve linearly. Due to limitations of LOFTTR2, the analysis presented in Ginna UFSAR section 15.6.3 modeled the operator action to open the intact steam generator ARV as a step open after a 15-minute delay, which resulted in an equivalent integrated steam flow through the ARV at the end of the 20-minute period. RETRAN-02 allows valve modeling that more closely models the expected response. The RETRAN-02 analysis performed for the EPU modeled linear opening of the ARV over a 10-minute period, starting 10 minutes from the time of cooldown initiation. Similarly, the LOFTTR2 analysis, presented in Ginna UFSAR section 15.6.3, modeled closing the intact steam generator ARV block valve as a step closed 5 minutes from the time the cooldown target temperature was reached, while the RETRAN-02 analysis performed for the EPU modeled linear closing of the valve over 5 minutes.

Following termination of tube rupture flow, the operator is required to perform additional actions to bring the plant to MODE 5 (cold-shutdown) conditions. The operator actions are defined in the WOG E-3 Guidelines. Only two of the actions were explicitly considered in the analysis.

The operator is required to cool the RCS to the RHRS in-service temperature by feeding and steaming the intact steam generator. The SGTR long-term mass-release analysis assumed the operator performs this action by dumping steam to the atmosphere via the ARV. Although other preferable cooldown methods (such as steam dump to the condenser to minimize activity releases) are identified in the WOG Guidelines, steam dump to the atmosphere was necessary because offsite power was assumed to be lost at the time of reactor trip, causing the condenser to be unavailable.

Cooldown of the ruptured steam generator is performed after the RCS is cooled to the RHRS in-service temperature. With a loss-of-offsite power, the operator immediately releases steam from the ruptured steam generator to the atmosphere. (This method is conservative for radiological calculations since it maximizes the activity released from the plant.) The operator maintains equal pressure between the RCS and ruptured steam generator secondary side using the PORV as needed until the RHRS is brought online.

With the exception of being on residual heat removal (RHR) in 8 hours, explicit operator action times were not defined since cooldown can proceed more gradually after tube rupture flow is terminated.

Input Parameters and Initial Conditions

Parameters and initial conditions common to the margin-to-overfill and mass-release analyses were:

The plant was at 100% rated thermal power, operating at the high (576.0°F), or low (564.6°F) end of the T_{avg} window, depending on the case analyzed. Other initial conditions are summarized in Table 2.8.5.6.2-3.

The highest worth rod cluster control assembly was stuck in its fully withdrawn position at reactor trip.

Reactor trip occurred when the overtemperature ΔT setpoint was reached. No reactor trip delay was assumed since it maximized the secondary side inventory in the ruptured steam generator and steam releases from both steam generators. It was also assumed that loss-of-offsite power occurred at the time of reactor trip.

The turbine automatically tripped following a reactor trip. Zero delay was assumed since it minimized the steam flow to the turbine, and maximized the secondary side water inventory in the ruptured steam generator and steam releases from both steam generators.

The condenser was unavailable for steam dump following reactor trip due to the assumed loss-of-offsite power. All subsequent steam relief was through the ARVs, and MSSV, if needed.

A low ARV setpoint of 1065 psia was used since control at lower steam generator pressures caused a greater primary-to-secondary side pressure differential and tube rupture flow.

Safety injection flow was from three safety injection pumps injecting into both reactor coolant loops (see Figure 2.8.5.6.2-1). This assumption conservatively increased the break flow through the ruptured tube.

Auxiliary feedwater from all three preferred pumps was automatically started following reactor trip and loss-of-offsite power. The flow was equally split between the steam generators, which were at nearly equal pressures until isolation.

Operation of charging and letdown systems and pressurizer heaters were not credited. Operating these systems delays the reactor trip, which reduces the severity of the analyzed transient.

Conservatively high decay heat rates were used. The increased heat input resulted in greater tube rupture flow after reactor trip due to the longer time needed for removing heat and depressurizing the RCS.

For the margin-to-overfill cases:

The initial water mass in both steam generators corresponded to 60% on the narrow range level. This mass represented the nominal (52%) steam generator water level at full power, with a +8% instrument uncertainty applied. A higher initial mass in the ruptured steam generator was conservative for reducing the margin to overfill. (The total fluid mass shown in Table 2.8.5.6.2-3 corresponded to T_{avg} at 564.6°F at full power, with 10% tube plugging level assumed and feedwater temperature of 390°F.)

The turbine runback on overtemperature ΔT at 10% per minute prior to reactor trip was simulated but not credited for delaying reactor trip. Turbine runback increased the secondary water mass with reduced load, because the feedwater controller attempts to maintain steam generator level as power decreased before the trip.

The ruptured steam generator's fluid mass was artificially increased to simulate a turbine runback to 89% power prior to trip. The mass modeled in the analysis corresponded to the initial maximum level at full power, plus the differential mass between 100% and 89% power.

The maximum auxiliary feedwater flow was modeled to maximize the mass of water in the ruptured steam generator at the time of isolation.

For the mass-release analyses:

A turbine runback was not assumed since it delays reactor trip. An earlier reactor trip results in greater steam releases to the atmosphere from both steam generators.

The steam generator water mass corresponded to 48% on the narrow range level. This mass represented the full-power, nominal steam generator water level with a -4% instrument uncertainty applied. A lower initial mass in the ruptured steam generator increases the predicted offsite doses. (The value shown in Table 2.8.5.6.2-3 corresponds to T_{avg} at 576.0°F with 10% tube plugging level assumed and feedwater temperature of 435°F.)

The minimum auxiliary feedwater flow was modeled to maximize the steam releases.

Acceptance Criteria

As noted in Ginna UFSAR section 15.6.3.4.3, no acceptance criteria are used for the margin-to-overfill and mass-release analyses. Both analyses are performed using conservative assumptions to demonstrate the ability of the operator to limit the system transient and establish parameters for providing a bounding radiological consequence assessment.

In order to demonstrate that water release from the ruptured steam generator did not have to be considered in the radiological consequences assessment, the margin-to-overfill analysis was performed to demonstrate that the secondary side of the ruptured steam generator did not completely fill with water. The available secondary side volume of a single Ginna Station steam generator is 4512.7 ft^3. Margin to overfill was demonstrated, provided the transient calculated steam generator secondary side water volume was less than 4512.7 ft^3.

The radiological consequences analysis acceptance criteria for the SGTR are discussed in LR section 2.9.6, "Radiological Consequences of a Steam Generator Tube Rupture."

2.8.5.6.2.2.3 Description of Analyses and Evaluations

The margin-to-overfill analyses were performed using the methodology in WCAP-10698 (Reference 1) with plant-specific parameters. The ruptured steam generator's secondary side water mass was calculated as a function of time to demonstrate that overfill did not occur. The analysis was performed from the start of the rupture until break flow was terminated at equalization of primary-and-secondary pressures. The methodology included the explicit modeling of operator actions in the WOG E-3 Guidelines required for mitigation of the SGTR accident.

The mass-release analyses were performed using the methodology in WCAP-10698 and its Supplement 1 (References 1 and 2). The plant response, the integrated primary to secondary break flow, the feedwater flows to both steam generators, and the steam releases to the condenser and to the atmosphere up to the time the tube rupture flow was terminated were all calculated using RETRAN-02 results. When calculating the amount of break flow that flashed to steam, 100% of the break flow was assumed to come from the hot leg side of the break.

The steam release and feedwater flow from the time of tube rupture flow termination to 2 hours, and from 2 to 8 hours, were determined from mass-and-energy balances using the RCS and intact steam generator conditions. Following termination of the tube rupture flow, the intact steam generator's ARV was assumed to cool down the plant at less than the maximum allowable rate of 100°F/hour to an RHRS in-service temperature of 330°F.

The ruptured steam generator was assumed to be depressurized to the RHRS in-service pressure of 340 psia immediately after the RCS cooldown. The amount of steam released was determined from mass-and-energy balances; no changes in thermodynamic conditions were assumed from termination of the tube rupture flow until depressurization was started since the ruptured steam generator was isolated. Steam releases from both steam generators are considered terminated at 8 hours when the RHRS in-service conditions were reached.

Evaluation of Impact of 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 plant systems and components whose performance is relied upon to support the inputs, assumptions, and results of the SGTR analysis are discussed in SER section 2.3, "Scoping and Screening Results: Mechanical Systems." EPU activities do not add any new components nor do they introduce any new functions for existing plant components relied upon to mitigate the effects of a SGTR event that would change the license renewal evaluation boundaries. The system and component performance capability in response to a SGTR event described in this section for the proposed EPU LR involves analytical techniques and methodology which are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of 10CFR50.67. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to performance capability of systems and components relied upon to mitigate the effects of a SGTR event.

2.8.5.6.2.3 SGTR Results

Only the results for the limiting margin-to-overfill and mass-release cases are presented.

SGTR Margin-to-Overfill Transient Analysis

Results are presented for the worst-case margin-to-overfill analysis (Case 3). The minimum margin-to-overfill occurred with a steam generator tube plugging level of 10% and with the reactor initially operating with T_{avg} at 564.6°F. The sequence of events is summarized in

Table 2.8.5.6.2-4 and Figures 2.8.5.6.2-2 to 2.8.5.6.2-7 show primary and secondary side responses until the SGTR flow was terminated.

To ensure proper initialization of the RETRAN-02 model, 100 seconds of steady-state operation were modeled prior to initiating the break, and all times listed include this 100 seconds. Once the break was initiated, the reactor coolant flow to the secondary side through the ruptured tube immediately caused the pressurizer level and pressure to decrease, as shown in Figure 2.8.5.6.2-2. The continued decrease in pressurizer pressure caused the overtemperature ΔT setpoint to be reached in 166 seconds, followed by immediate reactor and turbine trips. The reactor coolant pumps tripped due to the assumed loss-of-offsite power at the time of reactor trip. Immediately following reactor trip, the temperature differential across the hot and cold legs decreased as core power decayed. The temperature differential then increased as shown in Figure 2.8.5.6.2-4 as both pumps coasted down and natural circulation flow developed.

With the steam dump valves closed after trip (due to the loss-of-condenser vacuum resulting from the assumed loss-of-offsite power at the time of reactor trip), the secondary side pressures in both steam generators increased rapidly to the ARV setpoint as shown in Figure 2.8.5.6.2-3. The pressurizer level and pressure continued to drop, and safety injection was actuated via the low-pressurizer pressure setpoint at 396 seconds (see Figure 2.8.5.6.2-2 and Table 2.8.5.6.2-4).

The operator isolated the ruptured steam generator by isolating steam flow and throttling auxiliary feedwater flow at 10 minutes after break initiation (see Table 2.8.5.6.2-4). The operator actions were assumed at 10 minutes after break initiation since the ruptured steam generator's narrow range level had previously returned to greater than 33%. After auxiliary feedwater isolation, the increase in fluid mass in the ruptured steam generator (shown in Figure 2.8.5.6.2-3) was due to the ruptured tube flow.

There was a 5-minute operator delay time before initiating the cooldown (see Table 2.8.5.6.2-1). The intact steam generator's ARV was assumed to fail at the start of cooldown. An additional delay was required for the operator to identify and open the ARV (see Table 2.8.5.6.2-2). At 1601 seconds, the operators started opening the valve, and it was full open at 2201 seconds. The subsequent reduction in the intact steam generator's pressure is shown in Figure 2.8.5.6.2-3, and the resulting cooldown of the RCS temperature is shown in Figure 2.8.5.6.2-4. The pressurizer pressure also decreased during this cooldown, as shown in Figure 2.8.5.6.2-2. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure was 20°F, plus an allowance of 18°F for instrument uncertainty. After cooldown, it took the operator 5 minutes to close the ARV block valve (see Table 2.8.5.6.2-2). The valve was completely closed at 3135 seconds.

The intact steam generator's ARV was later re-opened to dump steam (see Figure 2.8.5.6.2-6) and maintain an adequate RCS subcooling margin. When the ARV was opened, the increased energy transfer from the primary to the secondary side also aided in the depressurization of the RCS to the ruptured steam generator's pressure (see Figures 2.8.5.6.2-2 and 2.8.5.6.2-3).

The operator began to depressurize the RCS using the PORV at 3255 seconds after a 2-minute delay (see Table 2.8.5.6.2-1). Depressurization was terminated at 3288 seconds when the RCS pressure was reduced below the ruptured steam generator's pressure and the pressurizer's level was greater than 5%. The depressurization reduced pressurizer pressure and the break flow and increased safety injection flow to refill the pressurizer, as shown in Figures 2.8.5.6.2-2 and 2.8.5.6.2-5.

A 1-minute delay was imposed prior to termination of safety injection flow (see Table 2.8.5.6.2-1). Safety injection was not terminated in the analysis until the safety injection termination criteria were satisfied. The RCS pressure was allowed to increase to 50 psi above the ruptured steam generator pressure to ensure that the RCS pressure was increasing when safety injection was terminated. The operator terminated safety injection at 3354 seconds when the safety injection termination criteria were satisfied and the RCS pressure began to decrease, as shown in Figure 2.8.5.6.2-2. The primary-to-secondary flow continued until the RCS and ruptured steam generator pressures equalized at approximately 4172 seconds.

The primary-to-secondary break flow rate and water volume in the ruptured steam generator are shown in Figure 2.8.5.6.2-5 and 2.8.5.6.2-7, respectively. Figure 2.8.5.6.2-7 shows a 220 ft^3 margin-to-overfill relative to the total steam generator's total volume of 4512.7 ft^3. Therefore, it was concluded that overfill of the ruptured steam generator would not occur for a design basis SGTR for the Ginna Station.

The analyses presented in Ginna UFSAR section 15.6.3 show 199 ft^3 margin. The net effect of the EPU and associated changes in initial conditions, and the change to the RETRAN-02 code modeling of the transient, are a minimal increase in the margin-to-overfill.

SGTR Mass-Release Transient Analysis

The maximum mass release occurred with a steam generator tube plugging level of 10%, and with the reactor initially operating with T_{avg} at 576.0°F (Case 5). The sequence of events is summarized in Table 2.8.5.6.2-5, and the primary and secondary side responses appear in Figures 2.8.5.6.2-8 to 2.8.5.6.2-13. Total mass releases for use in the dose analyses are summarized in Table 2.8.5.6.2-6.

The mass-release and margin-to-overfill results were similar until 10 minutes from break initiation. The mass-release transient modeled a low initial secondary inventory and minimum auxiliary feedwater flow. As a result, the ruptured steam generator level did not reach 33% until 942 seconds. Isolating the ruptured steam generator was therefore delayed until 942 seconds, consistent with Table 2.8.5.6.2-1. At 942 seconds, the ruptured steam generator's ARV was assumed to fail open. The failure of the ARV caused the steam generator to rapidly depressurize, and the primary-to-secondary flow through the ruptured tube to increase (see Figures 2.8.5.6.2-9 and 2.8.5.6.2-11). The ruptured steam generator's depressurization caused the RCS pressure and temperature to decrease more rapidly than the overfill case (see Figures 2.8.5.6.2-8 and 2.8.5.6.2-10), as well as a greater cooldown of the intact steam generator. The operator identified and locally closed the block valve for the failed ARV after 15 minutes (see Table 2.8.5.6.2-2). The depressurization of the ruptured steam generator stopped at 1842 seconds, and its pressure began to increase, as shown in Figure 2.8.5.6.2-9.

There was a 5-minute operator action delay time imposed prior to initiating cooldown after the failed ARV's block valve was closed (see Table 2.8.5.6.2-1). The cooldown was performed using the intact steam generator's ARV to dump steam to the atmosphere, and continued until the RCS subcooling at the ruptured steam generator pressure was 20°F, plus an allowance of 18°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator when the cooldown was initiated, the RCS had to be cooled to a lower temperature to satisfy the cooldown criterion. The net effect was that the cooldown period was longer, relative to the overfill case. The cooldown was completed at 4373 seconds when the operator closed the ARV on the intact steam generator. The reductions in the intact steam generator pressure and the RCS temperature during the cooldown period are shown in Figures 2.8.5.6.2-9 and 2.8.5.6.2-10, respectively. The intact steam generator's ARV was later reopened (see Figure 2.8.5.6.2-12) to maintain RCS temperature and subcooling margin.

The RCS depressurization began later than the limiting margin-to-overfill case. After a 2-minute delay (see Table 2.8.5.6.2-1), the operator used the PORV to depressurize, starting at 4493 seconds. Depressurization was terminated at 4546 seconds, when the RCS pressure was less than the ruptured steam generator's pressure and the pressurizer's level was above 5%. During depressurization, safety injection flow refilled the pressurizer while break flow was reduced, as shown in Figures 2.8.5.6.2-8 and 2.8.5.6.2-11, respectively.

At this point, a 1-minute operator delay (see Table 2.8.5.6.2-1) was assumed before shutting down safety injection at 4607 seconds. Like the overfill analysis, safety injection was terminated when the criteria were satisfied, and the RCS pressure reaches 50 psi above the ruptured steam generator's pressure. The RCS pressure began to decrease, as shown in Figure 2.8.5.6.2-8. Figure 2.8.5.6.2-11 shows that the primary-to-secondary flow continued until the RCS and ruptured steam generator pressures equalized at 5684 seconds.

The maximum integrated flashing break flow was 6586 lbm. Figure 2.8.5.6.2-14 shows the flashing fraction and integrated flashed break flow.

Following termination of the tube rupture flow, the RCS was cooled down using the intact steam generator. The steam releases are presented in Table 2.8.5.6.2-6. Since the condenser was in service until reactor trip, any radioactivity released to the atmosphere before reactor trip was through the condenser air ejector. After reactor trip, the releases were assumed to be via the ARVs. Table 2.8.5.6.2-6 indicates that approximately 82,900 lbm of steam was released to the atmosphere from the ruptured steam generator within the first 2 hours (i.e., the ruptured steam generator was isolated within this interval). After 2 hours, 26,800 lbm of steam was released to the atmosphere from the ruptured steam generator, when it was depressurized after the RCS was cooled to the RHRS in-service temperature. A total of 175,800 lbm of reactor coolant flowed through the tube rupture before break flow was terminated.

The break flow and releases calculated for the EPU were higher than those listed in Ginna UFSAR section 15.6.3. The higher power resulted in higher steam releases. Changes to the assumed initial secondary side water mass and assumed auxiliary feedwater flow resulted in delayed isolation of the ruptured steam generator. This delayed isolation delayed all subsequent actions, including break flow termination, resulting in increased total break flow. The RETRAN-02 model used for the EPU analysis and the LOFTTR2 model used for the analysis presented in Ginna UFSAR section 15.6.3 exhibited different steam generator secondary pressure responses following closure of the ruptured steam generator ARV block valve. The RETRAN-02 model did not re-pressurize as significantly as the LOFTTR2 model. This led to a lower target temperature for the cooldown and a lower primary pressure required for break flow termination. Both of these further delayed break flow termination in the EPU analysis compared to the analysis in the Ginna UFSAR.

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis has been completed and data tabulated for the limiting case. The results of the analysis were used as input to the radiological consequences analysis presented in LR section 2.9.6, "Radiological Consequences of a Steam Generator Tube Rupture."

2.8.5.6.2.4 SGTR References

1. WCAP-10698-P-A (Proprietary), *SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill*, Lewis, Huang, Behnke, Fittante, Gelman, August 1987. (Currently incorporated within the Ginna UFSAR.)

2. WCAP-10698-P-A (Proprietary) Supplement 1, *Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident*, Lewis, Huang, Rubin, March 1986. (Currently incorporated within the Ginna UFSAR.)

3. WCAP-11002 (Proprietary), *Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident*, Lewis, Huang, Rubin, Murray, Roidt, Hopkins, February 1986. (Currently incorporated within the Ginna UFSAR.)

4. WCAP-14882-P-A (Proprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, Love, Matthys, Monahan, O'Hair, Reck, Sechrist, Treleani, April 1999. (Planned to be

incorporated into the Ginna UFSAR via the 10CFR50.59 process prior to EPU implementation.)

Table 2.8.5.6.2-1
Operator Action Times For Design Basis SGTR Analysis

Action	Time
Identify and isolate ruptured steam generator	Maximum of 10 minutes or calculated time to reach 33% narrow range level in the ruptured steam generator
Operator action time to initiate cooldown	5 minutes from complete isolation of ruptured steam generator
Cooldown	Calculated time for RCS cooldown
Operator action time to initiate depressurization	2 minutes from end of cooldown
Depressurization	Calculated time for RCS depressurization
Operator action time to initiate safety injection termination	Maximum of 1 minute from end of depressurization or time to satisfy safety injection termination criteria
Pressure equalization	Calculated time for equalization of RCS and ruptured steam generator pressures

Table 2.8.5.6.2-2 SGTR Analysis Times for Operator Actions Due to Single Failures	
Action	**Time**
Faulted steam generator ARV block valve closing - local action[a]	15 minutes
Intact steam generator ARV opening - local action[b]	20 minutes
Intact steam ARV block valve closing - local action	5 minutes

Notes:

a. For the mass-release analysis, the ruptured steam generator ARV is assumed to fail open at steamline isolation.

b. For the margin to steam generator overfill analysis, the intact steam generator ARV is assumed to fail closed when the cooldown should be initiated. The analysis assumes that 10 minutes are required to identify and locate the failed intact steam generator ARV, and 10 minutes are required to open the valve.

Table 2.8.5.6.2-3
Plant Parameters Used in SGTR Analysis

	SGTR Overfill Analysis	SGTR Dose Analysis
Initial RCS pressure (psia)	2190	2190
Initial steam generator water mass (lbm)	96,000	68,000
Reactor trip delay (sec)	0.0	0.0
Turbine trip delay	0.0	0.0
Pressurizer pressure for safety injection (psia)	1785	1785
Steam generator atmospheric relief valve setpoint (psia)	1065	1065
Safety injection system pump delay (sec)	0.0	0.0
Auxiliary feedwater delay (sec)	0.0	60
Auxiliary feedwater flow rate per steam generator (gpm)	468	370
Auxiliary feedwater temperature (°F)	104	104
Safety injection flow vs. reactor coolant system pressure (lbm/sec vs. psia)	See Figure 2.8.5.6.2-1	See Figure 2.8.5.6.2-1
Decay heat	120% ANS	120% ANS

Table 2.8.5.6.2-4 Sequence of Events for Limiting Margin-to-overfill Analysis	
Event	**Time (seconds)**
SGTR	100
Reactor trip	166
Safety injection	396
Ruptured steam generator isolated	701
Manual opening of intact steam generator ARV initiated	1601
Manual opening of intact steam generator ARV completed	2201
RCS cooldown target temperature reached	2835
Manual isolation of intact steam generator ARV completed	3135
Pressurizer power operated relief valve (PORV) opened	3255
Pressurizer PORV closed	3288
Safety injection terminated	3354
Break flow terminated	4172

Table 2.8.5.6.2-5
Sequence of Events for Input to Radiological Consequences Analysis

Event	Time (seconds)
SGTR	100
Reactor trip	174
Safety injection	342
Ruptured steam generator isolated	942
Ruptured steam generator ARV fails open	942
Ruptured steam generator ARV block valve closed	1842
Intact steam generator ARV opened	2143
Break flow stops flashing	2636
Intact steam generator ARV closed	4373
Pressurizer PORV opened	4493
Pressurizer PORV closed	4546
Safety injection terminated	4607
Break flow terminated	5684

Table 2.8.5.6.2-6
Mass Releases
Total Mass Flow (Pounds)

	Time Period			
	Start of Event to Time of Reactor Trip[a]	**Time of Reactor Trip to Time at Which Break Flow is Terminated[a]**	**Time at Which Break Flow is Terminated to 2 Hours**	**2 Hours to Time at Which RCS Reaches RHR In-Service Conditions[a]**
Ruptured Steam Generator	189,100[b]	0	0	0
- Condenser	0	82,900	0	26,800
- Atmosphere	181,800[b]	22,600	0	0
- Feedwater				
Intact Steam Generator				
- Condenser	188,400[b]	0	0	0
- Atmosphere	0	108,800	68,000	515,900
- Feedwater	189,800[b]	153,300	70,000	521,500
Break Flow	4,200	171,600	0	0
Flashed Break Flow	746	5,840	0	0

Notes:
a. The break is initiated at 100 seconds. Reactor trip occurs at 174 seconds; break flow stops flashing at 2636 seconds; break flow is terminated at 5684 seconds; RHR conditions are reached at 8 hours.
b. Pre-trip releases to condenser and feedwater flows include 100 seconds steady-state operation prior to initiation of the break.

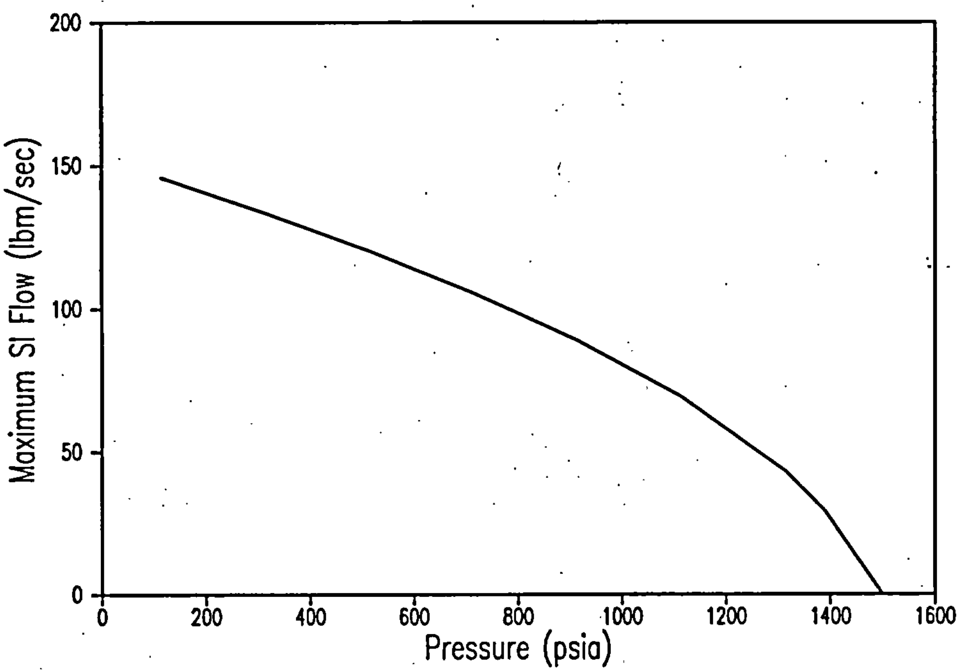


Figure 2.8.5.6.2-1
SGTR Maximum Safety Injection Flow vs. Pressure

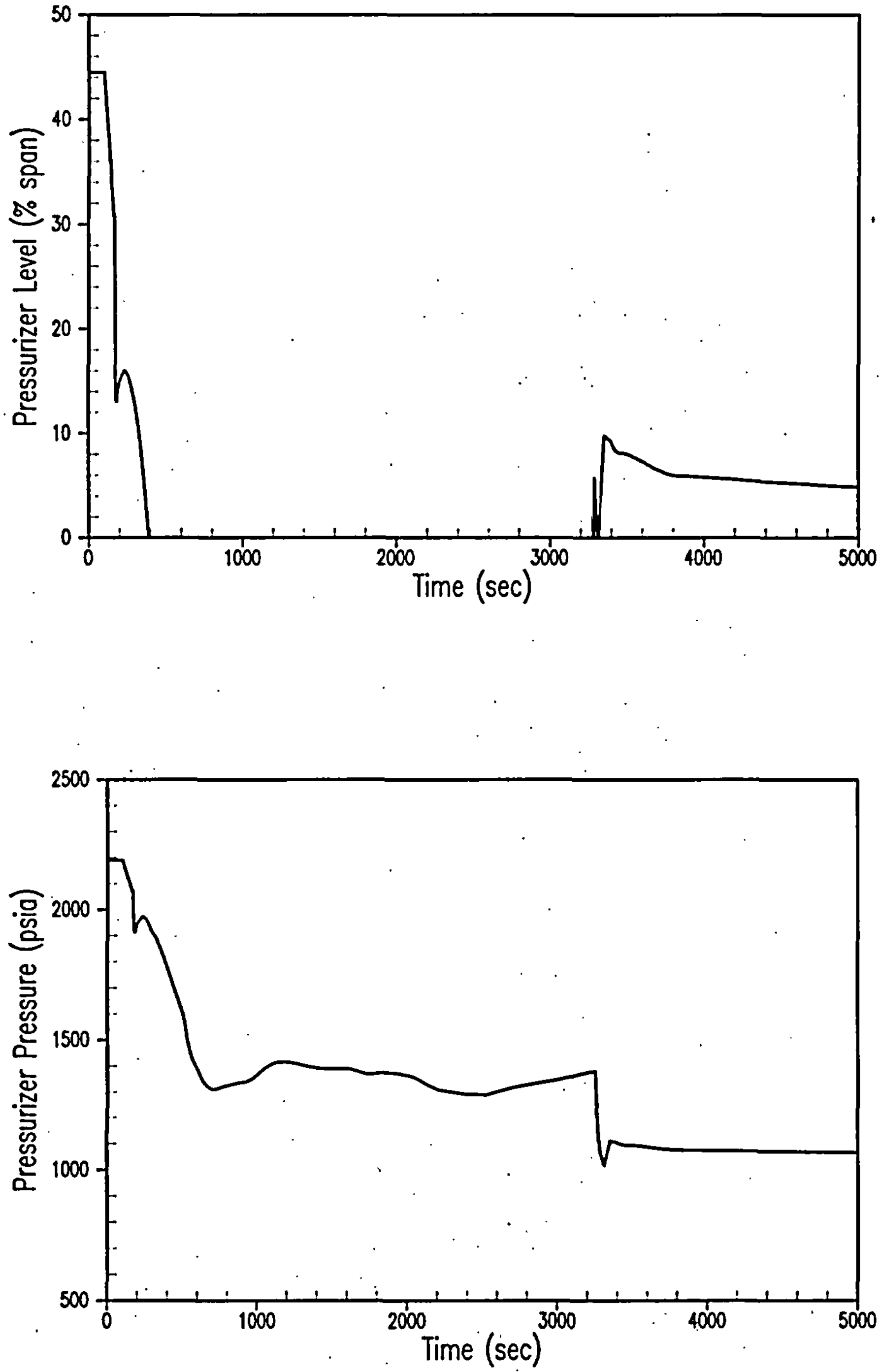
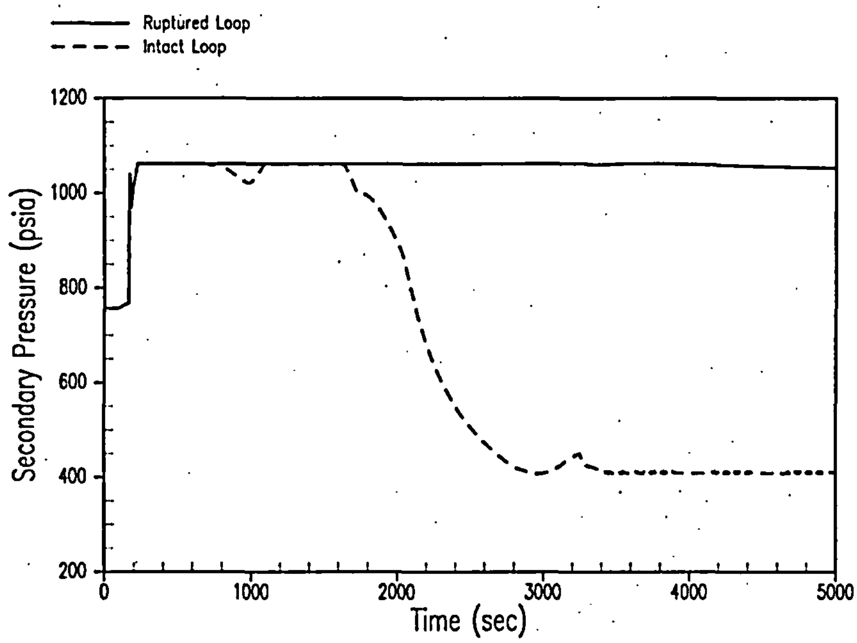


Figure 2.8.5.6.2-2
SGTR (Overfill), Pressurizer Level and Pressurizer Pressure vs. Time



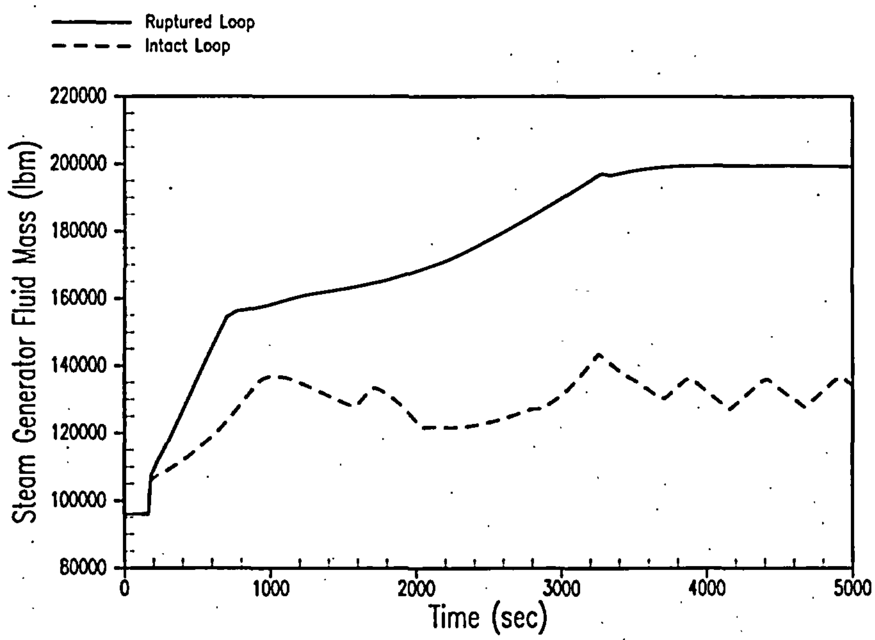


Figure 2.8.5.6.2-3

SGTR (Overfill), Secondary Pressure and Steam Generator Liquid Mass vs. Time

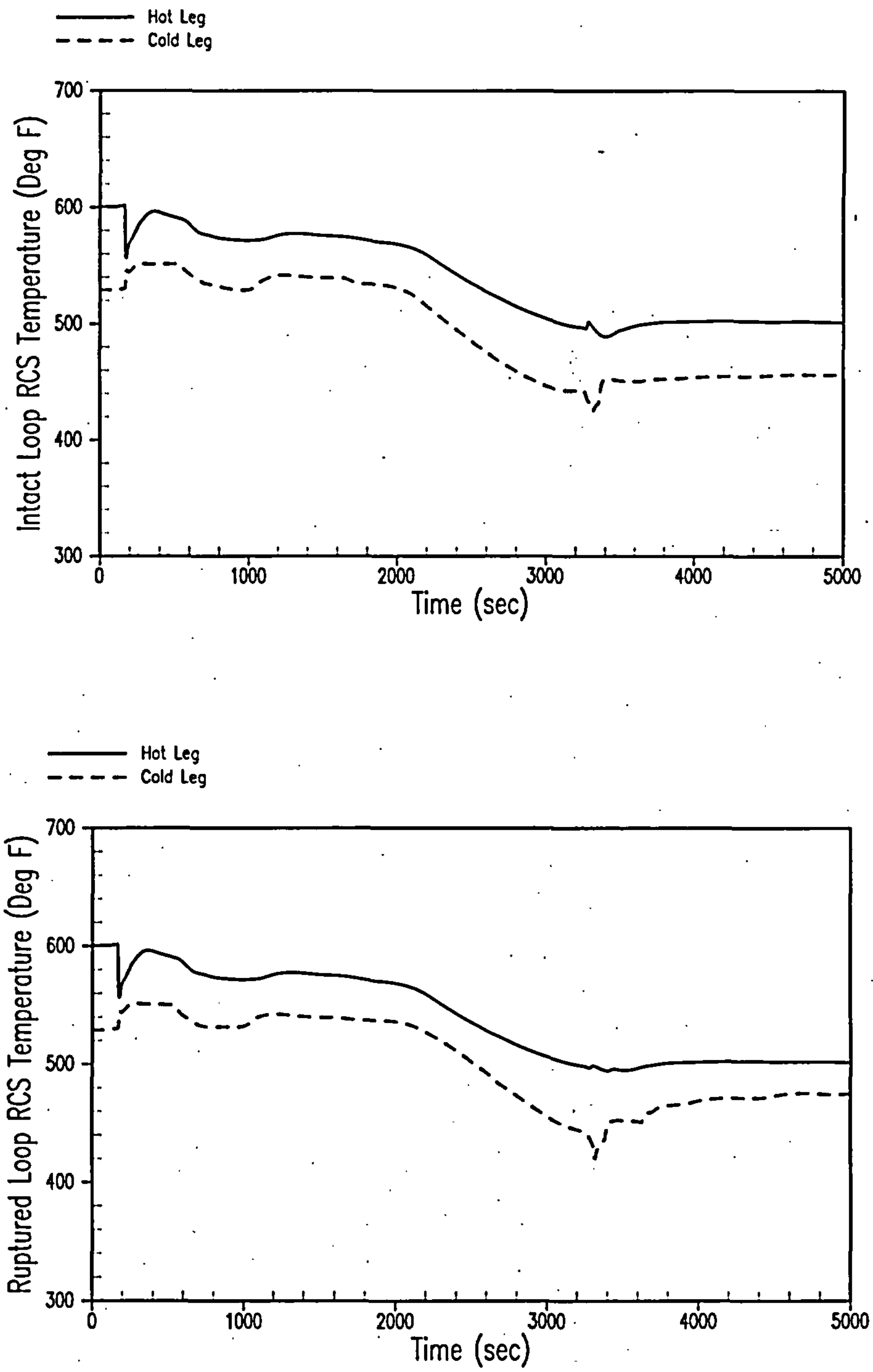


Figure 2.8.5.6.2-4
SGTR (Overfill), Hot and Cold Leg Temperatures for Intact and Ruptured Steam Generators vs. Time

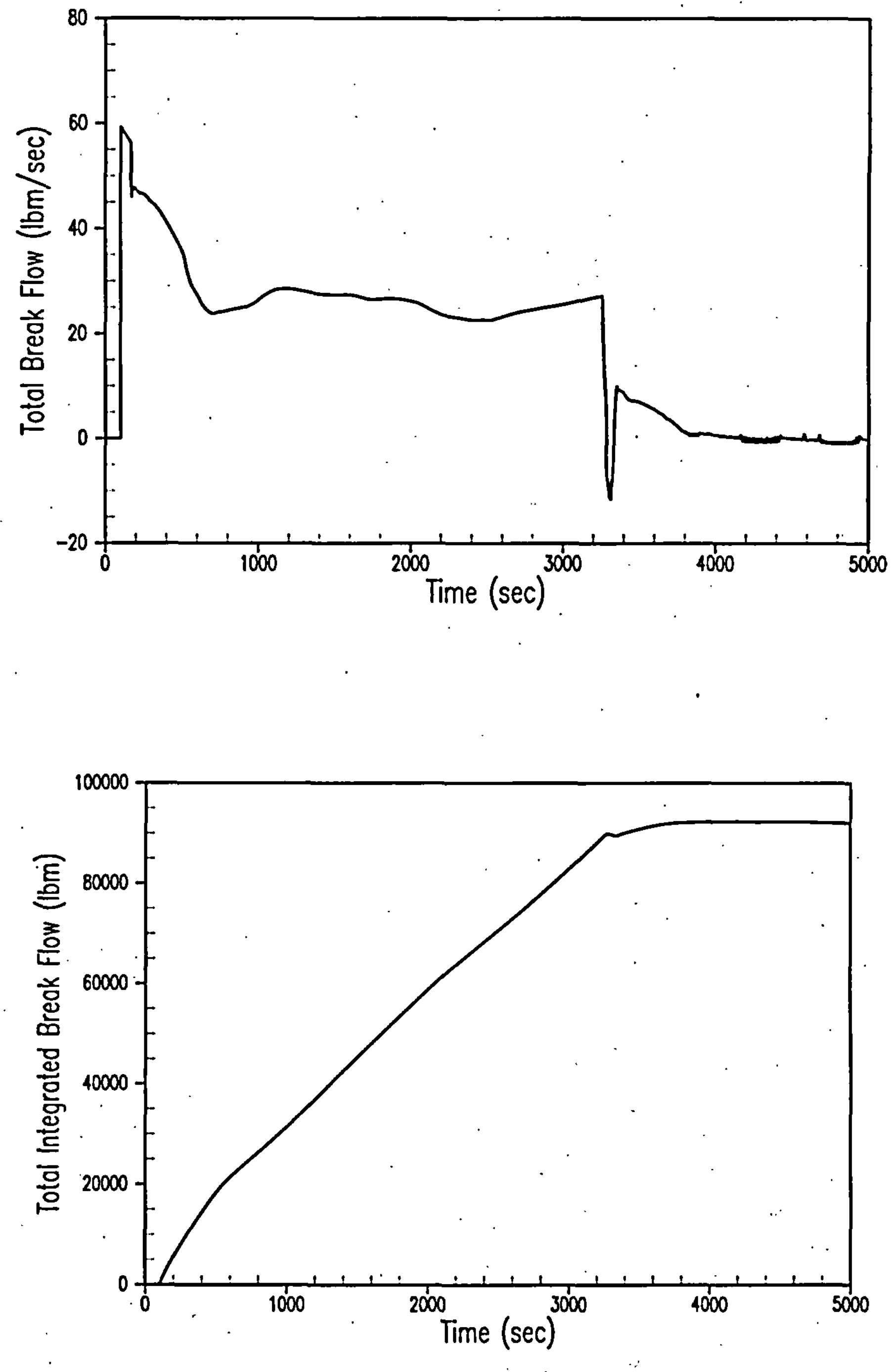


Figure 2.8.5.6.2-5
SGTR (Overfill), Total Primary to Secondary Leakage and Total Integrated Primary to Secondary Leakage vs. Time

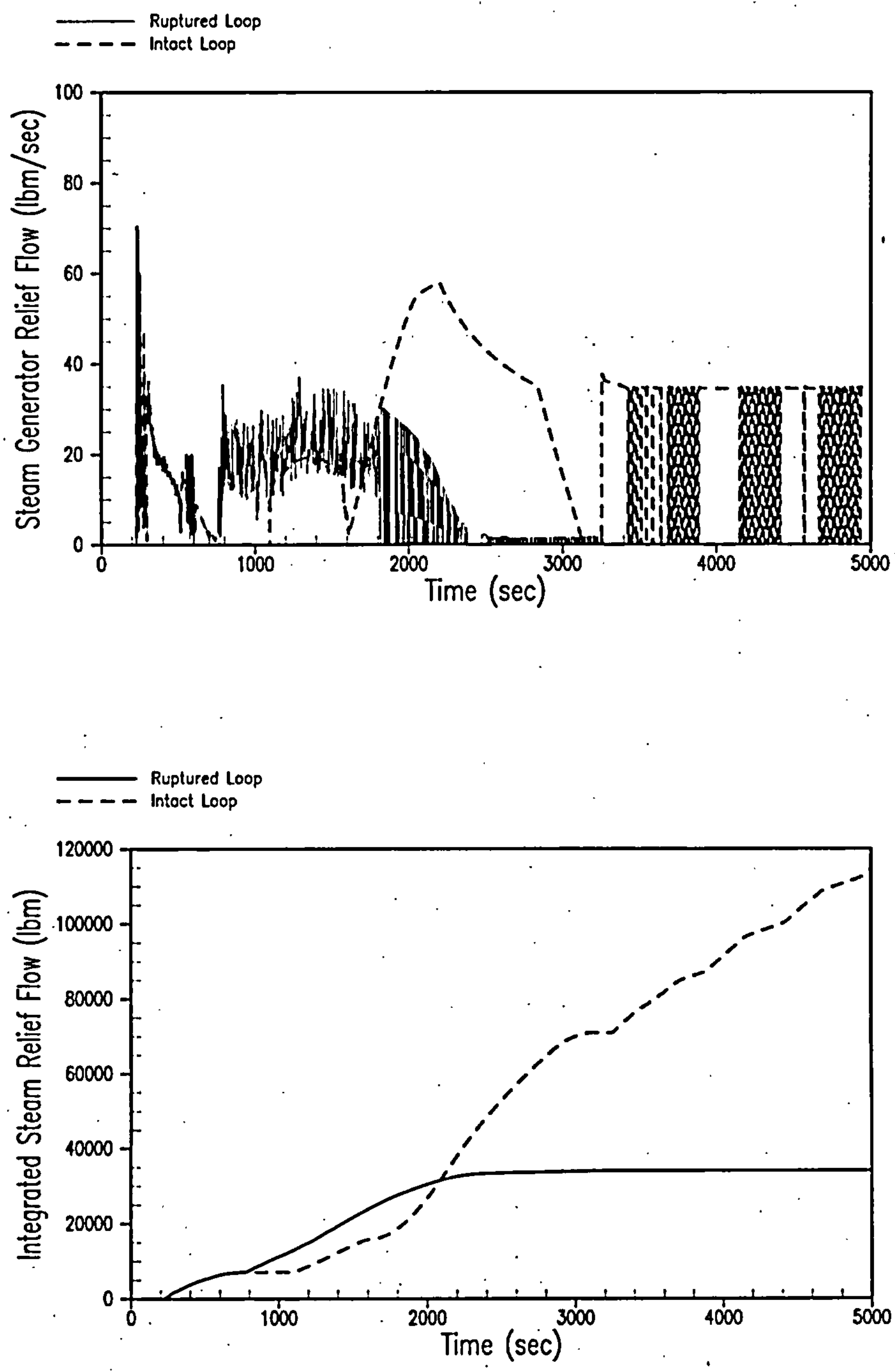


Figure 2.8.5.6.2-6
SGTR (Overfill), Steam Generator Relief Flow and Integrated Steam Generator Relief Flow vs. Time

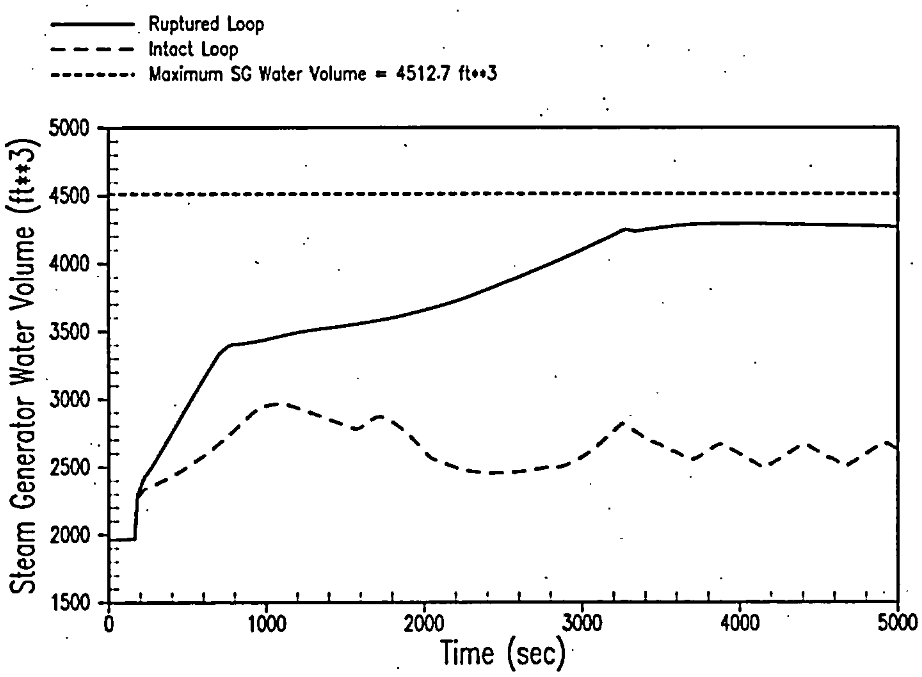
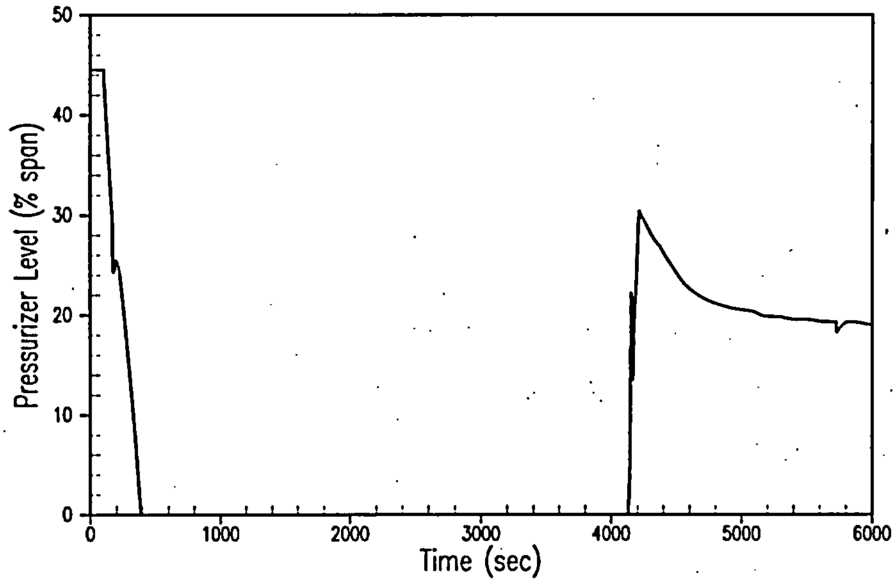


Figure 2.8.5.6.2-7
SGTR (Overfill), Steam Generator Water Volume vs. Time



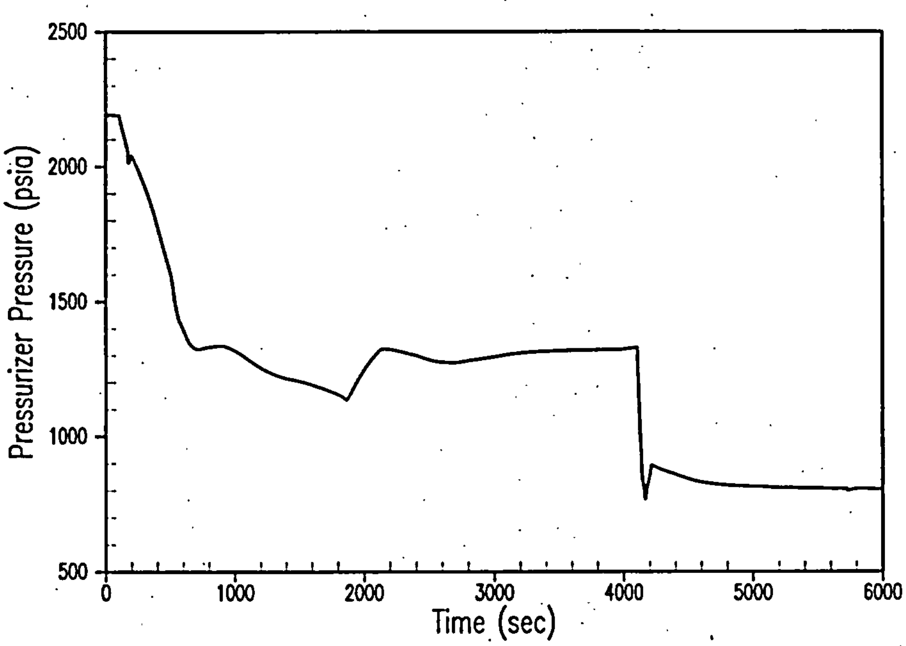


Figure 2.8.5.6.2-8
SGTR (Dose), Pressurizer Level and Pressurizer Pressure vs. Time

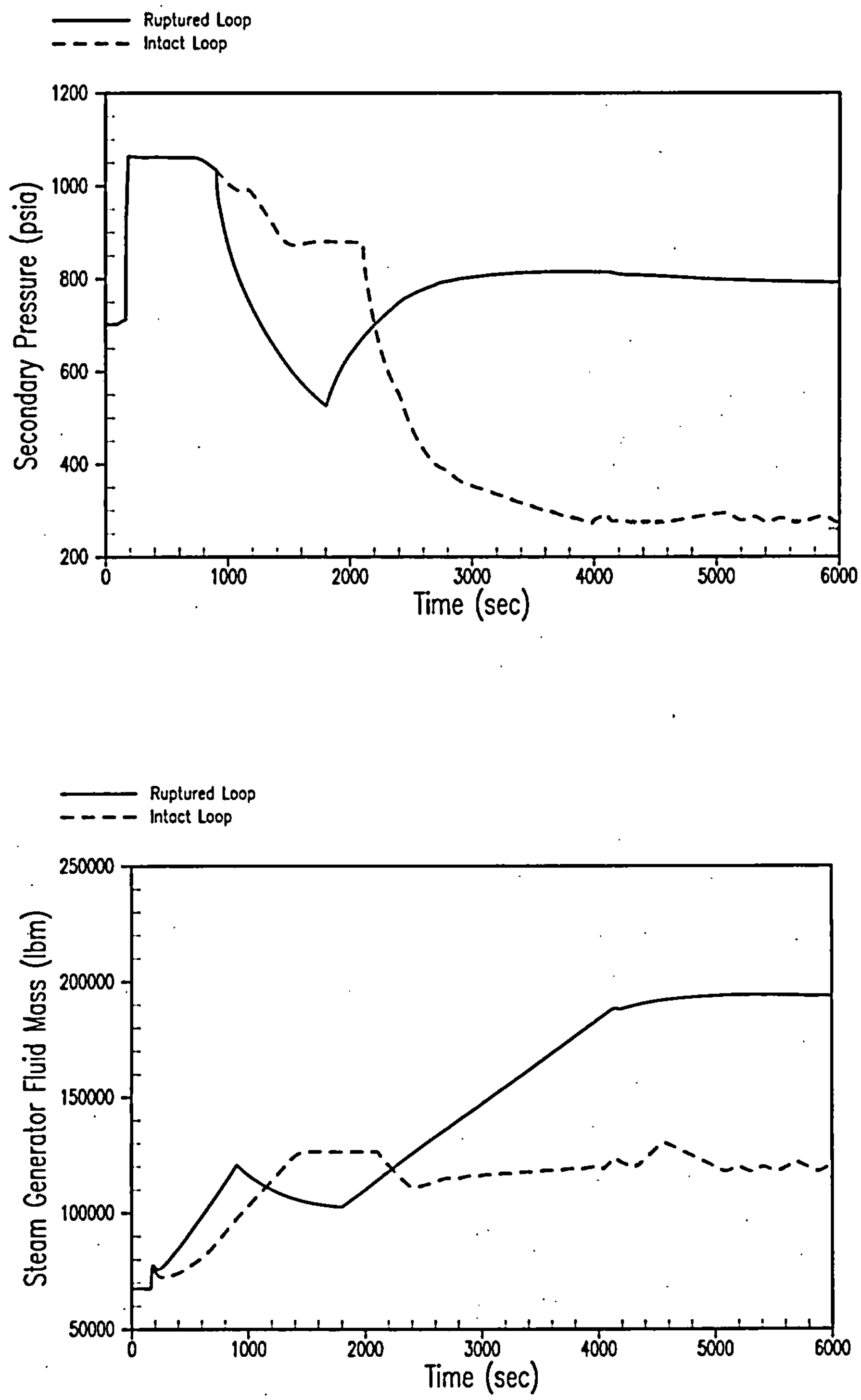


Figure 2.8.5.6.2-9

SGTR (Dose), Secondary Pressure and Steam Generator Liquid Mass vs. Time

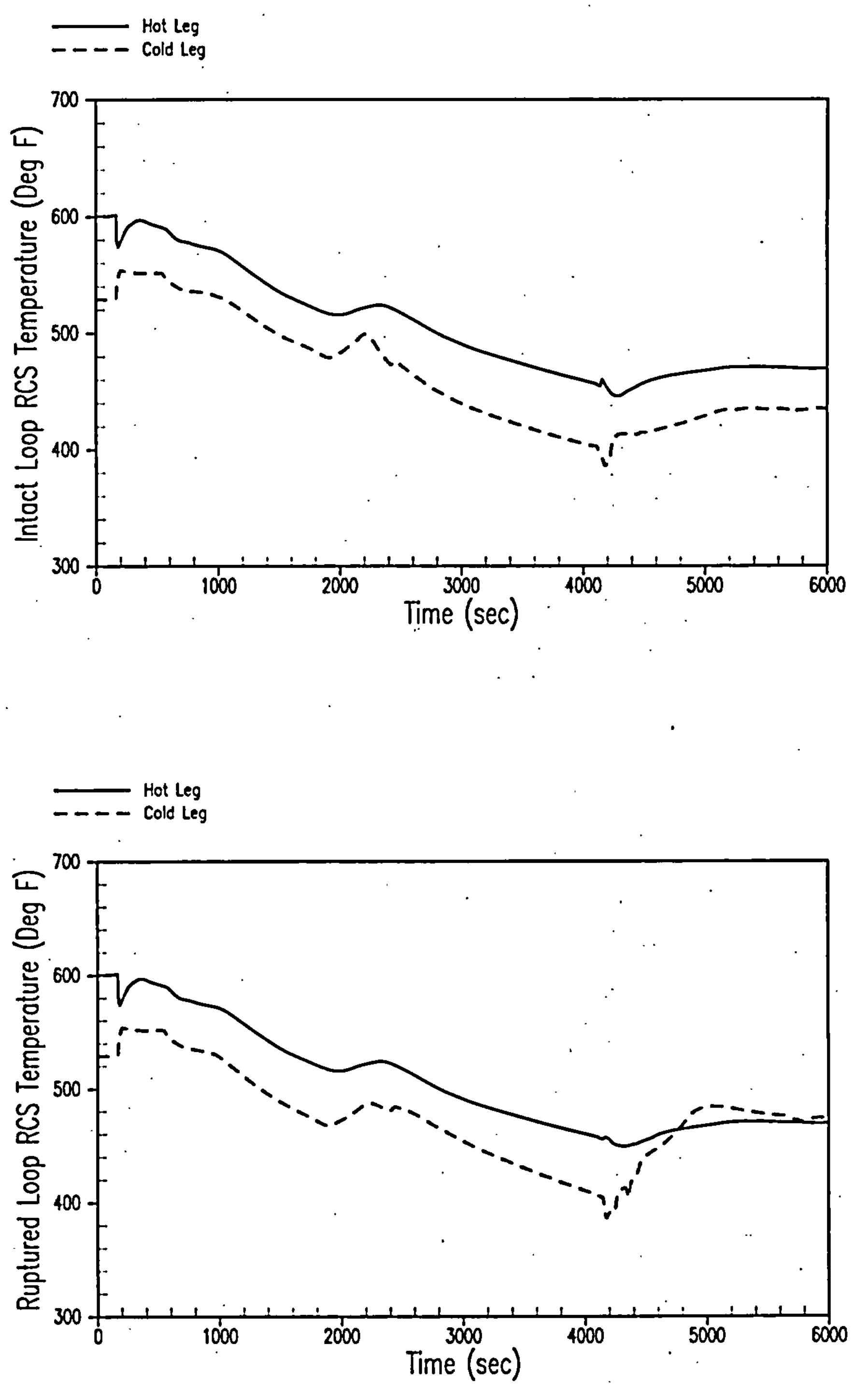
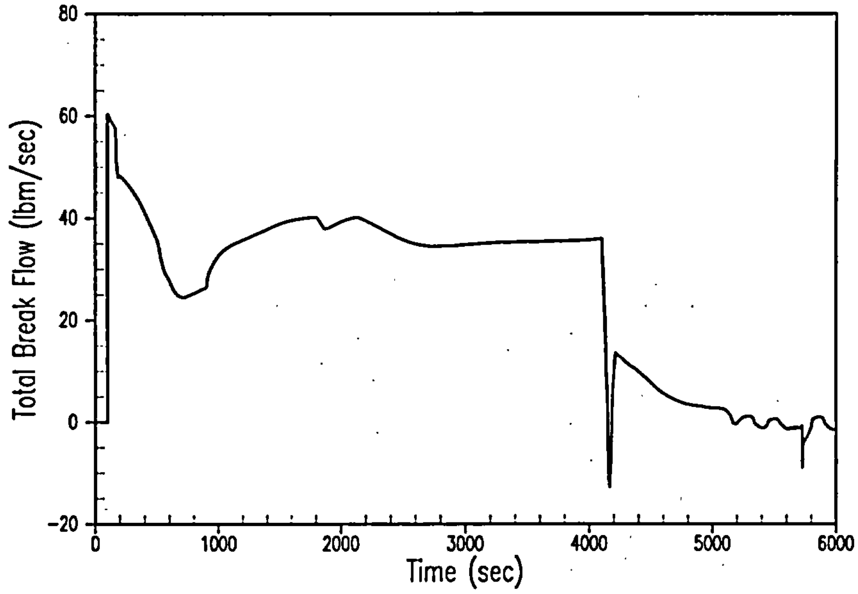


Figure 2.8.5.6.2-10
SGTR (Dose), Hot and Cold Leg Temperatures for Intact and Ruptured Steam Generators vs. Time



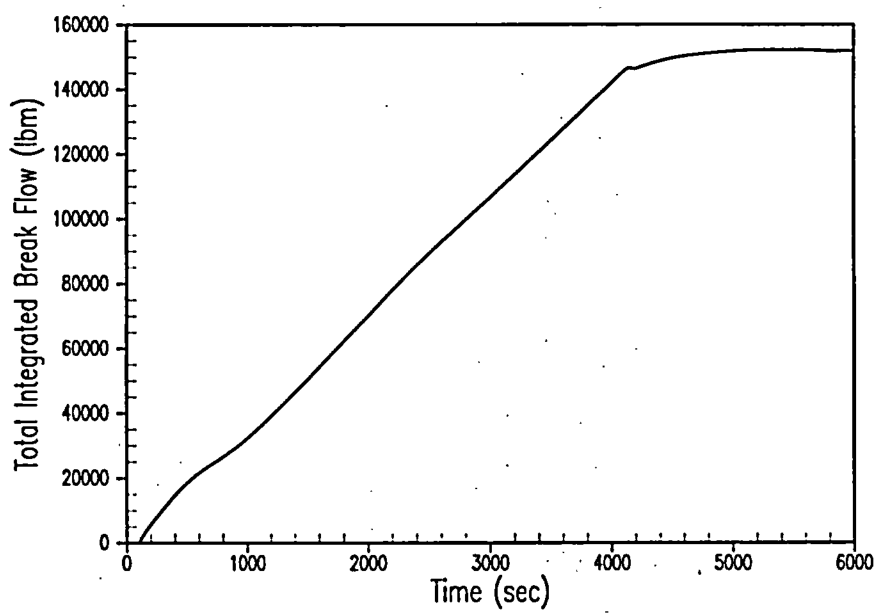
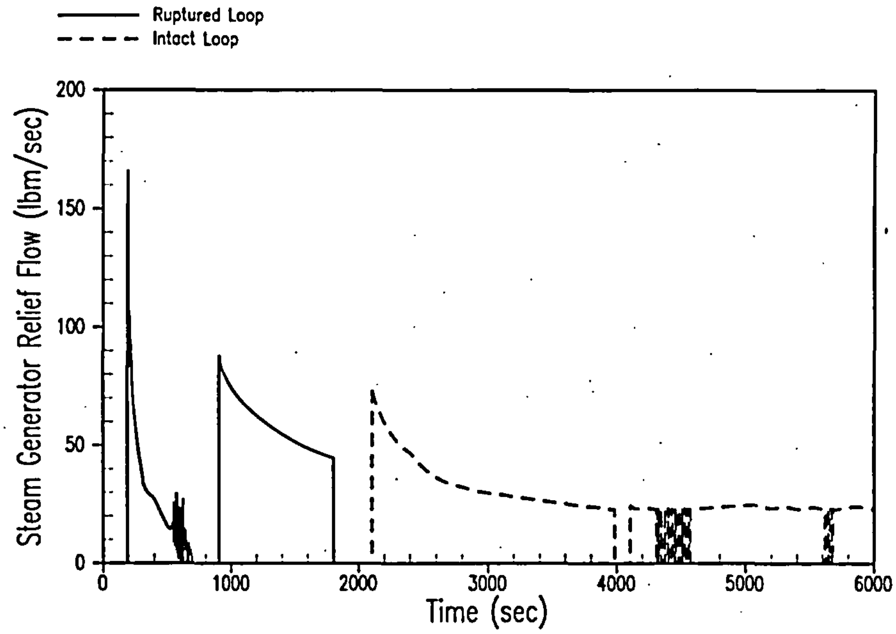


Figure 2.8.5.6.2-11
SGTR (Dose), Total Primary to Secondary Leakage and
Total Integrated Primary to Secondary Leakage vs. Time



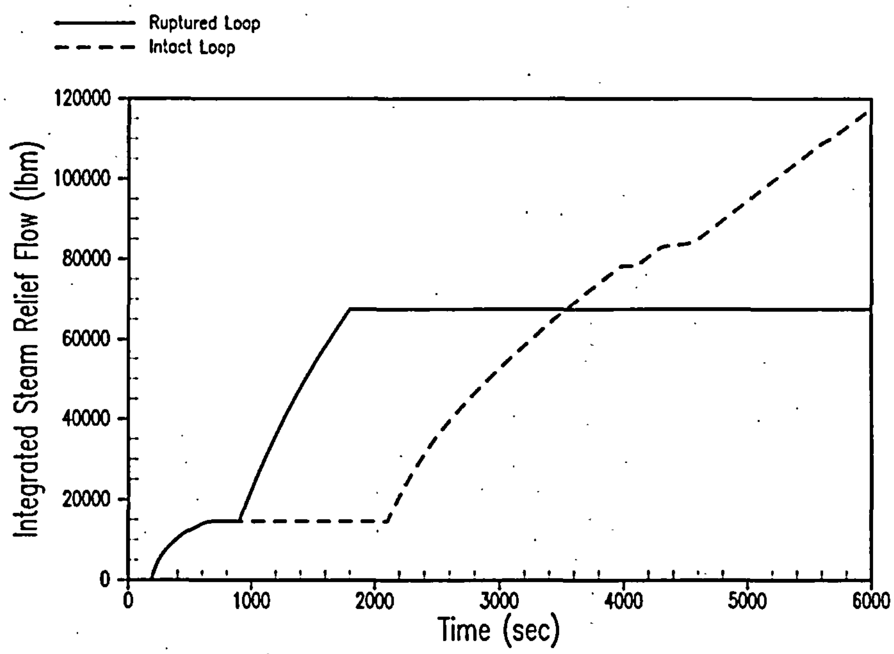


Figure 2.8.5.6.2-12
SGTR (Dose), Steam Generator Relief Flow and Integrated Steam Generator Relief Flow vs. Time

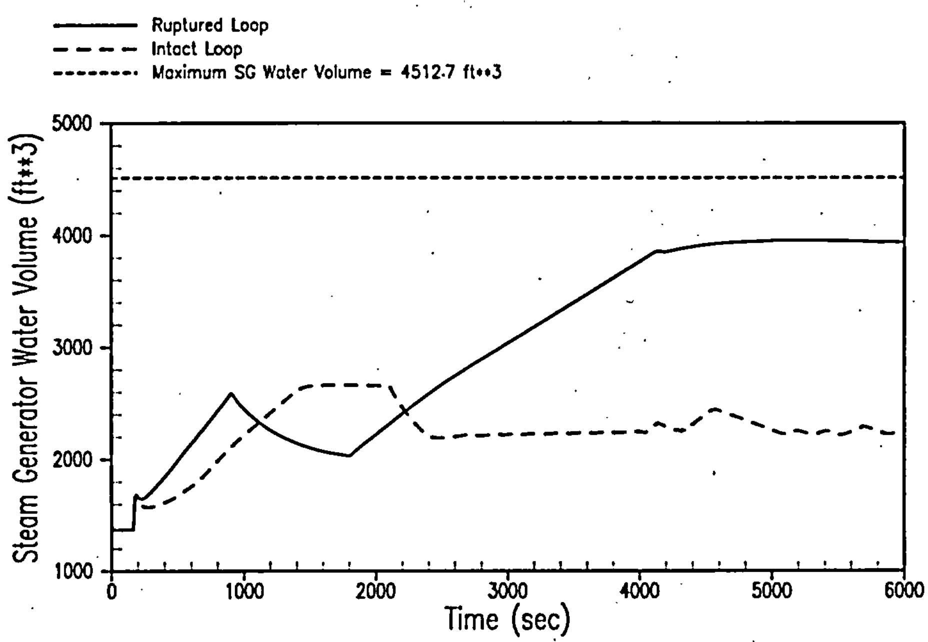
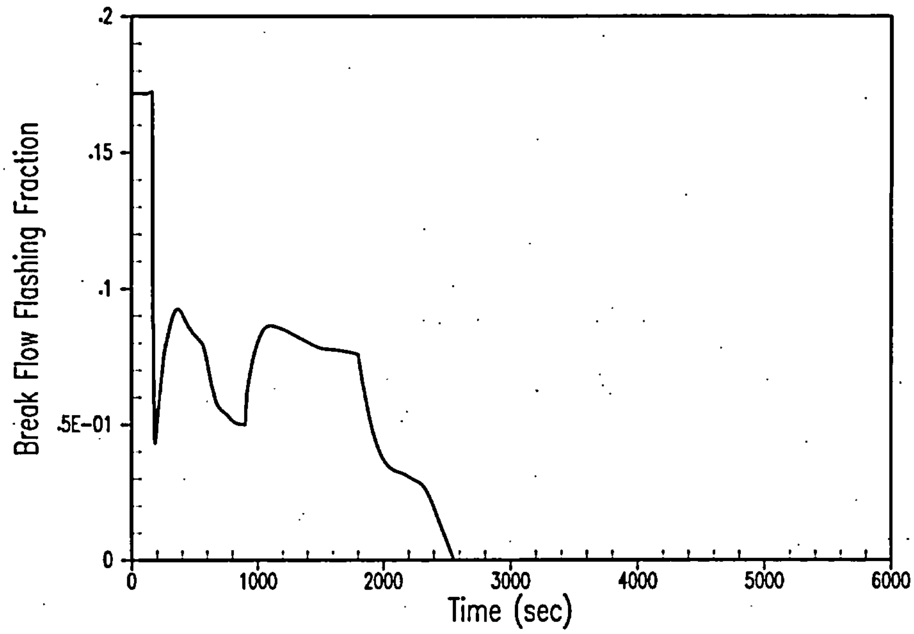


Figure 2.8.5.6.2-13
SGTR (Dose), Steam Generator Water Volume vs. Time



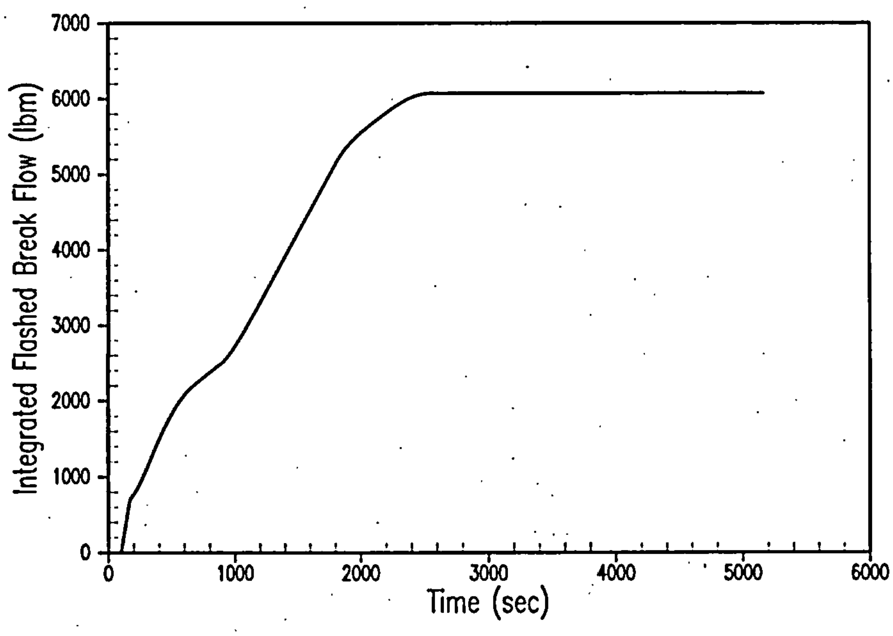


Figure 2.8.5.6.2-14
SGTR (Dose), Tube Rupture Flow Flashing Fraction and Integrated Flashed Break Flow vs. Time

2.8.5.6.2.5 Conclusion

The Ginna staff has reviewed the analysis of the SGTR accident and concludes that the analysis has adequately accounted for the plant operation at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The Ginna staff further concludes that the assumptions used in this analysis are conservative and that the event results in a 220 ft^3 margin-to-overfill of the faulted steam generator. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the SGTR event.

2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

2.8.5.6.3.1 Regulatory Evaluation

Loss of Coolant Accidents (LOCAs) are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection system (RPS) and emergency core cooling system (ECCS) are provided to mitigate these accidents. The Ginna Nuclear Power Plant, LLC (Ginna) review covered:

- The determination of break locations and break sizes
- The postulated initial conditions
- The sequence of events
- The analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients
- The calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling
- The functional and operational characteristics of the RPS and ECCS
- Operator actions

The NRC's acceptance criteria are based on:

- 10CFR50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance
- 10CFR50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA
- GDC-4, insofar as it requires that structures, systems, and components important-to-safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained

- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel and clad damage that could interfere with continued effective core cooling will be prevented

Specific review criteria are contained in SRP sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 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 confirm 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 design relative to

- GDC–4 is addressed in Ginna UFSAR section 3.1.2.1.4, "General Design Criterion 4 - Environmental and Missile Design Bases." A review of postulated pipe breaks both inside and outside containment was conducted as part of the SEP, including dynamic effects such as pipe whip and jet impingement. The spectrum of pipe breaks inside containment which were evaluated are discussed in UFSAR section 3.6.1. This discussion includes a spectrum of LOCA pipe break accidents which were analyzed and shown to have acceptable consequences as further discussed in UFSAR section 15.6.4, as supplemented by reference 2.

- GDC–27 is addressed in Ginna UFSAR section 3.1.2.3.8, "General Design Criterion 27 – Combined Reactivity Control System Capability." The Ginna Station reactivity control systems are used in conjunction with boron addition through the ECCS to control reactivity changes under postulated accident conditions, including LOCA events. LOCA events and their analyses are further discussed in UFSAR 15.6.4, as supplemented by reference 2.

- GDC–35 is addressed in Ginna UFSAR section 3.1.2.4.6, "General Design Criterion 35 – Emergency Core Cooling." The ECCS is provided to cope with any LOCA. In the unlikely event of a LOCA, the ECCS would provide cooling water to the reactor core at a sufficient rate to maintain the core in a coolable geometry and to ensure the fuel clad metal-water reaction is limited. The Ginna ECCS is capable of meeting the requirements of 10CFR50.46 and 10CFR50, Appendix K.

As noted in UFSAR section 15.6.4.2.4.2, the current Large Break LOCA analysis was performed using the Westinghouse large break LOCA WCOBRA/TRAC best-estimate methodology for plants which incorporate upper plenum injection in the safety injection system design (references 3 and 4). The Westinghouse best estimate methodology was developed consistent with guidelines set forth in SECY-83-472, "ECCS Analysis Methods," dated 12/13/83. The following statements relative to 10CFR50.46 acceptance criteria are noted in UFSAR section 15.6.4.2.4.4.

- The peak clad temperature is below the acceptance criteria limit of 2200°F.
- The maximum local cladding oxidation is below the embrittlement acceptance limit of 17%.
- The maximum core hydrogen generation is also less than 1%, in accordance with the acceptance criteria.
- The clad temperature transients are terminated at times when the core geometry is still amenable to cooling, and as a result the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period is provided.

These 10CFR50.46 acceptance criteria will continue to be met, although the LBLOCA analysis for EPU is being replaced by the Best Estimate LOCA model described below.

The small break loss-of-coolant accident (SBLOCA) analysis described in Ginna UFSAR Section 15.6.4.1 was previously performed for the steam generator replacement using the Westinghouse 1985 SBLOCA emergency core cooling system (ECCS) NOTRUMP evaluation model (References 6 through 8). The analysis of record licensing basis peak cladding temperature (PCT) is 1308°F and the current cumulative PCT is 1381°F which includes margin assessments for plant change evaluations and ECCS evaluation model errors.

The Ginna current licensing basis for post-LOCA subcriticality is embodied in the cycle-specific Reload Safety Evaluations. The Ginna current licensing basis for post-LOCA boric acid precipitation preclusion is embodied in the requirement to re-establish simultaneous hot and cold injection after 20 hours after the LOCA. This requirement is given in the Ginna Updated Final Safety Analysis Report (UFSAR), Section 6.3.3.4 and Technical Specification Section B 3.5.2

The current licensing basis, with respect to the LOCA hydraulic forces, is a combination of MULTIFLEX 1.0 analyses performed for the large branch line breaks, MULTIFLEX 3.0 analyses performed for baffle-former-barrel bolt distributions including fuel qualification (LR section 2.2.3 and LR section A.8), and analyses performed by Babcock and Wilcox, Canada (BWC), in support of the replacement steam generator (RSG) (see UFSAR section 6.2.1.2.2). The specific licensing basis analysis-of-record is dependent on the specific component analysis performed to demonstrate GDC-4 compliance.

The LOCA Forces analysis presented in UFSAR Section 3.9.2.3.2 is being superseded by this EPU analysis.

2.8.5.6.3.2 Technical Evaluation – Large Break LOCA

This section discusses the Large Break Best Estimate LOCA (LB BELOCA) analysis to support the proposed EPU for Ginna Station.

2.8.5.6.3.2.1 Introduction

The LB BELOCA is described in reference 2 for a major rupture of the reactor coolant pressure boundary (RCPB). A major rupture (large break) is defined as a breach in the RCPB with a total cross sectional area greater than 1.0 ft^2.

2.8.5.6.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The input parameters used in the LB BELOCA analysis to support an extended power uprate are contained in Table 1 of reference 2. The acceptance criteria are discussed in reference 2 as well as LR section 2.8.5.6.3.2.4.

2.8.5.6.3.2.3 Description of Analyses and Evaluations

The LB BELOCA analysis to support the proposed EPU is described in reference 2.

Evaluation of Impact of 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 plant systems and components whose performance is relied upon to support the inputs, assumptions, and results of the LOCA analyses are discussed in SER section 2.3, "Scoping and Screening Results: Mechanical Systems." EPU activities do not add any new components nor do they introduce any new functions for existing plant components relied upon to mitigate the effects of postulated LOCA events that would change the license renewal evaluation boundaries. The system and component performance capability in response to postulated LOCA events described in this section for the proposed EPU LR 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 performance capability of systems and components relied upon to mitigate the effects of postulated LOCA events.

2.8.5.6.3.2.4 Results

The results of the LB BELOCA analysis are summarized in Table 2.8.5.6.3-1, and presented in detail in Reference 2. The break location and sizes studied in this analysis are consistent with the evaluation model used for the LB BELOCA analysis (Reference 1). The postulated initial conditions for this analysis are contained in Table 1 of Reference 2. Table 3 provided in Reference 2 contains the sequence of events for the limiting PCT transient. The analytical models used in this analysis are consistent with the approved evaluation model (Reference 1). The peak cladding temperature, maximum local cladding oxidation, and core wide hydrogen generation calculated for the LB BELOCA analysis as provided in Table 2.8.5.6.3-1 meet the 10CFR50.46 acceptance criteria. Since these acceptance criteria are satisfactorily met, and the grid crush due to combined LOCA and seismic loads does not extend beyond the low power assemblies (as discussed in LR section 2.8.1, Fuel System Design), the changes in core geometry are such that the core remains amenable to cooling (Reference 5). The long term core cooling analysis is discussed in LR section 2.8.5.6.3.4, Technical Evaluation – Post LOCA. The Post-LOCA analysis results are applicable to the Large Break LOCA analysis using Best-Estimate methodology. No operator actions have been credited for the duration of the LB BELOCA analysis.

Based on the discussion in the previous paragraph, it is concluded that Ginna continues to maintain a margin of safety to the limits prescribed by 10CFR50.46 for the proposed EPU. The LB BELOCA evaluation model (Reference 1) conforms to 10CFR50, Appendix K, insofar as

Reference 1 provides the documentation required by Part II of Appendix K. The acceptance criteria meet the Ginna Station current licensing basis requirements with respect to GDC-4, insofar as it requires that safety-related structures, systems, and components be protected against dynamic effects associated with flow instabilities and loads such as those resulting from waterhammer, as discussed in LR section 2.6.1 and LR section 2.6.3.1. The acceptance criteria meet the Ginna current licensing basis requirements with respect to GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained, as discussed in the LB BELOCA analysis results and the Post-LOCA analysis results (LR section 2.8.5.6.3.4, Technical Evaluation – Post-LOCA). The acceptance criteria meet the Ginna current licensing basis requirements with respect to GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel and clad damage that could interfere with continued effective core cooling will be prevented, as discussed in the LB BELOCA analysis results and the Post-LOCA analysis (LR section 2.8.5.6.3.4, Technical Evaluation – Post-LOCA). The post-LOCA analysis results are applicable to the LB LOCA analysis using Best Estimate methodology. No operator actions have been credited for the duration of the LB LOCA analysis.

Table 2.8.5.6.3-2 contains a summary of the 10CFR50.46 criteria that are directly addressed by the LB BELOCA analysis (Peak Clad Temperature (PCT) less than 2200°F, Local Maximum Oxidation (LMO) less than 17%, and Core-Wide Oxidation (CWO) less than 1%) for the LB BELOCA analysis with ASTRUM to support the EPU compared to the current licensing basis using the SECY BELOCA EM. The additional margin to the safety limit for PCT for the EPU is generated because of the difference in methodology between the current licensing basis and the LB BELOCA analysis to support the EPU.

2.8.5.6.3.3 Technical Evaluation – Small Break LOCA

2.8.5.6.3.3.1 Introduction

The SBLOCA is described in Reference 2 for a minor rupture of the RCPB. A minor pipe rupture (small break) is defined as a breach in the reactor coolant pressure boundary with a total cross sectional area less than or equal to 1.0 ft^2.

2.8.5.6.3.3.2 Input Parameters, Assumptions, and Acceptance Criteria

The input parameters, assumptions, and acceptance criteria used in the SBLOCA analysis to support an extended power uprate are contained in Reference 2.

2.8.5.6.3.3.3 Description of Analyses and Evaluations

Details on the Ginna Station EPU SBLOCA analysis can be found in the Reference 2.

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

See LR section 2.8.5.6.3.2.3.

2.8.5.6.3.3.4 Results

The results are shown on Table 2.8.5.6.3-3. The peak cladding temperature is 1167°F, and the maximum local transient oxidation is 0.07 percent. The design limit 95% upper bound pre-transient oxidation value for each of the fuel designs that will be included in the EPU cores is <16%. The actual upper bound values predicted for each of the fuel designs are expected to be well below this value. Because the transient oxidation is so low, the sum of the transient and pre-transient oxidation remains below 16% at all times in life. The core-wide hydrogen generation remains well below the 10CFR50.46 acceptance limit of 1 percent, and the core geometry remains amenable to cooling.

The Ginna Station current licensing basis requirements with respect to GDC-4 are met by the SBLOCA analysis in that core coolability is maintained as demonstrated in LR section 2.8.1, Fuel System Design. The Ginna Station current licensing basis requirements with respect to GDC-27 are met by the SBLOCA analysis in that adequate poison is added by the ECCS to ensure the core remains subcritical as demonstrated in LR section 2.8.5.6.3.4, Technical Evaluation - Post LOCA. The Ginna Station current licensing basis requirements with respect to GDC-35 are met by the SBLOCA analysis since the 10CFR50.46 acceptance criteria are met by this analysis for the short term accident ECCS performance in conjunction with the long term cooling capability demonstrated in LR section 2.8.5.6.3.4, Technical Evaluation - Post LOCA.

2.8.5.6.3.4 Technical Evaluation – Post LOCA

The Post LOCA analysis to support an EPU, as well as the input parameters, acceptance criteria, and results are contained in Reference 2. The current licensing basis for Post-LOCA analyses and summary of the analysis results and adherence to various acceptance criteria are discussed below.

Results

A post-LOCA subcriticality boron limit curve was developed for the EPU plant conditions. Cycle-specific reload safety evaluations will ensure that the core will remain subcritical post-LOCA, and decay heat can be removed for the extended period required by the remaining long-lived radioactivity. This addresses the GDC-27 requirement for redundant shutdown capabilities.

The Ginna EPU Post LOCA boric acid precipitation calculations using conservative methodology resulted in a 6-hour timeframe to establish simultaneous cold leg SI and upper plenum injection through the core deluge valves after the termination of SI to in the cold leg. Actions to establish ECCS recirculation flow through the core to preclude boric acid precipitation addresses the requirements of 10CFR50.46 (b) (4) coolable geometry and 10CFR50.46 (b) (5) long-term cooling.

ECCS flows during sump recirculation were shown to be sufficient to remove decay heat after a LOCA for EPU plant conditions. This addresses the requirements of 10CFR50.46 (b) (5) long-term cooling.

Since Post LOCA analyses for the EPU show that no changes to the Ginna Station ECCS system are required, GDC-35 requirements continue to be met.

2.8.5.6.3.5 Technical Evaluation – LOCA Forces

2.8.5.6.3.5.1 Introduction

The analysis of the LOCA hydraulic forces generates the hydraulic forcing functions that act on reactor coolant system (RCS) components as a result of a postulated LOCA. The most recent qualification of the vessel internals and fuel was performed using an advanced beam model version of MULTIFLEX 3.0 (Reference 9), in accordance with methodology approved by the NRC in WCAP-15029-P-A (Reference 10). This same version of the MULTIFLEX code was used in the LOCA hydraulic forces analysis for the Ginna Station EPU.

2.8.5.6.3.5.2 Input Parameters, Assumptions, and Acceptance Criteria

To conservatively calculate LOCA hydraulic forces for Ginna Station, the following operating conditions were considered in establishing the limiting temperatures and pressures:

- Initial RCS conditions associated with a minimum thermal design flow of 85,100 gpm per loop
- Uprated core power of 1811 MWt (analyzed nuclear steam supply system [NSSS] power of 1817 MWt)
- A nominal RCS hot full power (HFP) T_{avg} range of 564.6° to 576.0°F. This provides an RCS T_{cold} range of 528.3° to 540.2°F.
- An RCS temperature uncertainty of ±4.0°F
- A feedwater temperature range of 390.0° to 435.0°F
- A nominal RCS pressure of 2250 psia
- A pressurizer pressure uncertainty of ±60 psi

Based on these conditions, the LOCA forces were generated at a minimum T_{cold} of 524.3°F, including uncertainty, and a pressurizer pressure of 2310 psia, including uncertainty.

GDC-4 allows main coolant piping breaks to be "...excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." This exemption is generally referred to as leak-before-break (LBB). Furthermore, Ginna had requested Westinghouse to exempt all the 10-inch piping connections to the RCS from the dynamic analysis of pipe break loads. Therefore, the next limiting RCS break sizes less than 10-inch diameter are the smaller auxiliary (or branch) lines connected to the RCS. The smaller branch line breaks analyzed for hydraulic forces are the 3-inch pressurizer spray line in the cold leg, the 4-inch upper plenum injection nozzle on the vessel, and the 2-inch safety injection line connection to the hot leg. The 4-inch pressurizer safety valve line on top of the pressurizer was not considered for the Forces analysis because the Forces analysis tracks the acoustic wave propagating through the subcooled fluid of the RCS, while the break for the safety valve line would occur in the voided region of the pressurizer. It would therefore be non-limiting as compared to breaks modeled in either the cold or hot legs of the RCS.

2.8.5.6.3.5.3 Description of Analyses and Evaluations

LOCA forces were generated with a focus on the component of interest loop, vessel, steam generator, or rod control cluster assembly (RCCA) guide tubes using the advanced beam model version of MULTIFLEX 3.0 (Reference 9), assuming a conservative break opening time of 1 millisecond.

Generally, this improved modeling results in lower, more realistic, but still conservative, hydraulic forces on the core barrel.

The MULTIFLEX computer code calculated the thermal-hydraulic transient within the RCS and considered subcooled, transition, and early two-phase (saturated) blowdown regimes. The code used the method of characteristics to solve the conservation laws, assuming one-dimensional (1-D) flow and a homogeneous liquid-vapor mixture. The RCS was divided into sub-regions in which each subregion was regarded as an equivalent pipe. A complex network of these equivalent pipes was used to represent the entire primary RCS.

For the reactor pressure vessel (RPV) and specific vessel internal components, the MULTIFLEX code generated the LOCA thermal-hydraulic transient that was input to the LATFORC and FORCE2 post-processing codes (Reference 11). These codes, in turn, were used to calculate the actual forces on the various components.

These forcing functions for horizontal and vertical LOCA hydraulic forces, combined with seismic, thermal, and flow induced vibration loads, were used in the structural evaluations to determine the resultant mechanical loads on the vessel and vessel internals. The vessel forces results are provided for use in the analyses described in LR section 2.2.3 Reactor Pressure Vessel Internals and Core Supports – Mechanical System Evaluations.

The loop forces analysis used the THRUST post-processing code to generate the X, Y, and Z directional component forces during a LOCA blowdown from the RCS pressure, density, and mass flux calculated by the MULTIFLEX code. The THRUST code is described and documented in WCAP-8252 (Reference 12). The loop forces results are provided for use in the analyses described in LR section 2.2.2.2.1, NSSS Piping, Components, and Supports, and in LR section 2.2.1 Pipe Rupture Locations and Associated Dynamic Effects.-

Steam generator hydraulic transient time-history data were extracted directly from the MULTIFLEX output, and evaluated against the steam generator data for the RSG project. Similarly, hydraulic transient time-history data used in qualification of some reactor vessel internal components, such as baffle bolts or RCCA guide tubes, were also extracted directly from the MULTIFLEX output.

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

See LR section 2.8.5.6.3.2.3.

2.8.5.6.3.5.4 Results

For the Ginna Station EPU, all relevant LOCA hydraulic forces analyses were performed directly at the analyzed NSSS power level of 1817 MWt, using models specific to the Ginna Station NSSS design. An evaluation which confirmed the acceptability of the steam generator forces was performed, and analyses of the forces acting on the reactor vessel internals, fuel, loop piping, and RCCA guide tube forces were performed. The results of the analyses were then used as input to the calculations for component qualification.

Discussion of Margin Change

As previously mentioned, the LOCA Forces are used as input to the various structural analyses, so margin quantification would be appropriately derived from the calculations for the specific component. Qualitatively speaking, margin in the Forces analyses is realized by analyzing smaller diameter lines, because larger diameter lines would yield higher forces.

2.8.5.6.3.6 References

1. 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

2. Letter from M.G. Korsnick (Ginna) to D.M. Skay (NRC), Subject: Revised LOCA Analyses – Changes to Accumulator, Refueling Water Storage Tank and Administrative Control Technical Specifications, dated April 29, 2005.

3. L. E. Hochreiter, W. R. Schwarz, K. Takeuchi, C-K Tsai, and M. Y. Young, Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, WCAP-10924-P-A, Volume 1, Revision 1, and Addenda 1, 2, and 3 (Proprietary Version), December 1988

4. S. I. Dederer, L. E. Hochreiter, W. R. Schwarz, D. L. Stucker, C-K Tsai, and M. Y. Young, 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 Addendum 1 (Proprietary Version), December 1988

5. S. I. Dederer, et. al., "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 (Proprietary Version), October 1999

6. WCAP-10079-P-A, (Proprietary), and WCAP-10080-A, (Non-Proprietary), NOTRUMP - A Nodal Transient Small Break And General Network Code, Meyer, P. E., August 1985.

7. WCAP-10054-P-A, (Proprietary), and WCAP-10081-A, (Non-Proprietary), Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code, Lee, N., Et Al., August 1985.

8. WCAP-11145-P-A, (Proprietary), and WCAP-11372-A, (Non-Proprietary), Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code, Rupprecht, S. D., Et Al., October 1986.

9. WCAP-9735, Rev. 2, (Proprietary), and WCAP-9736, Rev. 1, (Nonproprietary), *MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model*, K. Takeuchi, et al., February 1998.

10. WCAP-15029-P-A, (Proprietary), and WCAP-15030-NP-A, (Nonproprietary), *Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel*

Bolting Distributions Under Faulted Load Conditions, R. E. Schwirian, et al., January 1999.

11. WCAP-8708-P-A, (Proprietary) and WCAP-8709-A, (Nonproprietary), *MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics*, K. Takeuchi, et al., September 1977.

12. WCAP-8252, (Nonproprietary), Rev. 1, *Documentation of Selected Westinghouse Structural Analysis Computer Codes*, K. M. Vashi, May 1977.

2.8.5.6.3.7 Conclusion

The Ginna staff has reviewed the analyses of the LOCA events and the ECCS. The Ginna staff concludes that the analyses have adequately accounted for plant operation at the proposed power level and that the analyses were performed using acceptable analytical models. The Ginna staff further concludes that the evaluation has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the Ginna staff concludes that the Ginna Station will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-4, GDC-27, GDC-35, and 10CFR50.46 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the LOCA.

Table 2.8.5.6.3-1

Ginna Best Estimate Large Break LOCA Results

10CFR50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1870	< 2200
95/95 LMO[1] (%)	3.4	< 17
95/95 CWO[2] (%)	0.30	< 1

Note, these results are applicable to both the new fuel (422V+) and the resident fuel (OFA)

Table 2.8.5.6.3-2

Best Estimate Large Break LOCA Results for Extended Power Uprate Versus Current Licensing Basis

10CFR50.46 Requirement	Extended Power Uprate Analysis	Current Licensing Basis
95/95 PCT (°F)	1870	2087
95/95 LMO[1] (%)	3.4	< 17
95/95 CWO[2] (%)	0.30	< 1

1 Local Maximum Oxidation
2 Core Wide Oxidation

Table 2.8.5.6.3-3

Ginna Small Break LOCA Results

Parameter	Criterion	EPU Results	Current UFSAR
PCT	≤ 2200°F	1167°F	1381°F
Maximum Cladding Oxidation	≤ 17%	0.07%	0.074%
Maximum Core-Wide Oxidation	≤ 1%	< 0.07%	< 0.074%
Coolable Geometry	Yes	Yes	Yes

2.8.5.7 Anticipated Transients Without Scrams

2.8.5.7.1 Regulatory Evaluation

Anticipated transients without scram (ATWS) are defined as an anticipated operational occurrence followed by the failure of the reactor portion of the protection system as specified in GDC-20. The regulation at 10CFR50.62 (reference 1) requires that:

- Each pressurized water reactor (PWR) must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent of the existing reactor trip system, and.

- Each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a diverse scram system. This scram system must be designed to perform its function in a reliable manner and be independent of the existing reactor trip system.

The Ginna Nuclear Power Plant, LLC (Ginna) staff review was conducted to ensure that

- The above requirements were met, and

- That the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) remained valid for the EPU.

In addition, for plants where a diverse scram system is not specifically required by 10CFR50.62 (Reference 1), Ginna verified that the consequences of an ATWS were acceptable. Ginna Station is not required to install a diverse scram system. The acceptance criterion is that the peak primary system pressure should not exceed the ASME Boiler and Pressure Vessel Code (B&PV) (Reference 2), Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. The Ginna staff reviewed:

- The limiting event determination
- The sequence of events
- The analytical model and its applicability
- The values of parameters used in the analytical model
- The results of the analyses

The Ginna staff reviewed the justification of the applicability of generic vendor analyses to its plant and the operating conditions for the EPU. Review guidance was provided in Matrix 8 of RS-001.

Ginna Current Licensing Basis

The Final ATWS Rule, 10CFR50.62(c)(1) (Reference 1), requires the incorporation of a diverse (from the reactor trip system) actuation of the system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMSAC design, which is described in Ginna UFSAR Section 7.6.2, satisfies this Final ATWS Rule. The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in NS-TMA-2182 (Reference 3). To remain consistent with the basis of the Final ATWS Rule and the supporting analyses documented in NS-TMA-2182 (Reference 3), the peak RCS pressure reached should not exceed the ASME B&PV Code, (Reference 2) Service Level C service limit stress criteria of 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (RPV) (the nozzle safe end). There are no changes to the methods or acceptance criteria applied for ATWS in support of the EPU. The analysis of the ATWS event is described in Ginna UFSAR Section 15.8.

In addition to the evaluations described in the UFSAR, the Ginna loss of forced reactor coolant flow analysis 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 ATWS analysis is not within the scope of license renewal.

2.8.5.7.2 Technical Evaluation

2.8.5.7.2.1 Introduction

The final ATWS Rule, 10CFR50.62(c)(1) (Reference 1), requires the incorporation of a diverse (from the reactor trip system) actuation of the auxiliary feedwater (AFW) system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMSAC satisfies this Final ATWS Rule. The basis for this rule and the AMSAC design are supported by Westinghouse generic analyses documented in NS-TMA-2182 (Reference 3). These analyses were performed based on guidelines published in NUREG-0460 (Reference 4).

NS-TMA-2182 (Reference 3) also references WCAP-8330 (Reference 5) and subsequent related documents, which formed the initial Westinghouse submittal to the NRC for ATWS, and which were based on the guidelines set forth in WASH-1270 (Reference 6). For operation at EPU conditions, the Westinghouse generic ATWS analyses (Reference 3) were evaluated for their continued applicability.

NS-TMA-2182 (Reference 3) describes the methods used in the analysis and provides reference analyses for two-loop, three-loop, and four-loop plant designs with several different steam generator models available in plants at that time. The reference analysis results demonstrated that the Westinghouse plant designs would satisfy the criteria in NUREG-0460 (Reference 4).

The failure of the reactor scram is presumed to be a common mode failure of the control rods to insert into the core. The assumption of this common mode failure is beyond the requirement to address a single failure in the typical UFSAR transient analyses. In addition, the methodology of NS-TMA-2182 (Reference 3) uses control-grade equipment to mitigate consequences of the event, and uses nominal system performance characteristics in the evaluation of the event.

2.8.5.7.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The primary input to the loss of normal feedwater (LONF) ATWS evaluation for the EPU was the reference two-loop ATWS model and the reference ATWS analysis that were performed for an NSSS power level of 1520 MWt (Reference 3). The nominal and initial conditions were updated to the EPU NSSS design Performance Capability Working Group (PCWG) parameters for 1817 MWt, and the steam generator data were revised to reflect the current (replacement) steam generator parameters and heat transfer characteristics.

The reference ATWS analyses (Reference 3) were performed assuming that a plant will have a MTC that is more negative than -8 pcm/°F for 95% of the cycle. The ATWS evaluation for the EPU assumed a more positive MTC at hot full power (HFP) of -5.5 pcm/°F.

The Final ATWS Rule, 10CFR50.62(c)(1) (Reference 1), requires the incorporation of a diverse (from the reactor trip system) actuation of the system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMSAC design satisfies this Final ATWS Rule. The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in NS-TMA-2182 (Reference 3). To remain consistency with the basis of the Final ATWS Rule and the supporting analyses documented in NS-TMA-2182 (Reference 3), the peak RCS pressure reached in the Ginna EPU ATWS evaluation with current steam generators should not exceed the ASME B&PV Code, (Reference 2) Service Level C service limit stress criteria of 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component of the reactor pressure vessel (RPV), the nozzle safe end.

2.8.5.7.2.3 Description of Analyses and Evaluations

An evaluation was performed to assess the effect of the EPU on the reference two-loop LONF ATWS analysis documented in NS-TMA-2182 (Reference 3). The evaluation included revision of the reference two-loop ATWS model at a power level of 1520 MWt (Reference 3) to reflect the plant conditions at an NSSS power level of 1817 MWt, and the steam generator data were revised to incorporate the current steam generator parameters and heat transfer characteristics.

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

The ATWS analysis is not within the scope of license renewal since it is an analytical product of a postulated event. Systems and system components associated with this analysis that are within the scope of license renewal are addressed in their respective system sections 2.3.1 (Reactor Systems), 2.3.4 (Steam and Power Conversion Systems), and 2.5 (Electrical and Instrumentation and Controls) of NUREG-1786. Aging effects, and the programs used to manage the aging effects of these components are discussed in NUREG-1786 sections 3.1 (Reactor Coolant Systems), 3.4 (Steam and Power Conversion Systems), and 3.6 (Electrical and Instrumentation and Controls).

EPU activities do not add any new components nor do they introduce any new functions for existing plant components relied upon to mitigate the effects of postulated ATWS events that would change the license renewal evaluation boundaries. The system and component performance capability in response to postulated ATWS events described in this section for the proposed EPU LR involves analytical techniques and methodology which are unaffected by the proposed EPU, and the results of which remain bounded by the acceptance criteria of 10CFR50.62. Therefore, no new aging effects requiring management for the extended term of the operating license are identified with respect to performance capability of systems and components relied upon to mitigate the effects of postulated ATWS events.

2.8.5.7.3 Results

To remain consistency with the basis of the Final ATWS Rule and the supporting analyses documented in NS-TMA-2182 (Reference 3), the peak RCS pressure reached in the Ginna EPU ATWS evaluation with current steam generators should not exceed the ASME BP&E Code (Reference 2) Service Level C limit stress criteria of 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component of the RPV, the nozzle safe end.

The results of the LONF ATWS evaluation, using the revised reference two-loop LONF ATWS model at an NSSS power of 1817 MWt with current generators, demonstrated that the resulting peak RCS pressure was lower than the ASME B&PV Code (Reference 2) Service Level C limit stress criteria of 3200 psig. Therefore, the analytical basis for the Final ATWS Rule continued to be met for operation of Ginna for the EPU with current steam generators.

2.8.5.7.4 References

1. 10CFR50.62, *Requirements for Reduction of Risk from ATWS Events for Light Water-Cooled Nuclear Power Plants.*

2. *ASME Boiler and Pressure Vessel Code*, The American Society of Mechanical Engineers.

3. NS-TMA-2182, *Anticipated Transients Without Scram for Westinghouse Plants*, December 1979.

4. NUREG-0460, *Anticipated Transients Without Scram for Light Water Reactors*, April 1978.

5. WCAP-8330, *Westinghouse Anticipated Transient Without Trip Analysis*, August 1974.

6. NRC Report WASH-1270, *Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors*, September 1973.

2.8.5.7.5 Conclusion

The Ginna staff has reviewed the information related to ATWS and concludes that it has adequately accounted for the effects of the proposed EPU on ATWS. The Ginna staff concludes that the evaluation has demonstrated that the AMSAC will continue to meet the requirements of 10CFR50.62 following implementation of the proposed EPU. The evaluation has shown that the plant is not required by 10CFR50.62 to have a diverse scram system. Additionally, the evaluation has demonstrated, as explained above, that the peak primary system pressure following an ATWS event will remain below the acceptance limit of 3200 psig. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

2.8.6.1.1 Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

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 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 new fuel storage relative to conformance to:

- GDC-62 is described in Ginna UFSAR section 3.1.2.6.3, which states that 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. Criticality prevention is discussed in detail in UFSAR section 9.1.2.

In addition to the evaluations described in the UFSAR, the new fuel storage 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 new fuel storage system is addressed in section 2.3.3.3, Spent Fuel Pool Cooling and Fuel Storage. The concrete supporting structure for the new fuel pool and storage racks is discussed in SER Section 2.4.2.1, Auxiliary Building. Aging effects, and the programs used to manage the aging effects associated with new fuel storage, are discussed in section 3.3 of the SER for new fuel racks, and in SER section 3.5 for concrete structure of the new fuel pool.

Ginna has been granted an exemption from the requirements of 10CFR70.24 concerning criticality monitors (Reference 1).

2.8.6.1.2 Technical Evaluation

2.8.6.1.2.1 Introduction

The purpose of this section is to provide an assessment of the effect of fuel design changes for the Ginna extended power uprate (EPU) on the current analysis of record for the new fuel storage vault. Section 9.1.2.4.1 of the UFSAR describes the current criticality licensing basis for the new fuel storage vault.

2.8.6.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The analysis of the new fuel storage vault is completely independent of the core power level. However, as part of the EPU program the fuel assembly design will be changed to the 14x14 422V+ design (See Section 2.8.1, Fuel System Design) which affects the criticality analysis of the new fuel. The key features of the 14x14 422V+ fuel assembly design that affect the new fuel storage vault criticality analysis are pellet diameter and fuel stack height. The 14x14 422V+ design includes a fuel pellet diameter equal to 0.3659 inches and a fuel stack height equal to 143.25 inches. These changes have been evaluated for their affect on the existing licensing basis.

The acceptance criterion for the new fuel storage vault criticality analysis requires that the maximum K-effective value, including all biases and uncertainties, is less than 0.98 with optimum moderation condition. In addition, the maximum K-effective for the new fuel storage vault, with full density unborated water, must be less than 0.95.

2.8.6.1.2.3 Description of Analyses and Evaluations

The design method that ensures the criticality safety of fuel assemblies in the new fuel storage rack is described in section 9.1.2.4.1 of the Ginna UFSAR. This design method uses the AMPX system of codes for cross-section generation and KENO Va for reactivity determination. The selection of design basis fuel assembly types was based on a survey of the variety of fuel assemblies employed in the reactor and selecting the most reactive type. The candidate fuel assembly types included the Westinghouse Standard and Westinghouse OFA designs. The design basis fresh fuel assembly identified in the Ginna UFSAR is the Westinghouse OFA assembly.

The criticality evaluation for the new fuel storage building is based on past studies employing similar methods, fuel designs and storage cell characteristics (Reference 2). These past studies have demonstrated that the maximum reactivity for fresh fuel assemblies is limited by the relatively large center-to-center spacing of the cells in the new fuel storage vault. At full density moderation, the assemblies are neutronically decoupled. A single 14x14 fuel assembly, enriched to 5.0 w/o U-235, fully moderated and reflected meets the criticality design basis with both the OFA pellet diameter and pellet diameter of the 422V+ fuel assembly design. For low-density, optimum moderation conditions, the OFA fuel assembly design has been shown to easily meet the criticality design basis, with a calculated k_{eff} of 0.667, due to the neutron leakage that is inherent with the large center-to-center spacing between fuel assemblies in the new fuel storage racks. The reactivity of the 422V+ fuel assembly under low-density, optimum moderation conditions will meet the criticality design basis due to the similarly large neutron leakage term. The analysis of the new fuel storage vault is completely independent of the core power level, and is therefore unaffected by the EPU.

While the evaluation is based on geometric equivalence to the proposed fuel design, the current analysis of record contained in the UFSAR was performed for a fuel height equal to 141.4 inches. The fuel height that will be employed for the EPU program is 143.25 inches. Based on experience, this small change has only a minor effect on the overall reactivity results, and the existing margins to the acceptance criteria will not be challenged.

The current licensing basis analysis supports enrichments up to 5.0 weight percent, which bounds the enrichment required for the EPU.

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

The new fuel storage rack structures, included neutron shielding, and the concrete supporting structure of the auxiliary building are within the scope of License Renewal as identified in the License Renewal Safety Evaluation Report, NUREG-1786, Sections 2.3.3.3 and 2.4.2.1. Aging effects, and the programs used to manage the aging effects associated with new fuel pool storage, 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 new fuel storage system. System component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.8.6.1.2.4 Results

Based on the evaluation outlined above, the effects of the EPU and the fuel transition to 14x14 422V+ will provide results which maintain essentially the same margins as those presented in the current licensing basis. Therefore, the effective neutron multiplication factor, k_{eff}, in the fresh fuel storage racks will be less than or equal to 0.95, including uncertainties, under flooded conditions and less than or equal to 0.98, including uncertainties, under optimally moderated conditions for the storage of Westinghouse 14x14 OFA or 422V+ (STANDARD) fuel assemblies with nominal enrichments up to 5.0 weight percent. Compliance with these conditions meets the requirements of GDC-62.

2.8.6.1.2.5 References

1. Letter, Vissing, G. S. (NRC) to Mecredy, R. C. (RGE), *Issuance of Exemption from the Requirements of 10CFR70.24 - R. E. Ginna Nuclear Power Plant (TAC No. M98926)*, July 16, 1997.

2. Attachment A to Letter, Mecredy, R. C. (RGE) to Johnson, A. R. (NRC), *Technical Specification Improvement Program, Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244*, May 5, 1995.

2.8.6.1.3 Conclusion

The effect of the new 14x14 422V+ fuel that will be introduced with the EPU on the analyses of new fuel storage facilities has been evaluated and Ginna concludes that the geometric configuration of the new fuel storage racks will continue to ensure that criticality requirements are met.. Therefore, the current licensing basis requirements with respect to GDC-62 will be satisfied following implementation of the proposed EPU.

2.8.6.2 Spent Fuel Storage

2.8.6.2.1 Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The Ginna Nuclear Power Plant, LLC (Ginna) review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria 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 62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations

Specific review criteria are contained in SRP Section 9.1.2.

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, 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 spent fuel storage 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 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, as related to spent fuel storage, is described in the following:

Ginna UFSAR, Section 3.5.2
- Externally Generated Missiles

Ginna UFSAR, Section 9.1.2
- Spent Fuel Storage

- GDC-62 is described in Ginna UFSAR section 3.1.2.6.3, which states that 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. Criticality prevention is discussed in detail in UFSAR section 9.1.2.

In addition to the evaluations described in the UFSAR, the spent fuel storage 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 spent fuel storage system is described in section 2.3.3.3, Spent Fuel Pool Cooling and Fuel Storage. The concrete supporting structure for the spent fuel pool and storage racks is discussed in section 2.4.2.1, Auxiliary Building. Aging effects, and the programs used to manage the aging effects associated with spent fuel storage, are discussed in section 3.3 of the SER for spent fuel racks, and in section 3.5 for concrete structure of the spent fuel pool.

2.8.6.2.2 Technical Evaluation

2.8.6.2.2.1 Introduction

The purpose of this section is to provide an assessment of the effect of EPU changes on the current analysis of record for the spent fuel pool. Section 9.1.2.4.1 of the UFSAR describes the current criticality licensing basis for the spent fuel pool. The effects of EPU on the spent fuel pool cooling system are evaluated in <u>LR section 2.5.4.1</u>.

2.8.6.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The analysis of the spent fuel pool is completely independent of the core power level. However, as part of the EPU program the fuel assembly design will be changed to the 14x14 422V+ design (See Section 2.8.1, Fuel System Design) which affects the criticality analysis of the spent fuel. The key features of the 14x14 422V+ fuel assembly design that affect the spent fuel pool criticality analysis are pellet diameter and fuel height. The 14x14 422V+ fuel assembly design has a fuel pellet diameter of 0.3659 inches and a fuel stack height of 143.25 inches. These changes will be evaluated for their affect on the existing licensing basis.

The acceptance criteria for the spent fuel pool criticality analysis require that the maximum K-effective value, including all biases and uncertainties, is less than 1.0 with full density unborated water. In addition, the maximum K-effective for the spent fuel pool, with credit for soluble boron, must be less than 0.95.

The new fuel assembly dry weight with 96.5% theoretical density pellets is 1262 lbs. The weight of an RCCA is approximately 140 lbs, so the maximum weight of an assembly with insert is 1402 lbs. The structural analyses for the spent fuel racks used 1450 lbs as the weight of a fuel assembly with insert, and 2638 lbs as the maximum weight of a consolidated fuel canister.

2.8.6.2.2.3 Description of Analyses and Evaluations

The licensing basis analysis methods employ SCALE-PC, with the updated version of the 44 group ENDF/B-V neutron cross section library, and the two-dimensional integral transport code DIT. Validation of SCALE-PC and DIT are described in Ginna UFSAR section 9.1.2.4.1.

The selection of design basis fuel assembly types was based on an evaluation of the variety of fuel assemblies employed in the reactor and selecting the most reactive type for a given evaluation. The candidate fuel assembly types included the Westinghouse 14x14 STANDARD and 14x14 OFA designs. Studies discussed in the Ginna UFSAR have concluded that the Westinghouse 14x14 STANDARD assembly becomes more reactive than the OFA assembly beyond a typical Middle-of-Life burnup. Thus, the design basis burned fuel assembly employed for these analyses is taken to be a variant of the Westinghouse 14x14 STANDARD fuel assembly because of its burnup characteristics.

The Westinghouse 14x14 STANDARD fuel assembly, which was used at Ginna in fuel cycles 1A through 10, had a fuel pellet diameter of 0.3659 inches and a fuel stack height of 141.4 inches. The current licensing basis analysis uses a fuel pellet diameter of 0.3669 inches and a fuel stack height of 144 inches to bound the 14x14 STANDARD assembly.

The 14x14 422V+ fuel assembly that will be introduced with the EPU has the same 0.3659 inch pellet diameter as the 14x14 STANDARD design and is therefore bounded by the current analysis value of 0.3669 inches. In addition, the 143.25 inch fuel stack of the 14x14 422V+ assembly is bounded by the current analysis fuel stack height of 144 inches.

The EPU core power level (1775 MWt) does not change the limiting axial burnup profile that was assumed in the UFSAR analysis. Consequently, the current analysis contained in the

UFSAR bounds the overall reactivity results of the 14x14 422V+ design, and the existing margins to the acceptance criteria will not be challenged.

The current licensing basis analysis nominal enrichment limit of 5.0 weight percent bounds the enrichment required for the EPU.

The impact of the post-EPU fuel change on analyses regarding environmental effects performed in accordance with GDC-4 was considered. The only potentially affected analysis is for tornado missile strikes. The radiological consequences of a tornado missile accident were re-evaluated with the higher post-EPU source terms and found to be acceptable. The consequences of the tornado missile accident are discussed in detail in Section 2.9.

The weight of the new fuel assembly that will be introduced at the time of the uprate is bounded by the weights of the fuel assembly and consolidated canister used in the current fuel rack structural analyses, so those analyses remain bounding.

The increased gamma flux from recently discharged fuel assemblies would cause an increase in heat generation rates in the concrete walls of the spent fuel pool compared to the current fuel assemblies. Ginna will implement measures to ensure that only low-power assemblies are discharged into cells immediately adjacent to the walls of the spent fuel pool. The attenuation of the gamma flux from high-power assemblies that are at least one cell away from the wall will be greater than the increase in gamma flux due to EPU.

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

The spent fuel storage rack structures, included neutron shielding, and the concrete supporting structure of the auxiliary building are within the scope of License Renewal as identified in the License Renewal Safety Evaluation Report, NUREG-1786, Sections 2.3.3.3 and 2.4.2.1. Aging effects, and the programs used to manage the aging effects associated with spent fuel pool storage, 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 storage system. Notwithstanding the potential for a slight increase in the boric acid concentration in the spent fuel pool water, system component internal and external environments remain within the parameters previously evaluated. Thus, no new aging effects requiring management are identified.

2.8.6.2.2.4 Results

Based on the evaluation outlined above, the effects of the EPU and the fuel transition to 14x14 422V+ will provide results which maintain essentially the same margins as those in the current licensing basis. Therefore, the conclusion in the Ginna UFSAR that the total soluble boron concentration required to maintain k_{eff} less than 0.95, after including all biases and uncertainties and assuming the most limiting accident, is less than or equal to 965 ppm remains valid. Compliance with this condition meets the requirements of GDC-62. With the continued acceptability of doses resulting from a tornado missile strike, all potential environmental effects remain acceptable and thus the requirements of GDC-4 continue to be met.

The analyses in Section 2.9 demonstrate that the radiological consequences of a tornado missile accident remain acceptable.

The current fuel rack structural analyses remain bounding after the introduction of 14x14 422V+ fuel with the EPU.

The increased gamma flux from the fuel following EPU will be addressed by limiting the placement of high-power discharge assemblies to locations that are not immediately adjacent to the SFP walls.

2.8.6.2.3 Conclusion

The effect of the proposed EPU on the spent fuel storage capability has been evaluated and Ginna concludes that the effects of the proposed EPU on the Spent Fuel Storage system are adequately addressed. The evaluation also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, it is concluded that the spent fuel storage facilities will continue to meet the Ginna Station current licensing basis requirements with respect to GDC 4 and GDC 62 following implementation of the proposed EPU.

2.8.7 Additional Reactor Systems

2.8.7.1 Auxiliary Systems Pumps, Heat Exchangers, Valves, and Tanks

2.8.7.1.1 Introduction

The Ginna Nuclear Power Plant, LLC (Ginna) staff evaluated the Ginna Station auxiliary pumps, heat exchangers, valves, and tanks that are in the systems affected by the EPU for impact by the thermal transients and maximum operating temperatures and pressures resulting from EPU conditions. The evaluation consisted of a review of uprate parameters compared to equipment specifications for equipment supplied for Ginna Station.

2.8.7.1.2 Regulatory Evaluation

The Plant Auxiliary Systems components consist of the pumps, heat exchangers, valves, and tanks listed in Tables 2.8.7.1-1 through 2.8.7.1-4, below, that are the essential components of the various safety-related and non-safety-related plant auxiliary systems. These components include those installed in the reactor auxiliary cooling water systems required for safe shutdown of the plant during all conditions and for accident prevention and/or mitigation.

The Ginna staff's review of reactor auxiliary cooling water system components focused on the effects of the proposed EPU on the various systems' components continued functionality, including the capability to provide heat sink capacity, and withstand any adverse dynamic loads (e.g., water hammer, flow-induced vibration, thermal transients, maximum operating temperatures, pressures and flow rates). Ginna's acceptance criteria for these Auxiliary Systems' components are based on 10CFR50, Appendix A, General Design Criteria (GDC) as follows:

- GDC-2 requires that safety-related systems, structures and components (SSCs) be designed to withstand the effects of earthquakes.

- GDC-4 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-5 addresses sharing of SSCs among nuclear power units which is not applicable to the Ginna Station since it is a single unit installation.

- GDC-34 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.

- GDC-44 requires a system with the capability to transfer heat loads from SSCs important to safety to an ultimate heat sink under normal operating and accident conditions. Suitable component redundancy, isolation capability, leak detection, and ability to withstand single failure shall be provided

- GDC-61 requires that fuel storage and handling systems be designed to assure adequate safety under normal and accident conditions. System design shall include RHR capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of coolant inventory under accident conditions.

Specific review criteria are provided in Standard Review Plan (SRP) sections 5.4.7, 9.1.3, 9.2.1, 9.2.2, and 9.3.4, as supplemented by the guidance documents cited in Matrices 1, 2, 5 and 8 of NRC RS-001, Revision 0.

Ginna Current Licensing Basis

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 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 and to reconfirm and document their safety. The results of the SEP review for Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), which was completed in August 1983. The IPSAR describes the methods used by the NRC to assess the 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 Auxiliary Systems Components was assessed relative to conformance to:

- 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. 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:

- o UFSAR section 3.11, "Environmental Design Of Mechanical And Electrical Equipment"
- o UFSAR section 3.6, "Protection Against The Dynamic Effects Associated With The Postulated Rupture Of Piping"
- o UFSAR section 3.6 (SEP Topic III-5.A), "Pipe Breaks Inside Containment"
- o UFSAR section 3.6, (SEP Topic III-5.B), "Pipe Breaks Outside Containment"
- o UFSAR section 3.5, (SEP Topic III-4.C), "Missile Protection"

- 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. The UFSAR section 5.4.5, Residual Heat Removal (RHR) System, describes how the system is used to transfer fission product decay heat and other residual heat from the reactor core.

- GDC-44 is described in Ginna UFSAR section 3.1.2.4.15. General Design Criterion 44, Cooling Water, which states that a system to transfer heat from SSCs important to safety to the ultimate heat sink shall be provided. The safety function is to transfer the combined heat load of these SSCs under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure the safety function can be accomplished, assuming a single failure.

- GDC-61 is described in Ginna UFSAR section 3.1.2.6.2. General Design Criterion 61, Fuel Storage and Handling, and Radioactivity Control. The spent fuel pool (SFP) and cooling system, fuel handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed to ensure adequate safety under normal and postulated accident conditions.

In addition to the evaluations described in the UFSAR, the Ginna Station's auxiliary systems components were evaluated for plant license renewal. 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," dated May 2004.

The auxiliary systems components evaluated for the proposed EPU are described in NUREG-1786, sections 2.3.2 and 2.3.3, and the programs used to manage the aging effects associated with these components are discussed in NUREG-1786, section 3.3.

2.8.7.1.3 Technical Evaluation

2.8.7.1.3.1 Input Parameters, Assumptions, and Acceptance Criteria

The auxiliary system pumps are listed in Table 2.8.7.1-1, the heat exchangers are listed in Table 2.8.7.1-2, the valves evaluated are listed in Table 2.8.7.1-3, and the tanks are listed in Table 2.8.7.1-4. The component design information is contained in specific design documents. Tables 2.8.7.1-1 through 2.8.7.1-4 represent the original as-shipped components evaluated, and any component changes or replacements were reconciled under the plant quality assurance program to be in accordance with the original Westinghouse technical and quality assurance requirements.

2.8.7.1.3.2 Description of Analyses and Evaluations

The system design parameters as part of the System review were compared to the EPU conditions. The design conditions included design temperature, pressure, and flow rate. The auxiliary equipment technical documentation was then reviewed to establish the equipment original design conditions. The specified criteria again included design temperature, pressure, and flow rate. These parameters were compared to those used in the system review for the EPU to determine if the design parameters continue to bound those for the EPU.

Auxiliary System Pumps

The auxiliary pumps that Westinghouse provided to the Ginna Station are listed in Table 2.8.7.1-1. Based on the EPU conditions presented in LR Table 1-1, there was no impact on the auxiliary system pumps. The operating temperature and pressure ranges for these pumps remained bounded by the original design parameters. The EPU design transients are bounded by the original design transients for the auxiliary equipment. Based on this, the pumps are not impacted by the EPU.

Auxiliary System Heat Exchangers and Tanks

The auxiliary heat exchangers and tanks that Westinghouse provided to the Ginna Station are listed in Tables 2.8.7.1-2 and 2.8.7.1-4, respectively. Based on the EPU conditions presented in LR Table 1-1, there was no impact on the auxiliary system heat exchangers and tanks as a result of the EPU. The operating temperature and pressure ranges for these components remained bounded by the original design parameters. The EPU design transients are bounded by the original design transients for the auxiliary equipment. As a result, the heat exchangers and tanks are not impacted by the EPU.

Auxiliary System Valves

The auxiliary valves that Westinghouse provided to the Ginna Station are listed in the Westinghouse Valve Index by Locations/Sizes and Systems for the Ginna Station. Specifications are listed in Table 2.8.7.1-3. Based on the EPU conditions presented in LR Table 1-1, there was no impact on the auxiliary system valves. The operating temperature and pressure ranges for the valves remained bounded by the original design parameters. In addition, the original design transients for the auxiliary equipment bounded the transients associated with the EPU. As a result, the valves are not impacted by the EPU.

2.8.7.1.4 Results

The revised design conditions have been evaluated with respect to the impact on auxiliary heat exchangers, valves, pumps and tanks. The effect of the EPU has no impact on these components since the EPU conditions are bounded by the original design conditions.

2.8.7.1.5 Conclusion

The Ginna staff has reviewed the assessment of the effects of the EPU on the auxiliary systems and concludes that the evaluation has adequately accounted for the effects of changes in plant conditions on the design of the auxiliary systems. Ginna concludes that the auxiliary systems will maintain their ability to perform their required function. Ginna further concludes that the auxiliary systems will continue to meet the Ginna Station current licensing basis requirements with respect to GDC-2, GDC-4, GDC-34, GDC-44, and GDC-61. Therefore, Ginna finds the EPU acceptable with respect to the auxiliary systems components.

Table 2.8.7.1-1
Ginna Station Auxiliary Pumps

Description	# Pumps	Vendor
Boric Acid Transfer Pump	2	Chempump
Charging Pump Accumulator	3	Robertshaw
Chemical Drain Tank Pump	1	Goulds
Component Cooling Water Pump	2	IDP (I-R)
Conc Holdup Tank Transfer Pump	2	Chempump
Containment Spray Pump	2	IDP (I-R)
Gas Stripper Feed Pump	2	Chempump
Laundry and Hot Shower Tank Pump	1	Goulds
Monitor Tank Pump	1	Goulds
Positive Displacement Charging Pump	3	Ajax Iron Works
RCS Drain Tank Pump	1	Chempump
RCS Drain Tank Pump	1	Chempump
Refueling Water Purification Pump	1	Goulds
Residual Heat Removal Pump	2	IDP (Pacific)
Safety Injection Pump	1	IDP (Worthington)
SIS Pump Bypass Orifice	3	Byron Jackson
Spent Fuel Pit Pump	1	IDP (I-R)
Spent Fuel Pit Skimmer Pump	1	Duriron
Sump Tank Pump	2	Goulds
Vacuum Pump Package	1	Heraeus Eng.
Vacuum Pump Package	1	Heraeus Eng.
Waste Evaporator Condenser Pump	2	Goulds
Waste Evaporator Feed Pump	1	Goulds
Waste Gas Compressor Package	2	Nash Engineering

Table 2.8.7.1-2
Ginna Station Auxiliary Heat Exchangers (HXs)

Component	Vendor
Regenerative HX	Sentry Equipment Corp.
Non-Regenerative HX	Sentry Equipment Corp.
Excess Letdown HX	Sentry Equipment Corp.
Seal Water HX	Atlas Industrial
Residual HX	Joseph Oat
Component Cooling HX	Atlas Industrial
Spent Fuel Pit HX	Atlas Industrial
Sample HX	Sentry Equipment Corp.

Table 2.8.7.1-3 Ginna Station Auxiliary Valves	
Specification Number	**Title**
676281, Rev. 2	Diaphragm-Type Valves (Saunders Patent)
676257, Rev. 2	Auxiliary Relief Valves
676258, Rev. 1	Motor-Operated Valves
676241, Rev. 1	Manual "T" and "Y" Globe, Manual Gate and Self-Actuated Check Valves
676368, Rev. 1	Butterfly Valves
676270, Rev. 1	Control Valves
676279, Rev. 2	Pressurizer Safety Valves

Table 2.8.7.1-4 Ginna Station Auxiliary Tanks	
Tank	**Drawing**
1. Boric Acid Tank	684J809
2. Boric Acid Batch Tank	684J926
3. Chemical Drain Tank	684J937
4. Chemical Mixing Tank	882D789
5. Component Cooling Surge Tank	684J700
6. Concentrates Holding Tank	684J734
7. Gas Decay Tank	541F375
8. Laundry and Hot Shower Tank	684J788
9. Leakoff Drains Collection Tank	EDSK329519
10. Monitor Tank	684J946
11. Pressurizer Relief Tank	684J694
12. Reactor Coolant Drain Tank	684J787
13. Reactor Makeup Water Tank	4155-B-326-003
14. Reagent Tank	883D429
15. Resin Fill Tank	541F165
16. Sample Vessels	675C744
17. Safety Injection Accumulator Tank	B-326-010
18. Spent Resin Storage Tank	684J794
19. Sump Tank	684J786
20. Sodium Thiosulfate Tank	U-9913
21. Volume Control Tank	647J071
22. Waste Condensate Tank	684J924
23. Waste Holdup Tank	684J772

2.8.7.2 Natural Circulation Cooldown

2.8.7.2.1 Regulatory Evaluation

On January 18, 1970, Ginna plant performed a series of Natural Circulation Cooldown tests that are documented in UFSAR section 14.6.1.5.6, "Natural Circulation Test." Natural Circulation Cooldown became an explicit issue for all plants following the Three Mile Island event. For the Ginna EPU, an evaluation is performed that shows that the natural circulation cooldown has consistent performance at the uprated conditions.

The following criteria are used to show acceptable natural circulation cooldown behavior.

- The natural circulation ΔTs and temperatures should be reasonable (e.g., bounded by full-power conditions). This helps to avoid any concerns with thermal stresses and also helps to ensure adequate reactor coolant system (RCS) subcooling.
- The steam generator atmospheric relief valves (ARVs) should be capable of cooling down the plant to residual heat removal system (RHR) cut-in conditions (~400 psig, 350°F in the RCS) within a reasonable time. Allowing for 4 hours at hot standby and an emergency operating procedure (EOP) maximum cooldown rate of 25°F/hour for natural circulation (reference 3), the time frame for RHR cut-in should be on the order of 12 to 14 hours.
- Compare the hydraulic flow resistance coefficients with that of Diablo Canyon.

Reference 1 describes the NRC Safety Evaluation Report (SER) on the Ginna response to Generic Letter (GL) 81-21. The letter states commitments by Ginna Nuclear Power Plant, LLC (Ginna), namely that there are adequate procedural changes in place, a training program on the procedures for natural circulation cooldown, and that there is sufficient condensate storage water.

Procedural guidance (reference 3) follows the Westinghouse Owners Group Emergency Response Guidelines (WOG ERGs) (reference 2) and maintains sufficient shutdown margin (SDM). Further, the procedures specify the maximum RCS cooldown rate, appropriate wait times, Upper Head Vessel Cooling (as a function of fan status), and monitoring for subcooled margin.

Ginna Current Licensing Basis

The UFSAR section 14.6.1.5.6,"Natural Circulation Test," documents the results of the natural circulation test that demonstrated that natural circulation occurred and was in good agreement with predictions.

Natural circulation cooldown is not being impacted by the License Renewal Program. Higher RCS ΔTs and reduced resistance in the core due to the use of 222V+ fuel tend to increase natural circulation capability.

2.8.7.2.2 Technical Evaluation

2.8.7.2.2.1 Introduction

The purpose of the natural circulation cooldown evaluation is to show that the plant at the EPU conditions exhibits expected natural circulation behavior, similar to that previously observed. A comparison is made to hydraulic parameters measured at other plants in the industry and related to Ginna conditions. In addition, the evaluation will demonstrate the ability to cool down the plant on natural circulation to RHR cut-in conditions (~400 psig, and less than 350°F in the RCS) within a reasonable period of time.

2.8.7.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The input parameters for the evaluation are given in Table 2.8.7.2-1. The assumptions are summarized in Table 2.8.7.2-2. While there are no formal acceptance criteria for this evaluation, the guidelines used are listed in LR section 2.8.7.2.1.

2.8.7.2.2.3 Description of Analyses and Evaluations

To evaluate the natural circulation capability for the Ginna Station EPU, the WOG ERG methodology (reference 2) is used to estimate flow rates and core delta temperatures using core hydraulic resistance coefficients. These equations are evaluated for several decay heat assumptions (1, 2, 3, and 4%) over a range of temperature conditions.

The estimated loop delta temperatures at selected decay heat levels are compared to the original natural circulation test results at a lower power level. The EPU calculated loop delta temperature values are multiplied by the power ratio (1300 MWt / 1811 MWt) raised to the 2/3 power to obtain calculated loop delta temperatures at the original Ginna startup power level. The resulting scaled loop delta temperatures are shown in Table 2.8.7.2-4.

In addition, the ARV capacities are estimated as function of steam generator secondary pressure that is correlated with primary system saturated temperature. After four hours at hot standby conditions, the plant is assumed to cool down to the RHR cut-in conditions at the maximum Emergency Operating Procedure (EOP) rate (25°F/hour per Reference 3).

The hydraulic flow resistance coefficients for Ginna Station are compared with those of Diablo Canyon in Table 2.8.7.2-3. The loop delta temperatures are compared to the originally measured values in Table 2.8.7.2-4. The Ginna measured results are taken from UFSAR Figure 14.6-10. The calculated flow rates and loop delta temperatures are given in Tables 2.8.7.2-5 and 2.8.7.2-6.

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

Natural circulation cooldown was not specifically evaluated by the License Renewal Program. Higher RCS ΔTs and reduced resistance in the core due to the use of 422V+ fuel tend to increase natural circulation capability. Thus, no further consideration for the effects on the renewed license is warranted.

2.8.7.2.3 Results

As shown in Table 2.7.7.2-3, there is close agreement between the hydraulic resistance coefficients for the Diablo Canyon and Ginna plants at the uprated conditions, and the loop flow ratios are in good agreement.

As shown in Table 2.8.7.2-4, the calculated loop delta temperatures for Ginna show the same trends and slightly higher scaled values compared to the Ginna UFSAR reported measured values.

The values in Tables 2.8.7.2-5 and 2.8.7.2-6 are presented to illustrate the parametric behavior at 1, 2, 3, and 4% of decay heat.

The natural circulation flow rates listed in Table 2.8.7.2-5 show expected behavior – decreases as the decay heat decreases at a constant temperature and a decrease with temperature at a constant value of decay heat.

The loop delta temperature results listed in Table 2.8.7.2-6 show expected behavior – decrease as the decay heat decreases at a constant core average temperature and increases as the core average temperature decreases at a constant value of decay heat.

The establishment of natural circulation cooldown conditions is expected to take several loop transits after the reactor coolant pumps (RCPs) have tripped. The decay heat will be 3% at ~ 3.5 minutes after a reactor trip. This time approximates the minimum time required for establishing natural circulation cooling following a trip of the RCPs and co-incident reactor trip. Thus the conditions for the evaluation at hot standby period will be bounded by using the 3%

decay heat value to assess natural circulation cooldown flow rates and natural circulation cooldown flow temperature differentials.

For the following reasons, the Ginna EPU analyzed at 1811 MWt will not adversely impact the natural circulation cooldown capability of the plant.

- Acceptable results are found for natural circulation cooling during the hot standby period for realistic residual heat rates as high as 3% of 1811 MWt. The maximum expected hot leg temperature calculated for this case (~600.8°F) is bounded by the maximum expected hot leg temperature for full power operation for the high Tavg cases (611.8 °F) (PCWG Cases 3 and 4, LR section 1.1,"Nuclear Steam Supply System Parameters", Table 1-1).

- The calculated loop delta temperatures are scaled and compared to the UFSAR measured values. The scaled, calculated values show the same trends as the original measurements and are slightly larger than measured, due to several conservative assumptions in the calculations One of the conservative assumptions is that the hydraulic resistance for the RCP during natural circulation cooling is based upon a locked rotor hydraulic resistance value.

- The ARVs at the EPU conditions are adequate to achieve cooldown to the RHR entry point in a reasonable time period. RHR cut-in conditions can be achieved in ~ 14 hours at the maximum rate specified in ES-0.2, which includes four hours in hot standby conditions.

2.8.7.2.4 Natural Circulation Cooldown References

1. Crutchfield to Maier, "Natural Circulation Cooldown, Generic Letter 81-21, R.E. Ginna Nuclear Power Plant", 5/05/81

2. "Westinghouse Owners Group Emergency Response Guidelines," Revision 1C, dated 9/30/1997.

3. Rochester Gas and Electric Corporation, Ginna Station,

   - "Natural Circulation Cooldown," ES-0.2, Revision 13.

   - "Figure Natural Circulation C/D with Shroud Fans," Figure 3.0, Rev. 0, dated 5/1/1998.

   - "Figure Natural Circulation C/D without Shroud Fans," Figure 3.1, Rev. 1, dated 2/8/2001.

4. NSAC-176L, "Safety Assessment of PWR Risks During Shutdown Operations," EPRI Outage Risk Assessment and Management (ORAM) Program (prepared by Westinghouse), Final Report August 1992. (Appendix A contains the decay heat table).

Table 2.8.7.2-1 Input Parameters for Natural Circulation Cooldown Evaluation			
Name	**Units**	**Value**	**Comment**
Power Level	MWt	1811	Includes uncertainties
Tube Plugging	Percent	10	
Inlet Temperature	°F	540.2	Highest value in PCWG case
Core Flow Rate	10E+6 lbm/hr	64.8	**Table 1-1**
Maximum Cooldown Rate	°F/hour	25	Ginna EOP ES-0.2, Natural Circulation Cooldown

Table 2.8.7.2-2 Assumptions for Natural Circulation Cooldown Evaluation	
Number	**Assumption**
1	Decay heat rates are based on ANSI/ANS-5.1-1979, including 2-sigma uncertainty (Reference 4).
2	Nominal ARV capacities are assumed for the cooldown portion of the transient.
3	The hydraulic resistance of the RCP during natural circulation cooling is based on a Forward Flow locked rotor "K" value for an RCP.

Table 2.8.7.2-3 Diablo Canyon versus Ginna Hydraulic Resistance Coefficients for Normal Flow Conditions		
Hydraulic Resistance Coefficients for Normal Flow Conditions		
	Diablo Canyon [ft/(gpm)^2]	**Ginna [ft/(gpm)^2]**
Reactor Core and Internals	129.0E-10	110.4E-10
Reactor Nozzles	36.1E-10	28.3E-10
Reactor Coolant Loop Piping	20.9E-10	30.7E-10
Steam Generator	112.0E-10	109.6E-10
Total Hydraulic Flow Resistance Coefficient (without RCPs)	298.0E-10	279.0E-10

$$\text{Flow Ratio per loop} = \left[\frac{HFC_{TOT}\text{ For Diablo Canyon}}{HFC_{TOT}\text{ For Ginna}}\right]^{1/2} = 1.03$$

Table 2.8.7.2-4 Comparison of Measured versus Predicted Loop Delta Temperature for Natural Circulation			
	Measured Loop ΔT (°F)	**Calculated Loop ΔT (°F)**	**Loop ΔT (°F) Scale Calculation to Measured**
Full-Power Level	1300 MWt	1811 MWt	
Test Conditions			
2%	25.5	33.5	26.9
3%	N/A	43.2	34.6
4%	N/A	51.5	41.3
4.2%	40.0	N/A	N/A

Table 2.8.7.2-5 Summary of Ginna Natural Circulation Flow Rates for No-Load Conditions, 1811 MWt				
	Decay Heat Percentage of 1811 MWt			
	1%	2%	3%	4%
T_{avg} (°F)	Flow Rate lbm/hr	Flow Rate lbm/hr	Flow Rate lbm/hr	Flow Rate lbm/hr
300	1.872E+06	2.359E+06	2.701E+06	2.972E+06
350	1.920E+06	2.419E+06	2.770E+06	3.048E+06
400	1.962E+06	2.472E+06	2.829E+06	3.114E+06
450	2.000E+06	2.520E+06	2.885E+06	3.175E+06
500	2.039E+06	2.570E+06	2.941E+06	3.237E+06
550	2.084E+06	2.626E+06	3.006E+06	3.308E+06
600	2.148E+06	2.706E+06	3.097E+06	3.409E+06

Table 2.8.7.2-6 Summary of Ginna Natural Circulation Flow Temperature Differentials for No-Load Conditions, 1811 MWt				
	Decay Heat Percentage of 1811 MWt			
	1%	2%	3%	4%
T_{avg}(°F)	Delta-T(°F)	Delta-T(°F)	Delta-T(°F)	Delta-T(°F)
300	32.41	51.44	67.41	81.66
350	31.04	49.28	64.57	78.22
400	29.63	47.04	61.64	74.67
450	28.06	44.54	58.36	70.70
500	26.16	41.52	54.41	65.91
550	23.66	37.56	49.22	59.63
600	19.88	31.56	41.36	50.11

2.8.7.2.5 Conclusion

For the following reasons, the Ginna EPU analyzed to 1811 MWt will not adversely impact the natural circulation cooldown capability of the plant.

Acceptable results are found for natural circulation cooling during the hot standby period for residual heat rates as high as 3% of 1811 MWt. The core outlet temperatures calculated for this case (600.8°F) are bounded by those specified for full power operation for the high Tavg cases (611.8 °F) (PCWG Cases 3 and 4, LR section 1.1,"Nuclear Steam Supply System Parameters", Table 1-1).

The calculated loop delta temperatures are scaled and compared to the UFSAR measured values. The scaled, calculated values are larger than measured, due to several conservative assumptions in the calculations.

The ARVs at the uprated conditions are adequate to achieve cooldown to the RHR entry point in a reasonable time period. RHR cut-in conditions can be achieved in ~ 14 hours at the maximum rate specified in ES-0.2, which includes four hours in hot standby conditions.

The SER commitments of reference 1 are satisfied by procedural guidance (reference 3) which follows the WOG ERGs (reference 2) and thus maintains the cool down rate, Upper Head Vessel Cooling (as a function of fan status), and continues the monitoring for subcooled margin. Therefore, existing procedural guidance for performing a natural circulation cooldown of the plant are unaffected by the EPU operating conditions.

The Ginna staff has reviewed the assessment of the effects of the proposed EPU on the systems required for natural circulation cooldown and concludes that the evaluation has adequately accounted for the effects of changes in plant conditions on the design of those systems. The Ginna staff concludes that the systems will maintain their ability to perform a natural circulation cooldown following a trip from full power to RHR cut-in conditions. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the systems used for natural circulation cooldown.

2.8.7.3 Loss of Residual Heat Removal at Mid-loop

2.8.7.3.1 Regulatory Evaluation

NRC Generic Letter (GL) 88-17, Loss of Decay Heat Removal (reference 1), identified actions to be taken to preclude loss of decay heat removal during nonpower operations. These actions included operator training and the development of procedures and hardware modifications as necessary to prevent the loss of decay heat removal during reduced reactor coolant inventory operations, to mitigate accidents before they progress to core damage, and to control radioactive material if a core damage accident should occur. Procedures and administrative controls were required that cover reduced inventory operations and ensure that all hot legs are not blocked by nozzle dams unless a vent path is provided that is large enough to prevent pressurization and loss of water from the reactor vessel. Instrumentation was required to provide continuous core exit temperature and reactor water level indication. Sufficient equipment was required to be maintained in an operable or available status so as to mitigate the loss of the residual heat removal (RHR) cooling or loss of reactor coolant system (RCS) inventory should such an event occur during mid-loop or reduced inventory conditions.

There are no specific NRC acceptance criteria within NRC regulations for operations at mid-loop or reduced inventory conditions. However, the NRC requested all holders of operating licenses to respond to the following recommended actions identified in GL 88-17:

- Provide training prior to operating in a reduced inventory condition (reactor vessel level lower than 3 ft. below the reactor vessel flange)

- Implement procedures and administrative controls that reasonably ensure that containment closure will be achieved prior to the time at which core uncovery could result from a loss of decay heat removal coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory.

- Provide at least two independent, continuous temperature indications that are representative of the core exit conditions whenever the RCS is in a mid-loop condition and the reactor vessel head is located on top of the reactor vessel.

- Provide at least two independent, continuous RCS water level indications whenever the RCS is in a reduced inventory condition.

- Implement procedures and administrative controls that generally avoid operations that deliberately or knowingly lead to perturbations to the RCS and/or systems that are necessary to maintain the RCS in a stable and controlled condition while the RCS is in a reduced inventory condition.

- Provide at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal decay heat removal systems.

- Implement procedures and administrative controls that reasonably ensure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the reactor vessel.

Specific criteria and requirements are identified in GL 88-17 and the Ginna UFSAR, Section 5.4.5.4.1.

Ginna Current Licensing Basis

The adequacy of the Ginna Nuclear Power Plant, LLC (Ginna) design and the actions taken in response to GL 88-17 are described in the UFSAR, Section 5.4.5.4. The following are specific actions taken by the Ginna Station to conform to the NRC recommendations in GL 88-17:

- Thermal-hydraulic analyses have been performed to form the basis for the required operator actions, which are implemented in procedures and administrative controls, and for the equipment required to be available for providing core cooling in the event RHR cooling is lost.

- Various configurations the plant could be in while the RCS is in the reduced inventory mode have been identified. Ginna has committed not to enter those adverse configurations with a shortened time to core uncovery..

- Multiple methods for filling the RCS have been identified.

- Two independent temperature indications and water level indicators have been provided to monitor mid-loop operation conditions.

The above analyses and procedural guidance are described in generic Westinghouse Owners Group (WOG) reports (references 2 and 3) and plant-specific design analysis (reference 4). This information has been incorporated into plant-specific operating procedures and outage management guidance.

In addition to the evaluations described in the Ginna UFSAR, the system components associated with the control or mitigation of a loss of decay heat removal capability during non-power operations 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.

2.8.7.3.2 Technical Evaluation

2.8.7.3.2.1 Introduction

The purpose of the loss of RHR at mid-loop evaluation was to determine the time to saturation, boiloff rate, minimum makeup rate to match boiloff, and time to reach 200°F following the loss of RHR at mid-loop conditions. Operation at mid-loop or reduced inventory is part of Ginna's design basis as described in UFSAR Section 5.4.5.4. The USFAR section also pertains to GL 88-17.

2.8.7.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

The input parameters used for the evaluation are given in Table 2.8.7.3-1.

Assumptions

The assumptions are summarized in Table 2.8.7.3-2.

Acceptance Criteria for Loss of RHR at Mid-Loop Evaluation

While there are no specific regulations in place, GL 88-17 describes certain requirements for operation at reduced inventory conditions. For Ginna, reduced inventory is defined as less than sixty-four inches indicated level. Note that the reference point (zero inches) for indicated level at Ginna is an actual hot leg level of four inches from the bottom of the pipe per Ginna UFSAR section 5.4.5.4.3. Apart from specific equipment requirements in GL 88-17, (e.g., independent core exit temperature indications or diverse/independent means for adding inventory), procedures must be in place for recovery from a loss of RHR event.

The acceptance criteria for the loss of RHR at mid-loop includes maintaining core cooling and protecting the reactor core until RHR can be returned to service. If RHR flow cannot be rapidly restored, the operator starts trending core exit thermocouple temperatures and initiates contingency actions while trying to return RHR to service. This evaluation provides time estimates to reach saturation conditions. The analytical bases for this evaluation can be found in WCAP-11916, Loss of RHRS Cooling While the RCS Is Partially Filled (reference 2). The results of this evaluation would be utilized by the Ginna plant staff in responding to a loss of RHR at mid-loop.

2.8.7.3.2.3 Description of Analyses and Evaluations

As a result of the EPU, the decay heat at a given time after shutdown increases, roughly in proportion to the change in the EPU. This, in turn, reduces the time to boiling and the time to core uncovery following a postulated loss of RHR cooling. Analyses have been performed to determine the time to reach 200°F, the time to reach saturation, and makeup and boil-off rates. The Ginna specific Design Analysis (reference 4) and the applicable sections of the UFSAR relating to Generic Letter 88-17 have been reviewed to identify information that may be impacted due to the proposed EPU. All of the administrative and operating procedures applicable to operations at reduced inventory were also reviewed. Based on these reviews, the Design Analysis and the UFSAR section will be revised to reflect the shorter times available to reach saturation and core uncovery and the impact this has on shutdown operations. For example, one charging pump will no longer be sufficient to match the boil-off rate at 50 hours after shutdown. As a result, two charging pumps or one SI pump should be made available to compensate for boil-off if mid-loop or reduced inventory operation is considered at this early time in the outage.

In addition, the two-hour requirement for containment closure is being investigated to determine the need for further evaluation. If the time to core uncovery is reduced resulting in a harsh containment environment within two hours, the equipment hatch may need to be kept in place during certain plant evolutions. Operating and administrative procedures are being evaluated and new time to saturation and core uncovery tables are being developed. RCS vent paths for reduced inventory operations based on reference 2 and on the EPU boiling and core uncovery times are also being evaluated to ensure nozzle dam integrity and resultant earlier core uncovery.

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. Systems and components associated with the issues identified in GL 88-17 for mid-loop operation are addressed in their respective sections of the License Renewal SER. These systems and components are subject to the programs used to manage aging effects. Any changes associated with mid-loop operation for the EPU condition do not add any new or previously unevaluated materials to the systems or components associated with mid-loop operation. Component internal and external environments remain within the parameters previously evaluated. Therefore, the proposed EPU has no impact on the previously evaluated License Renewal Program.

2.8.7.3.3 Results

The results for time to 200°F, time to saturation, boil-off rate and make-up rate as a result of the EPU are provided in Table 2.8.7.3-3. UFSAR section 5.4.5.4.5 requires that preferred flow paths and equipment be available with power to the appropriate components prior to draindown for means of adding inventory to the RCS in the event of loss of RHR cooling. Generic Letter 88-17 requires at least two means of adding inventory to the RCS must be available independent of pumps. Evaluation for times after shutdown of less than 50 hours will need to be re-evaluated. However, changes in operator action times due to EPU power levels are relatively small and have been shown to be within the capabilities of the operating crews.

No issues have been identified with the existing instrumentation that has been provided to monitor the RCS level and RHR performance during mid-loop operation (UFSAR section 5.4.5.4.3) and no additional instrumentation is required for monitoring mid-loop operation at the EPU conditions

The times to reach 200°F and saturation are shorter due to the Ginna EPU as compared to previous evaluations. The current operating procedures and guidance for responding to a loss of RHR at mid-loop will need to be revised accordingly. The Ginna procedure, AP-RHR.2, follows the guidance provided in the generic Westinghouse Abnormal Response Guideline, ARG-1, Loss of RHR While Operating at Mid-Loop Conditions (reference 3). Additional guidance may need to be added to AP-RHR.2 to identify alternate means of cooling.

The Ginna Design Analyses that are the basis for the existing guidance provided to the operators for mid-loop operation will be revised to incorporate the impact of higher EPU decay heat on existing heat-up curves and to delete information that is no longer applicable (e.g., the effect of surge line flooding per Reference 3). Also due to the EPU plant-specific analyses will be performed to re-assess required vent areas, vent path resistance, and possible plant configurations for mid-loop operation. The re-assessment of existing plant operational curve and operator actions will be performed consistent with the requirements of GL 88-17. All plant specific changes to the Ginna operational procedural guidance required due to the EPU will be implemented prior to going to mid-loop operation after the EPU has been implemented.

2.8.7.3.4 References

1. NRC Generic Letter 88-17, *Loss of Decay Heat Removal,* October 17, 1988

2. WCAP-11916, *Loss of RHRS Cooling While the RCS Is Partially Filled,* July, 1988.

3. WOG Abnormal Response Guideline and Background Information, ARG-1, *Loss of RHR While Operating at Mid-Loop Conditions*, Rev. 1, issued via WOG-96-093, June 6, 1996.

4. NSL-0000-005 Design Analysis, *Thermal Hydraulic Analyses of the Loss of RHR Cooling While the RCS is Partially Filled (Generic Letter 88-17),* Revision 3, October, 1994.

Table 2.8.7.3-1 Input Parameters for Loss of RHR at Mid-loop Evaluation			
Name	**Units**	**Values**	**Comment**
Power Level	MWt	1811	Includes uncertainties
Time After Shutdown	hours	48, 100, 168, 240, 300, 416.6, 555.5, 833.3	Consistent with previous calculations
RCS Temperatures	°F	100°F, 140°F	Consistent with previous calculations
Fuel Weight	lbm	105,996	Existing UO_2 fuel weight.
Cladding Weight	lbm	26,983	Existing cladding weight.
422V+ Fuel Weight	lbm	119,081	UO_2 weight after the Fall, 2006 outage. The higher fuel weight would result in a higher fuel heat capacity and longer times to reach saturation (which are conservative).
Cladding Weight	lbm	28,427	Zirc/ZIRLO weight after the Fall, 2006 outage.
Heatup Volume	Ft^3	634.6	Water in core, upper plenum, portion of the hot legs

Table 2.8.7.3-2 Assumptions for Loss of RHR at Mid-loop Evaluation	
Number	**Assumption**
1	The evaluation does not take credit for the increase in time to saturation due to heatup of the RV metal mass (excluding the fuel mass listed in Table 2.8.7.3-1) nor the benefit of cooling due to generation of steam.
2	The initial level in the RCS was assumed to be at mid-loop for the determination of the heatup volume. For mid-loop operations, it is expected the initial level will be slightly higher (except for certain evolutions such as RTD replacement activities where the level would be approximately six inches indicated or ten inches off of the bottom of the pipe per plant procedure O-2.3.1).
3	Two different initial RCS and steam generator temperatures are assumed for this analysis: 100°F and 140°F. These temperatures are considered typical for the RCS and conservative (high) for the secondary.
4	The decay heat is based on the American Nuclear Society (ANS) ANS-5.1-1979 decay heat standard, including + 2-sigma uncertainty and a conservative Gmax (for fission product absorption). The values are expected to be conservatively high by 10 to 20% (Reference 4).

Table 2.8.7.3-3 Loss of RHR at Mid-loop Results						
Shutdown Time (hrs)	Time to Saturation (min)		Time to 200°F (min)		Boiloff Rate (lbm/s)	Makeup Rate (gpm)
	100°F	140°F	100°F	140°F		
48	10.2	6.5	9.1	5.4	8.9	64.3
100	13.1	8.3	11.7	6.9	7.0	50.1
168	16.1	10.3	14.4	8.6	5.6	40.6
240	18.7	11.9	16.7	9.9	4.9	35.0
300	20.5	13.0	18.3	10.9	4.4	32.0
416.6	23.8	15.2	21.3	12.6	3.8	27.5
555.5	27.1	17.3	24.2	14.4	3.4	24.2
833.3	31.6	20.1	28.2	16.8	2.9	20.7

2.8.7.3.5 Conclusion

The Ginna staff has reviewed the assessment of the effects of the EPU on the loss of RHR while operating at RCS reduced inventory conditions and concludes that it has adequately identified the changes required for the EPU to ensure Ginna Station maintains its ability to operate with the RCS in a reduced inventory condition and to mitigate the consequences of a loss of RHR at midloop. The Ginna staff further concludes systems, components, and administrative controls meet the acceptance criteria of GL 88-17. It has also been determined that the operator action times, though reduced for the EPU condition, are well within the performance capabilities of the operating crews. Therefore, the Ginna staff finds the EPU acceptable with respect to operation at reduced RCS inventory conditions contingent upon revising the existing procedural guidance due to the higher EPU decay heat.

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

These source terms are developed for use in assessing the effects of the EPU on the release of liquid and gaseous effluents, during normal plant operation.

2.9.1.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by Ginna Station as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The Ginna staff's review included the parameters used to determine

- The concentration of each radionuclide in the reactor coolant,
- The fraction of fission product activity released to the reactor coolant,
- Concentrations of all radionuclides other than fission products in the reactor coolant,
- Leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and
- Potential sources of radioactive materials in effluents that are not considered in Ginna's UFSAR, related to liquid waste management systems and gaseous waste management systems.

The NRC's acceptance criteria for source terms are based on

- 10CFR20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas;
- 10CFR50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; 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 NRC SRP section 11.1.

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 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 the radioactive source term determined for the proposed EPU operating conditions relative to conformance to:

- 10CFR20, Standards for Protection Against Radiation, is discussed in UFSAR section 11.1.1, General Design Criteria. As discussed in this UFSAR section, radioactive disposal facilities are designed so that discharge of radioactive effluents to the environment and offsite shipments of radioactive material are in accordance with applicable regulations, including 10CFR20. In addition, the concentration of tritium release to the environment is also controlled within the limits of 10CFR20, as discussed in UFSAR section 9.3.4.4.8.

- 10CFR50, Appendix I, As Low As Is Reasonably Achievable (ALARA) Guidelines, is discussed in UFSAR sections 3.1.2.6.1 and 11.1.1.2. As discussed in these UFSAR sections, implementation of the 10CFR50, Appendix I requirements to ensure that radioactive discharges are ALARA has been formalized in the Ginna Technical Specifications and the Offsite Dose Calculation Manual.

- GDC-60 is described in UFSAR section 3.1.2.6.1, General Design Criterion 60 – Control of Releases of Radioactive Material to the Environment. As discussed in this UFSAR section, 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 structures, systems and components (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

The radiological source term is not within the scope of license renewal since it is an analytical product of the operational performance of plant systems and components in conjunction with regulatory limits that have been imposed on radiological releases. No changes in those applicable regulatory limits are proposed for plant operation at EPU conditions which would change license renewal boundaries. Systems and components, the performance of which affect the source term, are discussed in their respective system sections in NUREG-1786.

2.9.1.2 Technical Evaluation

2.9.1.2.1 Introduction

Tritium Sources - Tritium in the reactor coolant is a radiological concern because of its long half-life (12.3 years) and the fact that it exists primarily as tritiated water. Removal of tritium from the coolant by conventional waste processing systems/techniques is not possible, and if the concentrations in the reactor coolant become too high, it can present a hazard to plant personnel accessing the containment during power operation and/or during refueling operations. The concentrations in the plant are generally controlled by discharging part of the tritiated reactor coolant to the lake via the plant circulating cooling water. When discharged in this manner, the dose consequences to the environment and the limiting effluent concentration limits must be considered.

Normal Operation Source Terms - The normal plant operational source terms establish the long-term concentrations of principal radionuclides in the plant fluid streams as input for subsequent prediction of the expected release of radioactive materials from various effluent streams. The fluid streams of the plant are the reactor primary coolant and the secondary steam generator water and steam. The normal operations source terms serve as input to assessments of the projected normal plant effluent released to the environment.

2.9.1.2.2 Parameters and Assumptions

The key cycle design parameters considered in the calculation of the tritium generation are described in Table 2.9.1-1. The design value for tritium release from ternary fissions and integral fuel burnable absorbers to the coolant is 10% of the total generation and the expected value is 2%. These values are based on industry experience with zircaloy-clad fuel, which are well below the conservative stainless steel clad fuel based 30% release value considered in the Ginna UFSAR Section 9.3. The results of the tritium source analysis serve as input to evaluate the buildup of tritium activity in the plant water volumes and in the plant liquid and gaseous effluents. They serve as input to establish the expected long-term impact of tritium on plant refueling operations and on the expected concentrations of tritium in plant effluents.

The assumptions and input parameters that served as the basis for the determination of the primary and secondary radiation sources are summarized in Table 2.9.1-2. The results of the normal plant operation source calculations serve as input to establish the long-term, expected concentrations of principal radionuclides in plant effluents. The results of the effluent activity and concentration calculations are summarized in LR section 2.10.1.

2.9.1.2.3 Analysis

Tritium generation in the reactor coolant system was determined based on the EPU parameters and the available operating plant data, including that of NUREG/CR-2907, relative to the modeling of the generation and release of tritium to the coolant. The analyses considered the generation and release of tritium produced by ternary fission in the fuel and subsequent migration through the fuel cladding and/or defective fuel. It also considered tritium generation from activation of material in the active core region due to interaction of neutrons with soluble boron and lithium in the coolant, as well as deuterium reactions in the coolant.

The normal or expected activity concentrations in the primary and secondary sides were based on the methodology of American Nuclear Society Standard ANSI/ANS-18.1-1999. The methodology applies adjustment factors to a set of "reference value" concentrations if plant parameters deviate from a prescribed set of nominal values. Application of this standard is consistent with the methodology included in Revision 1 of the GALE code that is considered by the NRC in its review of expected plant radioactive effluents for all light water reactor (LWR) plants. Normal sources for Ginna Station are established by appropriate scaling by thermal power and other pertinent EPU parameters as outlined in the standard. The methodology also considers a "Y" factor defined as the ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system to the purification system (not including the boron recovery system).

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

The radiation source terms were not revised as part of the Ginna operating license renewal. Therefore, there is no impact on the license renewal related to the EPU radiation source terms.

2.9.1.3 Results

Tritium Sources

A summary of the results of the tritium generation and release analysis is given in Table 2.9.1-3. The time-dependent release to the reactor coolant over an operating cycle for the design and expected cases is provided in Figure 2.9.1-1.

The total releases to the reactor coolant during an EPU fuel cycle were compared to the values currently identified in the Ginna UFSAR Section 9.3.4.4.9. Both the "design and "expected' values of total tritium in the coolant associated with the EPU are lower than the original annual production value identified in the Ginna UFSAR Section 9.3 (Table 9.3-11c). The lowering of total tritium for the EPU condition is attributable to the difference in release fraction of tritium from the core. For the existing, non-EPU conditions, 30% of the tritium generated in the core is assumed to be released to the coolant. This assumption was based on analysis made from stainless steel cores. It has been updated for the EPU based on more recent operating plant data, including NUREG/CR-2907, to a conservative "design" value of 10%, and the more realistic "expected" value of 2%.

Ginna Station currently maintains the reactor coolant tritium concentration at a level that precludes a personnel hazard due to discharging part of the tritiated reactor coolant to the lake via the plant circulating cooling water. Further, as noted in the Ginna UFSAR Section 9.3.4.4.8, the tritium released in this manner is between 0.1% and 1% of the 10CFR20 limits. Thus, the Ginna station operating experience confirms that neither plant operability nor the ability to continue to comply with Ginna station applicable regulations is compromised by water management procedures that control tritium buildup. Since the tritium sources associated with the EPU remain substantially below the original design basis production values, it is concluded that the EPU will not impact the current situation.

Normal Operation Source Terms

The calculated primary and secondary radiation sources are summarized in Table 2.9.1-4. The results of the normal plant operation source calculations served as input to establish the long-term concentrations of principal radionuclides in the fluid streams of the plant for subsequent application in estimating the expected release of radioactive materials from various effluent streams.

Summary

Ginna station continues to comply with 10CFR20 requirements for radioactivity in liquid and gaseous effluents and for maintaining personnel exposures. Ginna also continues to comply with10CFR50, Appendix I objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. And finally, Ginna Station meets current licensing basis requirements with respect to GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Table 2.9.1-1 **Parameters Used in the Calculation of Tritium Production in the Reactor Coolant – Assumptions**	
Core thermal power, MW(th)	1811
Reactor coolant water volume, ft^3	5441
Core water volume, ft^3	354
Core water mass, grams	7.32×10^6
Plant full power operating time (equilibrium cycle)	82.2 weeks (18.9 months)
Boron concentration (Peak hot full power equilibrium Xenon) – Equilibrium cycle, ppm	1693
Burnable poison boron content (total—all rods), kg	3.0
Cycle average reactor coolant lithium concentration 2.7 ppm.	2.7
Fraction of tritium in core (ternary fission + burnable boron) diffusing through clad ▪ Design value ▪ Expected value	 0.10 0.02
Ternary fission yield, atoms/fission	8×10^{-5}

Table 2.9.1- 2 ANSI/ANS 18.1 – 1999 Normal Source Input Parameters					
Parameter	**Symbol**	**Value**	**Units**	**Nominal Value**	
Core Thermal Power	P	1.811E+03	MWt	3.4E+03	MWt
Weight of Water in Reactor Coolant System	WP	4.07E+04	gal	2.5E+05	kg
Reactor Coolant Letdown Flow Rate (Purification)	FD	4.00E+01	gpm	4.7E+00	kg/sec
Reactor Coolant Letdown Flow Rate (Yearly Average for Boron Control)	FB	1.66E-01	gpm	6.3E-02	kg/sec
Flow Through the Purification System Cation Demineralizer	FA	4.00E+00	gpm	4.7E-01	kg/sec
Steam Flow Rate	FS	7.41E+06	lb/hr	1.9E+03	kg/sec
Weight of Secondary Side Water in all Steam Generators	WS	1.67E+05	lb	2.0E+05	kg
Steam Generator Blowdown Flow Rate (Total)	FBD	1.00E+02	gpm	9.5E+00	kg/sec
Density of RCS Water	Drcs	4.48E+01	lb/ft^3		
VCT Liquid Volume	VOL-L	1.00E+02	ft^3		
VCT Vapor Space Volume	VOL-V	1.00E+02	ft^3		
VCT Purge Rate	PR	0.00E+00	scfm		
Density of VCT Water	Dvct	6.16E+01	lb/ft^3		
VCT Temperature	TEMP	1.27E+02	deg F		
VCT Vapor Pressure	PRESS	1.50E+01	psig		

Table 2.9.1-3
Calculation of Tritium Production in the Reactor Coolant

Calculations	Equilibrium Cycle Design Value (Ci/cycle)	Equilibrium Cycle Expected Value (Ci/cycle)
Tritium from Core		
1. Ternary Fission	10,400	10,400
2. ^{10}B (n, 2α) T (In Poison Rods)	220	220
3. ^{10}B (n, α) ^{7}Li (n, nα) T (In Poison Rods)	1290	1290
4. Release Fraction	0.10	0.02
5. Total Released to Coolant	1,191	238
Tritium from Coolant		
1. ^{10}B (n, 2α) T	445	445
2 Li (n, nα) T	16	16
3. ^{6}Li (n, α)	106	106
4. D2 (n,γ)	2	2
5. Release Fraction	1.0	1.0
6. Total Released to Coolant	569	569
Total Tritium in Coolant	1760	806

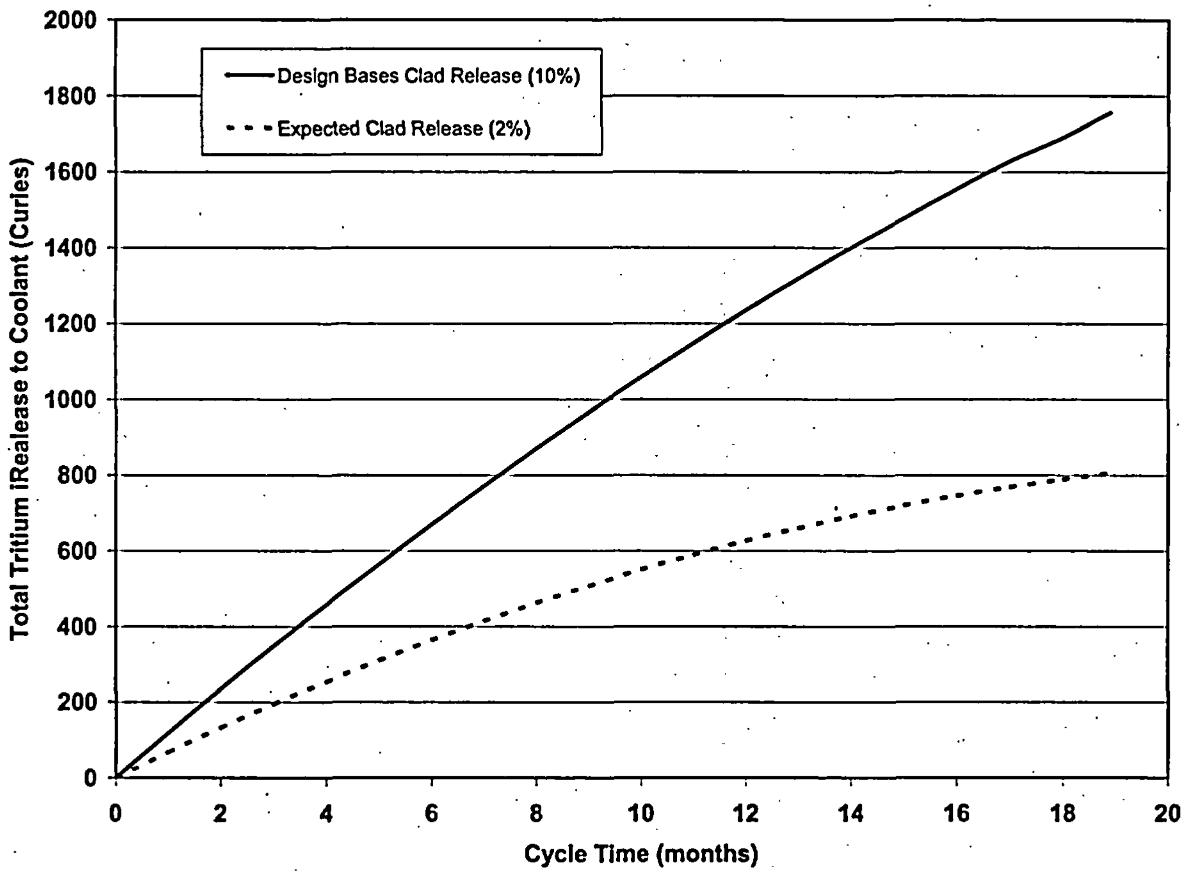
Table 2.9.1-4
R. E. Ginna Normal Plant Operation Sources (µCi/g)
Based on ANSI/ANS-18.1 – 1999

Nuclide	RCS	Secondary Side		Nuclide	RCS	Secondary Side	
		Water	Steam			Water	Steam
Class 1				**Class 6**			
Kr-85m	1.5E-02	nil	6.5E-09	Na-24	5.0E-02	2.6E-06	1.3E-08
Kr-85	1.3E+00	nil	5.6E-07	Cr-51	3.0E-03	1.9E-07	9.2E-10
Kr-87	1.8E-02	nil	2.2E-08	Mn-54	1.5E-03	9.5E-08	4.8E-10
Kr-88	1.8E-02	nil	7.7E-09	Fe-55	1.2E-03	7.2E-08	3.6E-10
Xe-131m	7.7E-01	nil	3.2E-07	Fe-59	2.9E-04	1.8E-08	8.9E-11
Xe-133m	6.3E-02	nil	2.8E-08	Co-58	4.5E-03	2.8E-07	1.4E-09
Xe-133	2.7E-02	nil	1.2E-08	Co-60	5.1E-04	3.2E-08	1.6E-10
Xe-135m	1.5E-01	nil	6.4E-08	Zn-65	4.9E-04	3.1E-08	1.5E-10
Xe-135	6.2E-02	nil	2.6E-08	Sr-89	1.4E-04	8.3E-09	4.2E-11
Xe-137	4.1E-02	nil	1.7E-07	Sr-90	1.2E-05	7.2E-10	3.6E-12
Xe-138	7.2E-0	nil	3.1E-08	Sr-91	1.0E-03	5.2E-08	2.6E-10
				Y-91m	5.4E-04	8.9E-09	4.4E-11
Class 2				Y-91	5.0E-06	3.1E-10	1.6E-12
Br-84	1.9E-02	2.2E-07	2.2E-09	Y-93	4.5E-03	2.2E-08	1.1E-09
I-131	2.0E-03	1.3E-07	1.3E-09	Zr-95	3.8E-04	2.3E-08	1.2E-10
I-132	7.0E-02	2.2E-06	2.2E-08	Nb-95	2.7E-04	1.6E-08	8.4E-11
I-133	2.8E-02	1.6E-06	1.6E-08	Mo-99	6.4E-03	3.8E-07	1.8E-09
I-134	1.2E-01	2.0E-06	2.0E-08	Tc-99m	5.2E-03	2.2E-07	1.2E-09
I-135	6.2E-02	2.8E-06	2.8E-08	Ru-103	7.3E-03	4.5E-07	2.3E-09
				Ru-106	8.7E-02	5.4E-06	2.6E-08
Class 3				Ag-110m	1.3E-03	7.7E-08	3.9E-10
Rb-88	2.3E-01	1.6E-06	7.8E-09	Te-129m	1.8E-04	1.1E-08	5.7E-11
Cs-134	3.6E-05	2.2E-09	1.2E-11	Te-129	2.8E-02	5.9E-07	2.9E-09
Cs-136	8.5E-04	5.3E-08	2.7E-10	Te-131m	1.5E-03	8.7E-08	4.4E-10

Table 2.9.1-4 (con't) R. E. Ginna Normal Plant Operation Sources (µCi/g) Based on ANSI/ANS-18.1 – 1999							
Cs-137	5.1E-05	3.3E-09	1.6E-11	Te-131	9.2E-03	8.5E-08	4.4E-10
				Te-132	1.7E-03	1.0E-07	5.0E-10
Class 4				Ba-140	1.3E-02	7.7E-07	3.8E-09
N-16	4.0E+01	2.6E-06	2.6E-07	La-140	2.5E-02	1.5E-06	7.3E-09
				Ce-141	1.5E-04	8.9E-09	4.5E-11
Class 5				Ce-143	2.8E-03	1.6E-07	8.2E-10
H-3	1.0E+00	1.0E-03	1.0E-03	Ce-144	3.9E-03	2.3E-07	1.2E-09
				W-187	2.6E-03	1.4E-07	7.3E-10
				Np-239	2.2E-03	1.3E-07	6.5E-10

Figure 2.9.1-1

Maximum Tritium Activity Released to Primary Coolant



2.9.1.4 Conclusion

The Ginna staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The Ginna staff further concludes that the proposed radioactive source term meets the requirements of 10CFR20, 10CFR50, Appendix I, and GDC-60. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

2.9.2.1 Regulatory Evaluation

Ginna performed Design Basis Accident (DBA) radiological consequences analyses using the guidance in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. Except as noted, the assumptions are the same as those provided in Ginna's Control Room Emergency Air Treatment/AST amendment SER, dated February 25, 2005 (reference 1), as supplemented by letter dated May 18, 2005 (reference 2). The radiological consequences analyses include the Main Steam Line Break (MSLB), Locked-Rotor Accident (LRA), Rod Ejection Accident (REA), Steam Generator Tube Rupture (SGTR), Loss of Coolant Accident (LOCA), Fuel Handling Accident (FHA) and Tornado Missile Accident (TMA). Ginna's analysis for each accident considered

- The sequence of events; and
- Models, assumptions, and values of parameter inputs used for the calculation of the total effective dose equivalent (TEDE).

The NRC's acceptance criteria for radiological consequences analyses using an alternate source term are based on:

- 10CFR50.67, insofar as it sets standards for radiological consequences of a postulated accident, and
- 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 TEDE, as defined in 10CFR50.2, for the duration of the accident.

Specific review criteria are contained in SRP section 15.0.1.

2.9.2.2 Technical Evaluation

The analyses performed for the EPU reflect the methodology of the licensing submittal and subsequent SER (References 1 and 2), and have all been updated, using input assumptions consistent with the proposed EPU. For each accident, the TEDE is determined at the Exclusion Area Boundary (EAB) for the limiting 2-hour period, at the Low Population Zone (LPZ) outer boundary for the duration of the accident, and in the Control Room for 30 days. These results are summarized in Table 2.9.2-1 along with the dose acceptance criteria.

2.9.2.2.1 Common Input Parameters and Assumptions

The assumptions and input described in this section are common to all analyses. They are consistent with those presented in reference 1, except as noted. Accident specific input and assumptions are described in detail in the sections that follow.

> **Dose Conversion Factors (DCFs)**, for the TEDE offsite and control room dose calculations, are derived from FGR 11 and FGR 12. The DCFs are unchanged from those used in references 1 and 2.
>
> **Atmospheric Dispersion Factors (X/Q) and Breathing Rates**, for the EAB, LPZ and control room, are presented in Table 2.9.2-2. These are unchanged from those previously submitted in references 1 and 2.
>
> **Control Room** radiological analysis parameters and assumptions are summarized in Table 2.9.2-3. The control room ventilation system is described in LR section 2.7.3.1. Iodine removal efficiencies have been revised from those of references 1 and 2.

Iodine Spike

A concurrent and a pre-accident iodine spike are modeled.

> **Pre-Accident Spike:** A reactor transient has occurred prior to the postulated accident and has raised the primary coolant iodine concentration to the maximum value (60 μCi/gm EDE I-131) permitted by the technical specifications (TS).
>
> **Concurrent Spike:** The primary system transient associated with the accident, causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per hour) increases to a value 500 times (335 for SGTR) greater than the release rate corresponding to the iodine concentration at the equilibrium value (1.0 μCi/gm EDE I-131) specified in TS. A concurrent iodine spike is not considered if fuel damage is postulated. The assumed iodine spike duration is 8 hours.

The EPU assumes that the initial primary coolant iodine concentrations are based on effective dose equivalent (EDE) rather than thyroid dose equivalent I-131, as used in references 1 and 2.

The accidents and associated iodine spike are presented in the table that follows.

Accident	Pre-Accident Spike	Concurrent Spike	Concurrent Spike Multiplier
MSLB	■	■	500
LR	■	n/a	n/a
RE	■	n/a	n/a
SGTR	■	■	335

2.9.2.2.2 MSLB Radiological Consequences

Description of Event

The MSLB accident considered is the complete severance of the 36-inch main steam header outside containment, but inside the turbine building. The header is the largest diameter steam line outside of the containment. The radiological consequences of a break inside containment are bounded by the consequences of a break outside containment. Thus, only the MSLB outside of containment is considered with regard to the radiological consequences. The single failure is assumed to be a failure of the main steam isolation valve on the faulted steam generator (SG). The faulted steam generator will rapidly depressurize and release the initial contents of the steam generator to the environment. A reactor trip occurs, main steam isolation occurs, safety injection actuates, and a loss of offsite power (LOOP) is assumed concurrent with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by release of steam directly to the environment.

No fuel damage is postulated to occur as the result of an MSLB. Two iodine spiking cases are considered. The first assumes that a pre-accident iodine spike occurred just before the event and the Reactor Coolant System (RCS) iodine inventory is at the maximum value permitted by TS. The second case assumes the event initiates a concurrent iodine spike.

At approximately 10 minutes, the faulted steam generator is isolated by operator action. The intact steam generator is then used for cool down, where steam is released to the atmosphere through the intact steam generator atmospheric relief valve (ARV). The faulted steam generator is assumed to boil dry within 10 minutes, releasing the entire liquid inventory and entrained radio-nuclides through the faulted steam line to the environment. Primary-to-secondary leakage liquid and radionuclide inventory is also released to the environment.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-4. The radioactivity transport model is shown in Figure 2.9.2-1.

A primary-to-secondary leak rate of one gpm to each SG is assumed for the duration of the event (eight hours). The faulted SG is assumed to be dry at ten minutes, and remain dry for the duration of the event. The intact SG is isolated from the break within the first minute and auxiliary feedwater maintains SG level for the duration of the event. Primary-to-secondary leakage, into the faulted SG, is released directly to the environment, with no credit for retention. Primary-to-secondary leakage and iodine activity into the intact SG mixes with the bulk SG water. Iodine in the bulk SG water is released to the environment in proportion to the steaming rate and the inverse partition coefficient. The elemental iodine release assumes an iodine partition of 100. The organic iodide release assumes a partition of 1.0. Noble gas activity, leaked to the SGs, is assumed to be directly released to the environment without mixing.

Comparison to References 1 and 2

The analysis of the MSLB radiological consequences, for the EPU, is consistent with the analytical methods and assumptions presented in reference 1, with changes made to reflect the increased power. Specific changes include:

- use of EDE iodine concentrations in-place of thyroid DE concentrations
- revised primary and secondary coolant initial water mass and nuclide activity
- revised steam release from both the faulted and intact SGs

Results

The results of the MSLB dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.3 LRA Radiological Consequences

Description of Event

The accident considered is the instantaneous seizure of a reactor coolant pump rotor (i.e., a locked rotor accident), which causes a rapid reduction in the flow through the affected RCS loop. The flow imbalance creates localized temperature and pressure changes in the core. These differences are predicted to lead to localized boiling and fuel damage. The main condenser is unavailable, and the plant is cooled down by releases of steam directly to the environment.

Analysis Assumptions and Parameters

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-5. Fuel rods producing 50% of the core power are assumed to experience departure from nucleate boiling (DNB), and are therefore assumed to release their gap activity into the RCS. The radio-nuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage. The leakage is assumed to mix with the bulk water of the steam generators, and the radio-nuclides in the bulk water are released at a rate that is a function of the steaming rate for the steam generators, and the partition coefficient.

The tubes in the steam generators remain covered by the bulk water. Primary-to-secondary leakage into the SGs mixes with the bulk SG water. Iodine in the SG water is released to the environment as a function of the steaming rate and the partition coefficient. The elemental iodine release assumes an iodine partition of 100. The organic iodide release assumes a partition of 1.0. Noble gas activity, leaked into the SGs, is assumed to be directly released to the environment without mixing. The steam releases, from the steam generators, continue until the RHR system can be used to complete the cooldown at approximately eight hours.

Comparison to References 1 and 2

The analysis of the LRA radiological consequences is consistent with the analytical methods and assumptions presented in References 1 and 2, with changes made to reflect the increased power. Specific changes include:

- revised core nuclide inventory, consistent with the EPU
- use of EDE iodine concentrations in place of thyroid DE concentrations
- revised primary and secondary coolant initial water mass and nuclide activity
- revised steam release from the SGs

Results

The results of the LRA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.4 REA Radiological Consequences

Description of Event

The accident considered is the mechanical failure of a control rod drive mechanism pressure housing, which results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected due to the reactivity spike. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam directly to the environment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through the steam generators.

The actual dose from a REA is a composite of the two pathways. However, the dose from each pathway is conservatively modeled independently of the other.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-6. The radioactivity transport model is shown in Figure 2.9.2-2.

Fifteen percent (15%) of the fuel rods are assumed to fail, releasing the radionuclide inventory in the fuel rod gap. It was further assumed that 10% of the core inventory of radioiodine and noble gas is in the fuel rod gap. A radial peaking factor of 1.75 was applied. In addition, localized heating is assumed to cause 0.375% of the failed fuel rods to melt, releasing 100% of the noble gases and 25% of the radioiodine contained in the melted fuel to the containment. For the secondary release case, 100% of the noble gases and 50% of the radioiodine contained in the melted fuel are released to the secondary.

The containment leakage case assumes that radionuclides released from the fuel are instantaneously and homogeneously mixed in the containment free volume. In addition, the containment leaks at the TS value of 0.2% volume per day for the first 24 hours and 0.1 % volume per day for days 2 through 30. Credit is taken for removal of iodine, in particulate form, by HEPA filters in the containment recirculation fan cooler system (CRFCS). The CRFCS is a safety-related system and its operational requirements are specified in the Ginna TSs.

No credit is taken for containment spray operation as a radionuclide removal mechanism. However, natural deposition processes are assumed to result in the removal of aerosols at a rate of 0.023 hr^{-1} based on the methodology of NUREG/CR-6189, A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments.

Comparison to References 1 and 2

The analysis of the REA radiological consequences is consistent with the analytical methods and assumptions presented in references 1 and 2, with changes made to reflect the increased power. Specific changes include:

- revised core nuclide inventory, consistent with the EPU
- use of EDE iodine concentrations in place of thyroid DE concentrations
- revised primary and secondary coolant initial water mass and nuclide inventory
- revised steam release from the SGs due to higher decay heat
- revised failed and melted fuel fractions

Results

The results of the REA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.5 SGTR Radiological Consequences

Description of Event

The accident considered is the complete severance of a single tube in one of the steam generators resulting in the transfer of RCS water to the ruptured steam generator. The primary-to-secondary break flow through the ruptured tube following a SGTR results in radioactive contamination of the secondary system. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam directly to the environment. The limiting single failure is a single, failed open ARV on the affected steam generator, providing a continuous release path to the environment. The failed ARV is assumed to be closed by manual operator action within 25 minutes after failing open.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-7. The radioactivity transport model is shown in Figure 2.9.2-3.

A portion of the break flow flashes to vapor, rises through the bulk water, enters the steam space, and is immediately released to the environment with no mitigation or holdup. The flashing fraction ranges from 0 to 0.17 averaging approximately 0.04. The portion of the break flow that does not flash is assumed to mix with the bulk water of the steam generator. In addition to the break flow, primary-to-secondary leakage is assumed at the maximum value permitted by TSs. Primary-to-secondary leakage is assumed to be 150 gpd into the bulk water of each SG.

The radionuclides in the bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. The tubes in both the affected and unaffected steam generators remain covered by the bulk water. Primary-to-secondary leakage into the unaffected SG mixes with the bulk SG water. Iodine in the SG water is released to the environment as a function of the steaming rate and the partition coefficient. The elemental iodine release assumes an iodine partition of 100. The organic iodide release assumes a partition of 1.0. Noble gas activity, leaked into the SGs, is assumed to be directly released to the environment without mixing. The steam releases from the steam generators continue until the RHR system can be used to complete the cooldown at approximately 8 hours.

Comparison to References 1 and 2

The analysis of the SGTR radiological consequences is consistent with the analytical methods and assumptions represented in references 1 and 2, with changes made to reflect the increased power. Specific changes include:

- use of EDE iodine concentrations in-place of thyroid DE concentrations
- revised primary and secondary coolant initial water mass and nuclide inventory
- revised steam release from the SGs
- revised rupture flow and flashing fractions

Results

The results of the SGTR dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.6 LOCA Radiological Consequences

Description of Event

The LOCA accident considered is double-ended rupture of a reactor coolant system pipe. Activity from the core is released to the containment and then to the environment by containment leakage or leakage from the Emergency Core Cooling System (ECCS) as it re-circulates sump solution outside the containment.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-8.

Fission products released to the containment atmosphere, following the postulated LOCA, are mitigated by three processes:

(1) Containment Spray System (CSS)
(2) Containment Recirculation Fan Cooler System (CRFCS)
(3) Radioactive decay

The CSS, in conjunction with the CRFCS, is designed to provide containment cooling and fission product removal following the postulated LOCA. 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 occurs. After 52 minutes into the accident, spray pump operation is terminated. No containment spray recirculation phase is assumed.

The CRFCS is designed to remove heat and 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 is directed through the charcoal adsorbers. However, the adsorbers are not credited for evaluating potential radiological consequences. Two CRFCS units re-circulate 12,000 cfm within the lower (unsprayed) containment volume, and 48,000 cfm is assumed to mix the sprayed and unsprayed volumes.

Sixty minutes after the start of the event, the Residual Heat Removal (RHR) system starts to draw water from the containment sump. RHR operation circulates contaminated sump water outside of the containment, where system leakage is assumed to provide a path for the release of radionuclides to the environment. The assumed leakage rate of four gallons per hour is two times the program value, consistent with the guidance provided in Regulatory Guide 1.183. This leakage assumption also includes any potential back-leakage into the RWST during the recirculation phase.

It was conservatively assumed that all of the radioiodine and non-iodine particulate released from the fuel is instantaneously moved to the containment sump water, and that noble gases are assumed to remain in the containment atmosphere. It was further assumed that chemical form of radioiodine in the sump water, at the time of recirculation is 97% elemental iodine and 3% organic. This assumption was made even though the sump pH is greater than 7.0. Since the containment sump pH is maintained greater than 7.0, the radioiodine in the sump solution is in the nonvolatile iodide or iodate form. Regardless, the more conservative iodine chemical form assumption, of Regulatory Guide 1.183, was used. The total iodine in the leaked fluid is assumed to become airborne and leak to the environment via the back-draft damper's louver on the North wall of the auxiliary building for 30 days after the start of recirculation. This release point has the most conservative atmospheric dispersion factor for the control room.

The amount of iodine that is assumed to become airborne ranges from 5% at the start of the containment sump water recirculation, gradually decreasing to 2%, as a function of time, for the duration of 30-day accident period. This estimate is based on a constant enthalpy equation and actual Ginna-specific sump water temperature and pressure. The calculated flashing fractions and the assumed airborne fractions are presented in Figure 2.9.2-4. The airborne fractions conservatively bound the calculated flashing fractions.

Comparison to References 1 and 2

The analysis of the LOCA radiological consequences is consistent with the analytical methods and assumptions presented in references 1 and 2, with changes made to reflect the increased power. Specific changes include:

- core inventory consistent with EPU
- CRFCS particulate removal continues for duration of accident
- sump water temperatures and corresponding flashing fractions consistent with EPU

Results

The results of the LOCA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.7 FHA Radiological Consequences

Description of Event

The FHA assumes the dropping of a spent fuel assembly during refueling, and it is postulated to occur either inside the containment or in the fuel storage (auxiliary) building. The dropped assembly may strike the fuel storage rack, the reactor vessel flange, or another fuel assembly.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-9. The total number of damaged fuel rods is assumed equivalent to one (1) assembly. The damaged assembly is assumed to have the highest inventory of radionuclides of all the assemblies in the core. The radionuclide inventory, in the gaps of the damaged fuel rods, is assumed to be instantaneously released. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool depending on their physical and chemical form. A decay time of 100 hours prior to moving irradiated fuel was assumed for both the FHA in the containment and in the spent fuel pool.

No decontamination is assumed for noble gases. An effective water pool decontamination factor of 200 is assumed for radioiodine, and infinite DF (100% retention) is assumed for all aerosol and particulate radionuclides. The FHA in the containment assumed that 100% of the radionuclides, released from the reactor cavity, are released to the environment in two hours. No credit is taken for filtration, holdup, or dilution. Iodine removal by the ABVS charcoal filters (90% for elemental iodine and 70% for organic iodine) is assumed for an FHA in the spent fuel pool. The Ginna TSs require operation of the auxiliary building ventilation system during irradiated fuel movement within the auxiliary building when one or more fuel assemblies in the auxiliary building has decayed less than 60 days since being irradiated. The charcoal filters are tested in accordance with the Ginna TS Section 5.5.10, "Ventilation Filter Testing Program."

Comparison to References 1 and 2

The analysis of the FHA radiological consequences is consistent with the analytical methods and assumptions presented in references 1 and 2, with changes made to reflect the increased power. Specific changes include:

- core inventory consistent with EPU

Results

The results of the FHA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.2.8 TMA Radiological Consequences

Description of Event

The TMA assumes damage to stored spent fuel from the impact of a tornado missile. The assumed hypothetical tornado missile, a 1490 pound wooden pole, 35 feet in length and 13.5 inches in diameter, propelled by the wind, penetrates the auxiliary building roof and impacts nine fuel assemblies in the spent fuel storage pool.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-10. The missile is assumed to impact nine fuel assemblies in the spent fuel storage pool (five fuel assemblies decayed for 100 hours and four fuel assemblies decayed for 60 days). Neither control room isolation nor re-circulating filtration is assumed.

Comparison to References 1 and 2

The analysis of the TMA radiological consequences is consistent with the analytical methods and assumptions presented in References 1 and 2, with changes made to reflect the increased power. Specific changes include:

- core inventory consistent with EPU

Results

The results of the TMA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Table 2.9.2-1
Summary of EPU Doses and Acceptance Criteria, rem TEDE

Accident	EAB Max. 2-hr	LPZ	Offsite Limit	Control Room (5 rem limit)
MSLB w/con. spike	0.45	0.12	2.5	0.58
MSLB w/ PA spike	0.07	0.03	25	0.17
LRA	1.16	0.35	2.5	1.87
REA (containment + secondary)	1.34	0.41	6.3	1.83
SGTR w/con. spike	0.17	0.03	2.5	0.22
SGTR w/ PA spike	0.44	0.06	25	0.94
LOCA (containment + ECCS)	3.1	1.2	25	4.6
FHA in containment	0.61	0.07	6.3	1.4
FHA in aux. bldg	0.17	0.02	6.3	0.12
TMA	0.03	0.01	6.3	0.63

Table 2.9.2-2
Atmospheric Dispersion Factors (sec/m^3)
and
Breathing Rates (m^3/sec)
for the
EAB, LPZ and Control Room

OFFSITE X/Q

Boundary	2 hr[1/]	0 - 8 hr	8 – 24 hr	24 – 96 hr	96 – 720 hr
EAB	2.17E-04[2/]	-	-	-	-
LPZ		2.51E-05	1.78E-05	8.50E-06	2.93E-06

[1/] Any two hour period.
[2/] 0 to 1 min tornado value is 1.87E-6

OFFSITE BREATHING RATES

Boundary	2 hr	0 - 8 hr	8 – 24 hr	24 – 96 hr	96 – 720 hr
EAB	3.47e-04				
LPZ		3.47E-04	1.75E-04	2.32E-04	

CONTROL ROOM X/Q

Release Point	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
Main Steam Header	2.59E-03	1.88E-03	8.28E-04	5.90E-04	4.77E-04
Intact SG ARV	3.72E-03	2.51E-03	1.15E-03	8.35E-04	6.88E-04
Containment shell	1.77E-03	1.25E-03	4.80E-04	4.24E-04	3.66E-04
Auxiliary Building	4.69E-03	3.97E-03	1.40E-03	1.32E-03	1.11E-03
Containment Equipment Hatch Roll-up Door	5.58E-03	4.66E-03	1.65E-03	1.58E-03	1.32E-03
Plant Vent	1.99E-03	1.46E-03	6.35E-04	5.01E-04	4.47E-04
Spent Fuel Pool	5.14E-05[1/] 1.44E-03[2/]	1.22E-03	4.54E-04	4.17E-04	3.38E-04

[1/] 0 to 1 minute (tornado conditions)
[2/] 1 minute to 2 hours (normal accident meteorology)

The CR breathing rate is 3.47E-04 m^3/sec.

Table 2.9.2-3
Control Room Parameters

Condition	In leakage		Recirculation	
	cfm	iodine removal efficiency, % (elemental / organic / particulate)	cfm	iodine removal efficiency, % (elemental / organic / particulate)
normal operation	2200	0/0/0	0	0/0/0
isolation	300	0/0/0	0	0/0/0
emergency operating conditions	300	0/0/0	5400	94/94/99

Control Room Isolation and CREATS Operation Time

Accident	Time, sec	CREATS Operating
MSLB, LRA, REA, LOCA, FHA	60	70
SGTR	360	370
TMA	no isolation	n/a

Table 2.9.2-4
MSLB Dose Analysis Assumptions

Parameter	Value
Reactor power, MWt (including fuel management factor)	1811
Initial reactor coolant activity, pre-accident iodine spike iodine, µCi/gm of EDE I-131 noble gas fuel defect level, %	 60 1
Initial reactor coolant activity, concurrent iodine spike iodine, µCi/gm of EDE. I-131 noble gas fuel defect level, %	 1.0 1.0
Concurrent iodine spike factor	500
Duration of concurrent iodine spike, hours	8
Initial secondary coolant iodine activity, µCi/gm of EDE I-131	0.1
Primary-to-secondary leakage leak rate per SG (during accident), gpm duration of leakage, hours	1 8
Mass of primary coolant, gm	1.28E+08
Initial mass of secondary coolant, gm affected SG unaffected SG	 5.65E+07 5.65E+07
Steam Releases Affected SG 0 - 610 sec 610 sec - 8 hours (cold conditions) Unaffected SG 0 - 2 hr 2 - 8 hr	 5.12E+07 gm 1 gal/min 8.23E+05 gm/min 5.65E+05
Steam generator iodine partition coefficients (mass-based) affected SG elemental and methyl unaffected SG elemental methyl (organic) Noble gas, all SGs	 1 100 1 1
Iodine fractions in the reactor coolant and SG water elemental iodine organic iodide	 0.97 0.03

Table 2.9.2-5
LRA Dose Analysis Assumptions

Parameter	Value
Reactor power, MWt (including fuel management factor)	1811
Failed fuel, rods producing % of core power	50
Initial reactor coolant activity, pre-accident iodine spike iodine, µCi/gm of EDE I-131 noble gas fuel defect level, %	 60 1
Initial secondary coolant iodine activity, µCi/gm of EDE I-131	0.1
Primary-to-secondary leakage (post accident) to SGs leak rate (cold conditions) per SG, gpd duration of leakage, hours	 500 8
Mass of primary coolant, gm	1.28E+08
Initial mass of secondary coolant in 2 SGs, gm	7.72E+07
Steam Releases (2 SGs), lb 0 - 2 hr 2 - 8 hr	 210,300 484,500
Steam generator iodine partition coefficients (mass-based) elemental methyl (organic) Noble gas	 100 1 1
Iodine fractions in the reactor coolant and SG water elemental iodine methyl (organic) iodide	 0.97 0.03
Fuel rod gap fractions: I-131 other halogens Kr-85 other noble gases	 0.08 0.05 0.1 0.05

Table 2.9.2-6
REA Dose Analysis Assumptions

Parameter	Value
Reactor power, MWt (including fuel management factor)	1811
Failed Fuel, % of core	15
Melted fuel, % of core	0.375
Power peaking factor	1.75
Initial Primary Coolant Activity iodine noble gas and alkali metals	 60 µCi/gm of EDE I-131 1% defects
Containment net free volume, ft^3	1.0E+6
Containment leak rate, %/day 0-24 hours > 24 hours	 0.2 0.1
Containment fan cooler flow and operation number of operating units flow rate per unit, cfm total filtered flow rate, cfm carbon (1 unit) HEPA (2 units) initiation delay carbon HEPA (automatic actuation) termination of iodine removal, hours	 2 30,000 n/a 60,000 n/a 53 sec n/a
Containment fan cooler iodine removal efficiency, % elemental methyl particulate	 n/a n/a 95
Fuel rod gap fractions iodine and noble gas (containment and secondary) Rb, Cs (containment only)	 0.1 0.12

Table 2.9.2-6 (continued)
REA Dose Analysis Assumptions

Melt release fractions	
Containment leakage	
iodine	0.25
noble gas	1.0
non-iodine particulate	Per RG 1.183, Table 2
Secondary system release	
iodine	0.5
noble gas	1

Table 2.9.2-7
SGTR Dose Analysis Assumptions

Parameter	Value
Reactor power, MWt (including fuel management factor)	1811
Initial reactor coolant activity, pre-accident iodine spike iodine, μCi/gm of EDE I-131 noble gas fuel defect level, %	 60 1.0
Initial reactor coolant activity, accident initiated iodine spike iodine, μCi/gm of EDE I-131 noble gas fuel defect level, %	 1.0 1.0
Concurrent iodine spike factor	335
Duration of concurrent iodine spike, hours	8
Initial secondary coolant iodine activity, μCi/gm of EDE I-131	0.1
Primary-to-secondary leakage to unaffected SG leak rate (cold conditions) duration of leakage, hours	 150 gal/day 8
Mass of primary coolant, gm	1.28E+08
Initial mass of secondary coolant, gm affected SG unaffected SG	 3.86E+07 3.86E+07
Steam generator elemental iodine partition coefficients (mass-based) Activity release from affected SG via boiling of bulk water via flashing break flow Activity release from unaffected SG	 100 1.0 100
Steam generator partition coefficient for organic (methyl) iodide and noble gas release	1
Iodine species assumed in the reactor coolant and SG water elemental iodine organic (methyl) iodide	 0.97 0.03

Table 2.9.2-7 (continued)
SGTR Steam Release and Break Flow

Affected SG

Time	0 - 174sec		174 - 5234sec		to 2 hr	to 8 hr
	0 - 100	100 -174	to 2596	to 5234		
Steam release, lb	189100		76000		0	28300
Break flow, lb	0	4200	86839	62961	0	0
Flashed Break, lb	0	746	5444	0	0	0

Unaffected SG

Time	0 - 174 sec	to 5234 sec	to 2 hr	to 8 hr	to 40 hr
Steam release, lb	188,400	104,700	88,800	513,100	1,760,100

The analysis conservatively treats steam released to the condenser the same as a direct release to the atmosphere, i.e., elemental iodine partition is 100.

Table 2.9.2-8
LOCA Dose Analysis Assumptions

Parameter	Value
Reactor power, Mwt (including fuel management factor)	1811
Containment net free volume, ft^3	106
Containment sprayed fraction	0.78
Containment leak rate, %/day	
0-24 hours	0.2
> 24 hours	0.1
Containment fan cooler flow and operation	
number of operating units	2
flow rate per unit, cfm	30,000
total filtered flow rate, cfm	
carbon (1 unit)	n/a
HEPA (2 units)	60,000
initiation delay, sec	50
termination of iodine removal, hours (modified from CREATS submittal)	n/a
Flow recirculation in lower compartment, cfm	12,000 (20% of total flow)
Mixing flow, cfm	48,000
Containment fan cooler iodine removal efficiency, %	
elemental	n/a
organic (methyl)	n/a
particulate	95
Containment injection spray	
flow rate, gpm	1300
initiation delay, sec	80
termination (end of spray injection), min	52
Iodine and particulate removal by spray, hr-1	
elemental	20
particulate	3.5
Containment sump volume, gal	264700

Table 2.9.2-8 (Cont)
LOCA Dose Analysis Assumptions

ECCS Leakage	
Continuous leakage rate, gal/hour	4
Start time, hr	1
Termination time, hr	720
Airborne fraction	see Figure 2.9.2-4

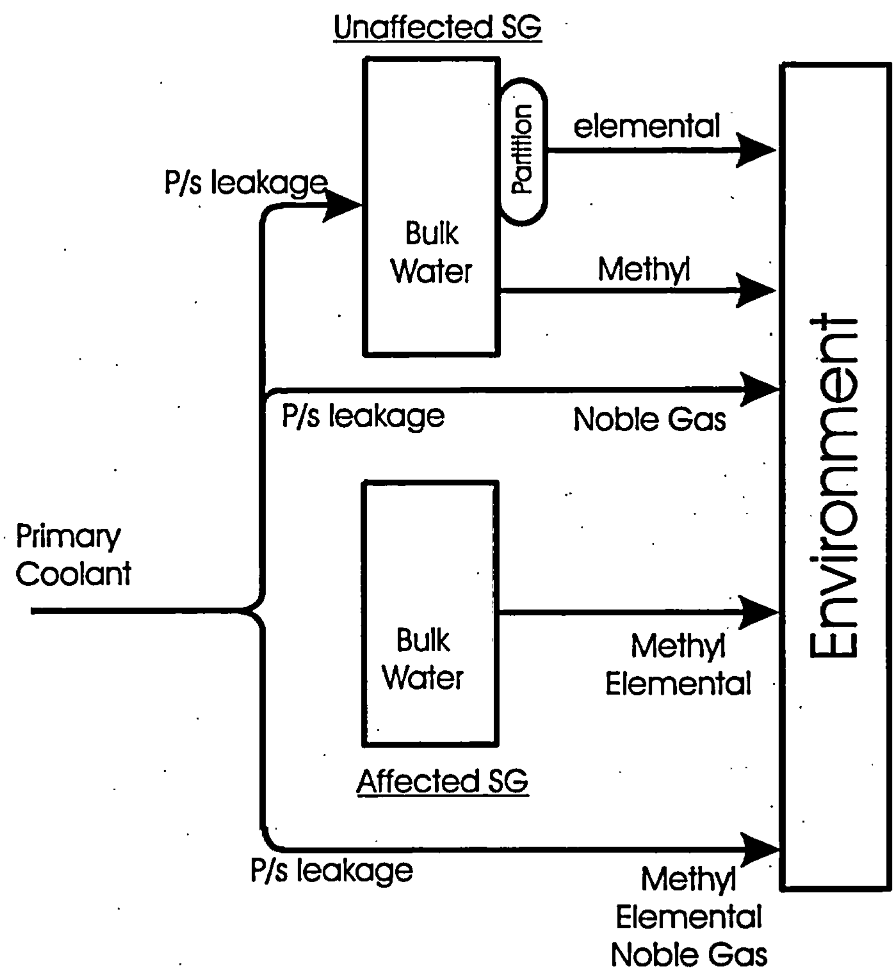
Table 2.9.2-9
FHA Dose Analysis Assumptions

Parameter	Value
Reactor power, Mwt (including fuel management factor)	1811
Power Peaking Factor	1.75
Number of damaged fuel assemblies	1
Time after reactor shut-down, hr	100
Fuel rod gap fractions (non-LOCA) I-131 other halogens Kr-85 other noble gases	 0.08 0.05 0.1 0.05
Iodine species above water elemental iodine organic (methyl) iodide	 0.57 0.43
Pool DF elemental iodine organic (methyl) iodide particulate Overall Pool DF	 500 1 ∞ 200
Containment net free volume, ft^3	1.0E+06
Exhaust flow rate, cfm	7.68E+04
Duration of activity release, hr	2
Iodine removal efficiency Containment release (all iodine forms) Fuel Pool release elemental iodine organic (methyl) iodide particulate	 0 0.9 0.7 n/a

Table 2.9.2-10
TMA Dose Analysis Assumptions

Parameter	Value
Reactor power, Mwt (including fuel management factor)	1811
Power Peaking Factor	1.75
Number of damaged fuel assemblies Region 1 Region 2	 5 hot, 4 cold 9 cold
Time after reactor shut-down hot assemblies cold assemblies	 100 hours 60 days
Fuel rod gap fractions I-131 other halogens Kr-85 other noble gases	 0.08 0.05 0.1 0.05
Iodine species above water elemental iodine organic iodide	 0.57 0.43
Pool DF elemental iodine organic iodide particulate Overall Pool DF	 500 1 ∞ 200
Exhaust flow rate, cfm puff (5 second activity release)	 1.11E+08
Iodine removal efficiency for all forms to environment	0

Figure 2.9.2-1
MSLB Activity Transport Model



Iodine and Noble Gas Transport Model

Figure 2.9.2-2
LRA Activity Transport Model

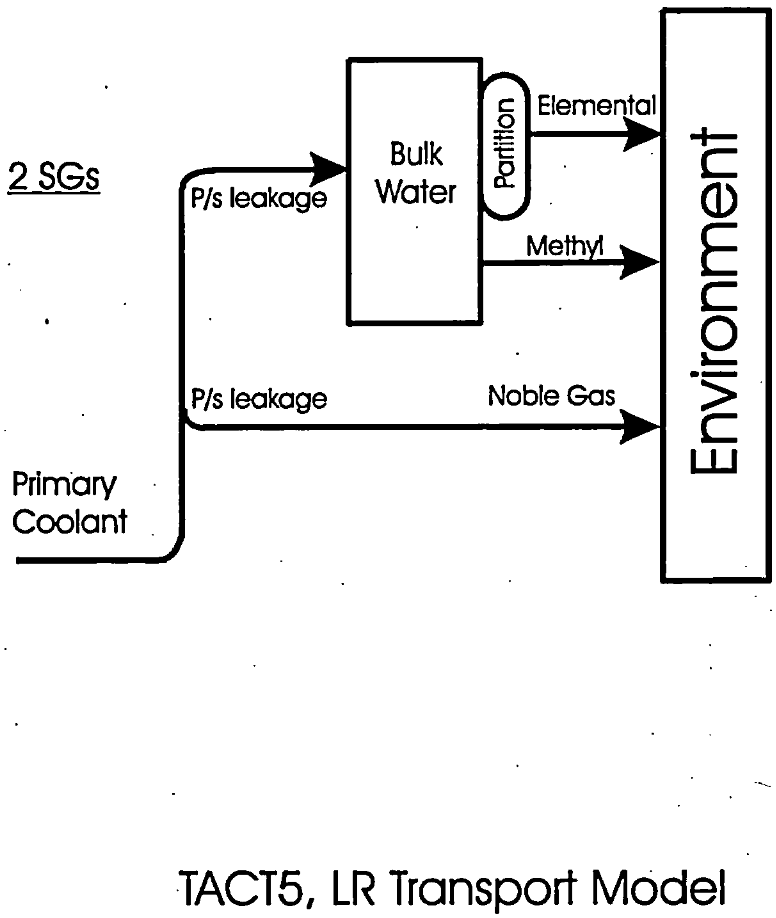


Figure 2.9.2-3
SGTR Activity Transport Model

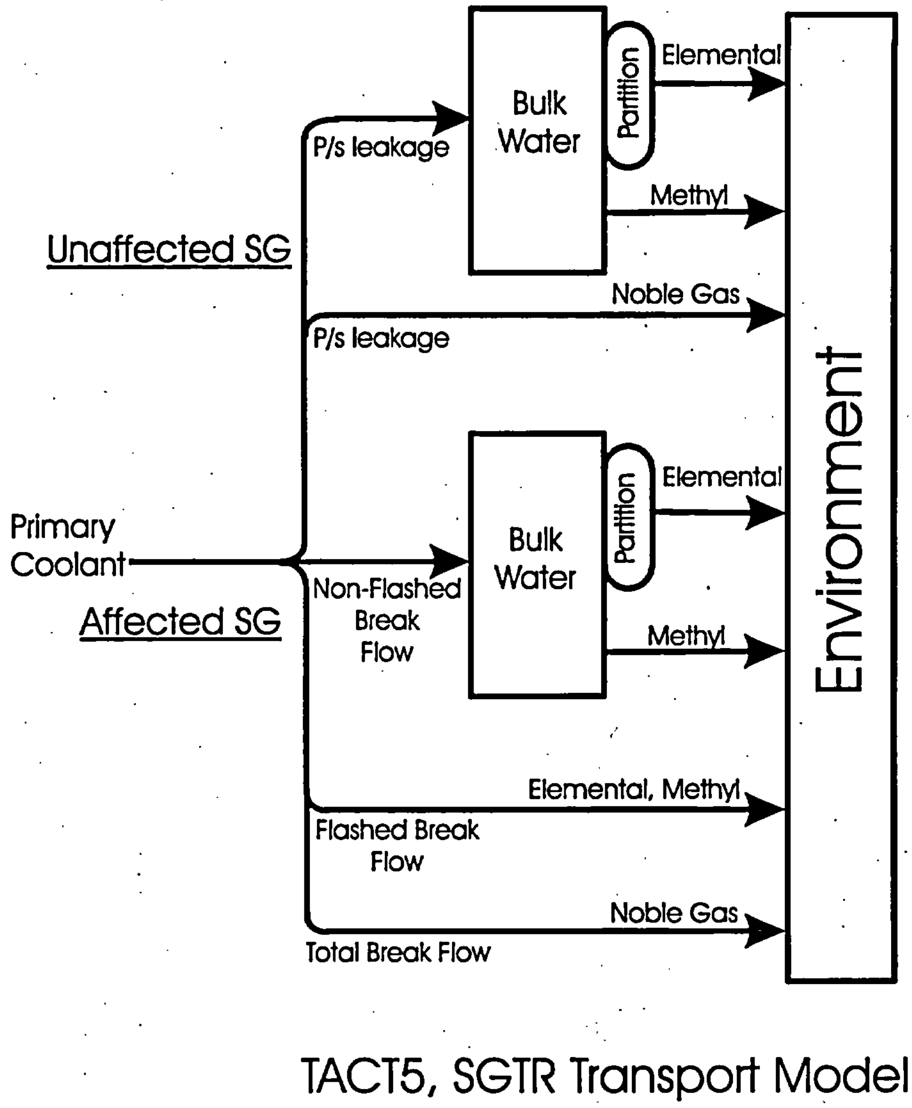
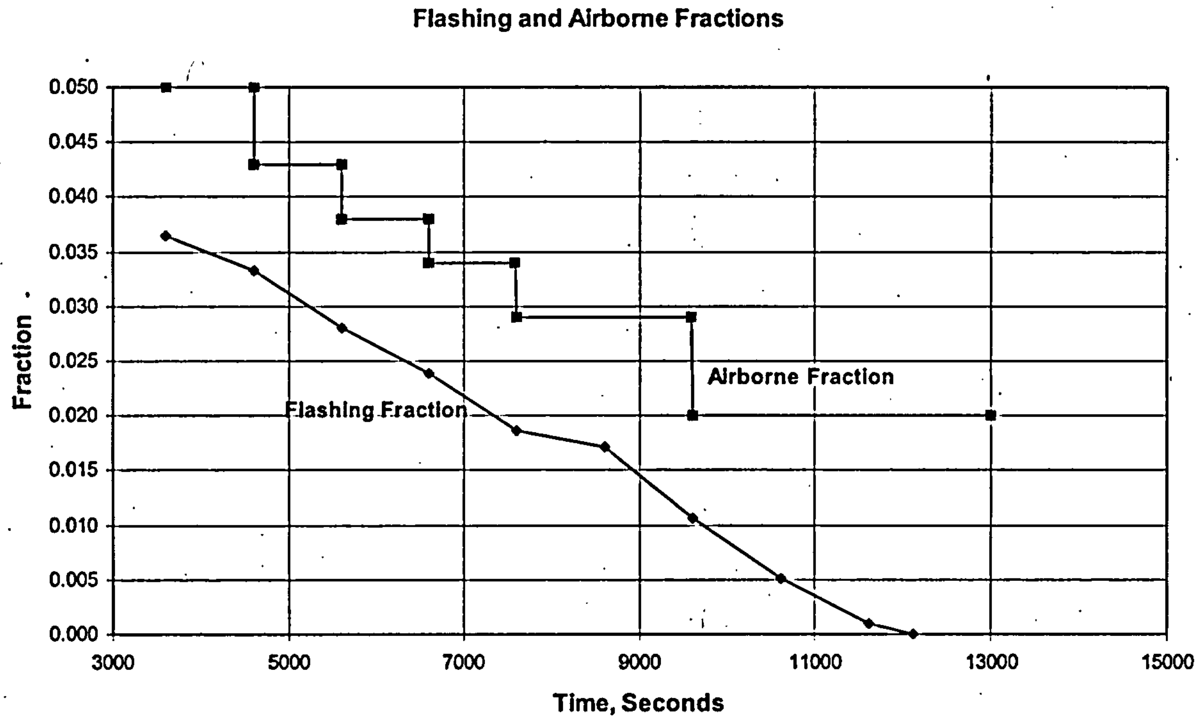


Figure 2.9.2-4
ECCS Leakage
Flashing and Airborne Fractions



2.9.2.3 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 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, 2005.2.9.2.3.

2.9.2.4 Conclusion

The Ginna staff has reviewed the various design basis accident (DBA) analyses performed in support of the proposed EPU for their potential radiological consequences and concludes that the analyses adequately account for the effects of the proposed EPU. The Ginna staff further concludes that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10CFR50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP 15.0.1. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the radiological consequences of DBAs.

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

2.10.1.1 Regulatory Evaluation

The Ginna Nuclear Power Plant, LLC (Ginna) staff conducted its review to ascertain the overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that Ginna has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable (ALARA). The Ginna staff review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones and plant area accessibility. The Ginna staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The Ginna staff considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary.

The NRC's acceptance criteria for occupational and public radiation doses are based on 10CFR20 and General Design Criterion (GDC)-19. Specific review criteria utilized by NRC are contained in the Standard Review Plan (SRP) Sections 12.2, 12.3, 12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

2.10.1.1.1 Ginna Current Licensing Basis

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 and to reconfirm and document their safety. The results of the SEP review for Ginna Station were published in NUREG-0821, Integrated Plant Safety Assessment Report (IPSAR), which was completed in August 1983. The IPSAR describes the methods used by the NRC to assess the 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. The adequacy of the Ginna Station design relative to conformance with GDC 19 is described in the Ginna UFSAR section 3.1.2.2.10, General Design Criteria 19 – Control Room. LR section 2.9.2, "Radiological Consequences Analyses Using Alternative Source Terms," summarizes the EPU assessment of impact on post-accident dose consequences at the site boundary and at locations on-site that require continuous occupancy, such as the control room.

Ginna Station's current licensing basis with respect to radiation protection of plant personnel and the public includes the following:

1. Normal Operation Radiation Levels and Shielding Adequacy

In accordance with 10CFR20, and as discussed in Ginna UFSAR section 12.3, radiation shielding at Ginna Station is designed for operation at the maximum calculated thermal power and to limit the radiation levels during normal operation at the site boundary to below those levels allowed for continuous non-occupational exposure. Shielding design is discussed in Ginna UFSAR section 12.3.2. Accessibility of the plant is facilitated by plant shielding and radiation monitoring and is controlled by procedures which take into account the requirements of 10CFR20. Plant shielding design and procedural controls ensure that operator exposure is maintained below the levels allowed for occupational exposure set by 10CFR20.

2. Radiation Monitoring Setpoints

As discussed in Ginna UFSAR sections 11.5 and 12.3.4, the radiation monitors installed at Ginna Station can be classified into four categories: a) area, b) airborne, c) process and d) effluent. Area and airborne radiation monitors are included as radiation protection features and provide radiation / radioactivity monitoring to support control of radiation exposure of plant personnel. Process and effluent radiation monitors are provided in support of radioactivity monitoring in gaseous or liquid process streams, or effluent release points to unrestricted areas, to support control of radiation exposure of both plant personnel and the public. Post-accident monitoring is provided in accordance with Regulatory Guide 1.97 requirements to give notice of significant radioactive releases from the plant. The high alarm and alert setpoints for the radiation monitors are based on meeting the above objectives.

3. Post Accident Vital Area Accessibility

As discussed in Ginna UFSAR section 12.3.2.2.6, in response to NUREG 0737, Item II.B.2, a plant radiation shielding design review of vital areas and equipment was conducted in order to ensure adequate personnel access to vital areas and protection of safety equipment for post-design basis accident operations.

The design basis vital area access review that supports Ginna Station's licensing basis relative to vital area dose rates / operator doses while performing post-LOCA vital missions is 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. NRC acceptance and approval of the vital area access assessment performed by Ginna was documented in an NRC Safety Evaluation Report (SER) dated May 23, 1984. Ginna UFSAR Amendment 81 dated January 2002 (elimination of Post-Accident Sampling System), Amendment 87 dated February 24, 2005 (implementation of Alternate Source Term to estimate the dose consequences at the site boundary and control room), and Amendment 90, dated May 5, 2005 (elimination of the hydrogen recombiners and hydrogen monitors from the Technical Specifications, see LR section 2.6.4) have impacted the vital access requirements / assessment documented in the

1979 Design Review Report. Further information regarding the Ginna vital access review is provided in LR section 2.10.1.2.3, Post-Accident Vital Area Accessibility.

4. Normal Operation Radwaste Effluents and Annual Dose to the Public

As discussed in Ginna UFSAR sections 11.2.3 and 11.3.3, radiation exposure of the public due to normal operation radwaste effluents is determined by compliance with the Offsite Dose Calculation Manual which in turn invokes compliance with 10CFR20, 10CFR50, Appendix I and 40CFR190. Conformity with the design objectives of 10CFR50, Appendix I is also discussed in the referenced UFSAR sections.

5. Ensuring that Occupational and Public Radiation Exposures are ALARA

As discussed in Ginna UFSAR section 12.1, the Radiation Protection Program at Ginna Station shall ensure that internal and external radiation exposures to station personnel, contractor personnel, and the general population resulting from station operation will be within applicable limits and will be ALARA. The bases of the Radiation Protection Program are that doses to personnel will be maintained within the limits of 10CFR20.

Ginna UFSAR section 11.1.1.2 discusses implementation of the overall requirements of 10CFR50, Appendix I as to utilization of radwaste treatment equipment to ensure that radioactive discharges and public exposure are ALARA has been formalized in the Technical Specification requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual.

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

In accordance with NRC SER related to the License Renewal of R. E. Ginna Nuclear Power Plant (Docket No. 50-244), dated May 2004, Ginna Station is approved for 60 years of operation. License Renewal focuses on age-related degradation of structures, systems and components that fall within the scope of license renewal. System and system component materials of construction, operating history and programs used to manage aging effects are documented in the above Report. NRC SER section 2.3.3.13 discusses radiation monitoring system components within the scope of license renewal.

The EPU assessment of impact on normal plant radiation levels including shielding adequacy, radiation monitoring setpoints, post-accident vital area accessibility, and normal operation radwaste effluents does not impact the list of components that fall within the scope of license renewal. No new components are being added to the scope of license renewal. Consequently, the EPU evaluation discussed in this section has no effect on the License Renewal of the Ginna Station.

2.10.1.2 Technical Evaluation

2.10.1.2.1 Normal Operation Radiation Levels and Shielding Adequacy

Introduction

Cubicle wall thickness is specified not only for structural and separation requirements, but also, to provide radiation shielding in support of radiological equipment qualification, and to reduce operator exposure during all modes of plant operation, including maintenance and accidents.

Conservative estimates of the radiation sources in plant systems and personnel access requirements form the bases of normal operation plant shielding and radiation zoning. These radiation source terms are primarily derived from conservative estimates of the reactor core and reactor coolant (also called primary coolant) isotopic activity inventory and are referred to as "design basis" source terms. EPU will impact the isotopic activity inventory in the core. In addition, since the "design basis" reactor coolant source term is based on 1% fuel defects, the EPU will result in an increase in the "design basis" reactor coolant activity concentration.

The "expected" radiation source terms in the coolant will also be impacted by the EPU. "Expected" source terms are less than that allowed by the plant Technical Specifications and are usually significantly less than the "design basis" source terms. The impact of the EPU on the normal operation dose rates and the adequacy of existing shielding are evaluated to ensure continued safe operation within regulatory limits. The assessment is broken into two parts; the impact of EPU on a) plant radiation levels during normal operation, and b) adequacy of existing shielding for normal plant operation.

The shielding design basis for Ginna Station is summarized in Ginna UFSAR section 12.3. The original plant shielding design was based on a core power level of 1520 MWt and a one-year fuel cycle length. The analysis in this section includes operation at a core power of 1811 MWt and an 18-month fuel cycle. An increase of fuel cycle length will increase the inventory of long-lived isotopes in the core and in the reactor coolant. The activity inventory of a few isotopes which are produced primarily by neutron activation of stable or long-lived fission products will also increase due to longer accumulation time.

The EPU requires an increase of the nuclear fission rate and consequently, an increase of neutron flux and the fission product generation rate. This leads to an increase of the fission product inventory in the core and spent fuel, and an increase of neutron and gamma flux leaking out of the reactor vessel. The increase of neutron flux results in an increase of neutron activation products in the reactor cooling system, control rod assemblies, reactor internals and the pressure vessel. The increase in the core inventory of fission products and actinides due to the EPU will also increase the activity concentrations in the reactor coolant due to fuel defects. The activity concentrations in the secondary system will also increase due to primary-to-secondary leakage in the steam generators. The radiation source in the downstream systems will undergo a corresponding increase. This increase in radioactivity levels and the associated increase in radiation source strength result in an increase of radiation levels in the containment building, auxiliary building, intermediate building, turbine building, all-volatile-treatment building, and other locations, including offsite, which are subject to direct shine from radiation sources contained in these buildings.

Description of Analyses and Evaluations

The EPU evaluation utilizes scaling techniques to determine the impact of EPU on plant radiation levels. This evaluation takes credit for conservatism in existing shielding analyses and the site ALARA Program to demonstrate continued adequacy of current plant shielding to support compliance with the operator exposure limits of 10CFR20.

a. <u>Normal Operation Radiation Levels</u>

For the same source-shield-detector configuration, the dose rate at a given detector point is directly proportional to the radiation source strength in the source region. The impact of increasing the reactor power from the current licensed level of 1520 MWt to the analyzed power level of 1811 MWt on the neutron flux and gamma flux in and around the core, fission product and actinide activity inventory in the core and spent fuels, N-16 source in the reactor coolant, neutron activation source in the vicinity of the reactor core, and fission/corrosion products activity in the reactor coolant and downstream systems, was examined, and the increase quantified. This flux or activity increase factor for a given radiation source was determined to be the EPU scaling factor for the expected dose rate due to that source.

The EPU assessment with regard to normal operation radiation levels is divided into four areas:

<u>Areas near Reactor Vessel</u>: During normal operation, the radiation source in the reactor core is made up of neutron and gamma fluxes which are approximately proportional to the core power level. The radiation sources during shutdown are the gamma fluxes in the core and the activation activities in the reactor internals, pressure vessel, and primary system piping walls, which also vary approximately in proportion to the reactor power.

The radiation dose rate near the reactor vessel is determined by the leakage flux from the reactor vessel. Therefore, an uprate from the current licensed core power of 1520 MWt to an analyzed core power of 1811 MWt is expected to increase the normal operation radiation levels in areas near the reactor vessel by a factor of approximately 1.19, i.e., 1811/1520.

<u>In-Containment Areas Adjacent to the Reactor Coolant System</u>: During normal operation, the major radiation source in the reactor coolant system components located within containment is N-16. With the core power increase from 1520 MWt to the analyzed core power of 1811 MWt, the fast neutron flux is expected to increase by approximately 19%. The coolant residence time in the core and the transit time are not expected to change significantly due to uprate. Therefore, the EPU scaling factor for the areas subjected to the N-16 source is 1.19.

The deposited corrosion product activity depends on the reactor coolant chemistry and the approximately 19%. The corrosion product activity deposits and the associated shutdown dose rate is also expected to increase by 19%.

Areas near Irradiated Fuels and Other Irradiated Objects: These areas include the refueling canal, spent fuel pool, incore instrumentation drive assembly area, and other areas housing neutron irradiated materials. The radiation source is the gamma rays from the fission products and activation products, which are determined by the fission rate, neutron flux level and the irradiation time.

Since both the fission products and the activation products are expected to increase by approximately 19% for a core power increase from 1520 MWt to the analyzed core power level of 1811 MWt, the EPU scaling factor for the areas subjected to irradiated fuels and other irradiated sources is 1.19.

Areas outside Containment where the Radiation Source Is Derived from the Primary Coolant Activity: In most areas outside the reactor containment, the radiation sources are fission products and corrosion products in the primary coolant or down-stream sources originating from the primary coolant activity. Following EPU, both the fission products and the activated corrosion products are expected to increase by approximately 19% for a core power increase from 1520 MWt to the analyzed power level of 1811 MWt.

The EPU scaling factor for the areas outside containment where the radiation source is derived from the primary coolant activity is, in general, 1.19 with the exception of the area near the condensate polishing system. The radiation level near the condensate polishing system may increase slightly greater than the percentage of EPU due to the increased steam flow rate and moisture carryover fraction associated with EPU.

b. Plant Shielding Adequacy

Shielding is used to reduce radiation dose rates in various parts of the station to acceptable levels consistent with operational and maintenance requirements and to maintain the dose rates at the site boundary to below those allowed for continuous non-occupational exposure.

The original Ginna Station shielding design was based on plant operation at a core power level of 1520 MWt, upon generalized occupancy requirements in various radiation zones of the station, and upon conservative reactor coolant source terms assuming 1% fuel defects.

The EPU evaluation takes into consideration that the occupancy requirements are not affected by EPU. Similarly, the layout / configuration of systems containing radioactivity are unchanged by the EPU. Consequently, the EPU evaluation focused on determining an EPU scaling factor based the radiation source terms used in the original plant shielding design as documented in Ginna UFSAR 12.3.2, and the EPU radiation source terms discussed in LR section 2.9.1; "Source terms for Radwaste Systems Analyses," specifically, the design basis fission and corrosion product activity concentrations in the reactor coolant at the analyzed core power level of 1811 MWt and with an 18-month fuel cycle length.

The source terms at the analyzed power are compared to the source terms used in the original shielding design to evaluate the adequacy of the shielding design. The EPU evaluation takes into consideration a) the conservative analytical techniques used to establish plant shielding design, b) the Technical Specification limits on the reactor coolant activity concentrations, and c) the station ALARA program for minimizing the radiation exposure to plant personnel.

<u>Reactor Primary Shield</u>: As discussed in Ginna UFSAR 12.3.2.2.1, the primary shield is a reinforced concrete structure that surrounds the reactor vessel. The primary function of the primary shield is to attenuate the neutron and gamma fluxes leaking out of the reactor vessel. Fuel cycle length has insignificant impact on the maximum dose rates around the reactor vessel which are based on the neutron and gamma flux during power operation.

The Ginna staff reviewed the fluence calculations and confirmed that the original design calculations remain bounding for EPU conditions. With continued use of low leakage fuel management following EPU, the existing primary shielding remains adequate, and the estimated dose rates adjacent to the reactor vessel / primary wall remain within original design.

<u>Reactor Secondary Shield</u>: As discussed in Ginna UFSAR section 12.3.2.2.2, the secondary shield is a reinforced-concrete structure that surrounds the reactor coolant system and the steam generators. The secondary shield also includes the reactor containment structure and the concrete operating floor over the primary coolant loops. The primary function of the secondary shield is to attenuate the N-16 source, which emits high-energy gammas. The secondary shield was designed to limit the full power dose rate outside the containment building to less than 1 mr/hr. The N-16 source is expected to increase by approximately 19%. The N-16 activity is not impacted by fuel cycle length. The impact of the estimated 19% increase in source terms is bounded by the conservative analytical techniques used to establish plant shielding design, and the current secondary shield is determined adequate for continued safe operation following EPU.

<u>Fuel Handling Shielding</u>: This shielding provides protection during all phases of removal and storage of spent fuel and control rods. As noted in Ginna UFSAR section 12.3.2.2.4 and Tables 12.3-2b and 12.3-5, the design basis spent fuel source term utilized for shield design is based on a reactor power of 1520 MWt, a full power exposure of 1000 days, and a minimum fuel removal delay time of 100 hours. The fuel handling shield was designed to ensure a calculated dose rate in the auxiliary building general areas to be less than 1.0 mr/hr.

EPU is expected to increase the gamma source from irradiated fuel by approximately 19%. Fuel cycle length will increase the inventory of long-lived isotopes in the irradiated fuel. However, this is not a significant concern as the dose rates near the refueling canal and the spent fuel pool are dominated by the shorter half-life isotopes in the freshly discharged spent fuel assemblies. The impact of the estimated 19% increase in source terms is bounded by the conservative analytical techniques used to establish plant shielding design, and the current spent fuel shielding is determined adequate for continued safe operation following EPU.

All Other Shielding Outside Containment: In support of shielding provided outside the containment where the radiation sources are either the reactor coolant itself or down-stream sources originating from coolant activity, a review was performed of the EPU design primary coolant source terms (fission and activation products) vs. the original design basis primary coolant source terms. It is noted that the analyzed design primary coolant source terms utilized for the EPU reflect a core power level of 1811 MWt, operation with an 18-month fuel cycle, and more advanced fuel burn-up modeling/libraries as compared to the computer codes used in the original analyses.

The EPU assessment concluded that the Plant Technical Specifications will limit the EPU reactor coolant and degassed reactor coolant source terms, and associated dose rates assuming various shielding configurations, to less than 81% of the original design basis values. The EPU assessment also showed that the Technical Specification limits on the reactor coolant gross activity will maintain the EPU reactor coolant gas activity, and associated dose rates assuming various shielding configurations, at approximately the original design basis values. It is therefore concluded that the shielding design based on the original design basis primary coolant activity remains valid for the EPU condition.

Results

Since the plant is already operating with an 18 month fuel cycle, the normal operation radiation levels in most of the plant area are expected to increase by approximately 19%, i.e., the commensurate percentage of core uprate. The exposure to plant personnel and to the offsite public is also expected to increase by the same percentage.

The increase in expected radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant. This is because the increase is offset by the:

a. conservative analytical techniques typically used to establish shielding requirements,

b. conservatism in the original "design basis" reactor coolant source terms used to establish the radiation zones, and

c. plant Technical Specifications that limit the reactor coolant concentrations to levels at or below the original design basis source terms.

As indicated in Ginna UFSAR sections 12.1.3 and 12.5.2, individual worker exposures will be maintained within the regulatory limits of 10CFR20 for occupational exposure by the site ALARA program that controls access to radiation areas. In addition, the Offsite Dose Calculation Manual ensures that the radiation levels at the site boundary due to direct shine from radiation sources in the plant will be maintained within the regulatory limits of 10CFR20 and 40CFR190 for continuous non-occupational exposure.

The above EPU assessment also demonstrates continued compliance with GDC 19 with regard to radiation protection, insofar that actions can continue to be taken in the control room to operate the nuclear power unit safely during normal operation.

2.10.1.2.2 Radiation Monitoring Setpoints

Introduction

The function of area monitor alarm setpoints is to provide an early warning of changing radiological conditions in a specified area. The function of alarm setpoints for process/effluent monitors is to indicate leakage or malfunction of equipment, or a potential for an activity release that may exceed the release rate limit. The high alarm setpoint of many effluent monitors will also initiate interlocks that terminate activity release to the environment. The function of the post-accident radiation monitors is to give notice of significant radiation levels within plant areas or in environmental releases from the plant.

EPU will increase the activity level of radioactive isotopes which will result in an increase of radiation levels in various plant areas and potentially increase the radioactive environmental releases from the station.

Description of Analyses and Evaluations

The EPU evaluation examined the impact of increased radioactivity levels in the monitored streams / areas, and the associated background radiation levels, to assess the applicability of the current radiation monitor setpoint basis / values following EPU.

As discussed earlier, the EPU will increase the activity level of radioactive isotopes in most streams/components and the associated radiation levels by approximately the percentage of the core power uprate. The relative isotopic compositions in the process and effluent streams are not expected to change due to EPU.

The bases of the radiation monitor setpoints at Ginna station are either a regulatory commitment (i.e., the definition of a high radiation zone, or radioactivity in environmental releases that are fractions or multiples of the release rate limits and are intended to give notice of releases approaching the limits of 10CFR20), a multiple of the background, or a "high" value indicating an unusual event (such as leakage or malfunction of systems), that leads to a sudden increase of the activity level in the monitored stream. The setpoint basis are not power level dependent, and the setpoint values are established using plant operating data and are reviewed frequently and adjusted as required.

Results

The EPU evaluation determined that all of the radiation monitor setpoint bases, and the methods of setpoint determination, continue to be valid following EPU.

2.10.1.2.3 Post Accident Vital Area Accessibility

Introduction

In accordance with NUREG-0737, II.B.2, and its predecessor NUREG-0578, Item 2.1.6.b, vital areas are those areas within the station that will or may require access / occupancy to support accident mitigation following a loss of coolant accident (LOCA). In accordance with the above regulatory document, all vital areas and access routes to vital areas must be designed such that operator exposure while performing vital access functions remain within regulatory limits.

This section focuses on areas that may require short-term, one-time or infrequent access following a LOCA. On-site locations that require continuous occupancy and a demonstration of 30-day habitability are addressed in LR section 2.9.2, "Radiological Consequences Analyses Using Alternative Source Terms."

As indicated earlier, the design basis vital area access review that supports Ginna Station's licensing basis relative to vital area dose rates / operator doses while performing post-LOCA vital missions is 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. Table 4-3 of the above report summarizes the estimated operator mission doses for the 14 target areas that were determined to require short term occupancy.

Several of the access requirements listed in the 1979 Design Review Report are no longer required due to changes in licensing basis since 1979, specifically access requirements for post-accident sampling and access requirements to the hydrogen recombiner panel. The access requirements for sampling were predicated upon the perceived need for samples of the containment sump, containment atmosphere, and reactor coolant system within a relatively short period of time after an accident occurred. However, post-TMI studies have shown that other means can be employed to determine the degree of core damage and classify events for emergency planning purposes. Consequently, the Post-Accident Sampling System (PASS) was removed from the Technical Specifications in Amendment 81 using the consolidated line item improvement process (CLIIP) per TSTF-366. Hydrogen recombiners were removed from the Technical Specifications in Amendment 90 per TSTF-447, and the associated NRC SER. In general, post-TMI information determined that hydrogen production in a DBA was sufficiently slow (see UFSAR Figures 6.1-10 and 6.1-12) that other means could be employed to reduce the concentration to below combustible limits, if needed. In the event of a severe accident, the rate of hydrogen production exceeds the capability of the recombiners, causing the recombiners to become an unwarranted ignition source. Therefore, entry into this area is no longer considered necessary for short-term post-accident operations.

Also, a recent review of the Emergency Operating Procedures indicated that there are no Emergency Operating Procedure steps that discuss the need to change the auxiliary building, spent fuel pool or control room accident filters. In addition, the Emergency Operating Procedure review established that the vital area access requirements noted in the 1979 Report are not considered "required" steps but "enhancements" to be undertaken only if the environment is considered acceptable by Health Physics (HP) personnel. Regarding the time when access may be envisioned, the review determined that immediate access was not required for any of the operator actions listed in the 1979 Design Review Report. Based on the above review, and except as noted, it has been determined that for purposes of demonstrating availability, the earliest access time to be evaluated is at 1 day following the accident. Access to the radwaste panel is to be evaluated at 10 days following the accident.

The updated list of operator access requirements includes an additional action identified subsequent to the issuance of the 1979 Design Review Report, i.e., throttling of the service water flow to the component cooling water heat exchangers to support cooling of the residual heat removal system while in the containment recirculation phase of operation. This action must be completed prior to initiation of the recirculation phase during which sump water is recirculated back to the reactor coolant system following a loss-of-coolant accident.

Core power uprate will typically increase the activity level in the core by the percentage of the uprate. The radiation source terms in equipment / structures containing post-accident fluids, and the corresponding environmental radiation levels, will increase proportionately to the uprate. In addition, factors that impact the equilibrium core inventory, and consequently the estimated radiation environment, are fuel enrichment and burnup. These additional changes could result in activity levels in the core that are typically higher than the core power ratio associated with the uprate.

As discussed earlier, Ginna Station has been approved for use 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. However, for the reasons summarized in SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining of Operating Reactors," dated June 30, 1998, the EPU assessment, for purposes of evaluating the impact on operator exposure while performing vital functions in areas that require infrequent access, is based on TID 14844, "Calculation of Distance Factors for Power and Test Reactors," dated 1962, source terms. The alternative source term benchmarking study reported in SECY-98-154 concluded that results of analyses based on TID 14844 would be more limiting earlier in the event, after which time the alternative source term results would be more limiting. Post-LOCA access to vital areas for purposes of accident mitigation and safe shutdown occurs earlier on in the event when the original TID 14844 source term is more limiting.

Description of Analyses and Evaluations

The EPU assessment is based on an analyzed core power level of 1811 MWt and implementation of an 18-month fuel cycle. The methodology utilized in the EPU evaluation is to demonstrate, using scaling techniques or location-specific analyses, continued compliance with the operator exposure dose limits of 5 Rem provided in NUREG-0737, II.B.2 and its predecessor NUREG-0578, Item 2.1.6.b. When the pre-EPU vital area assessment establishes radiation levels in the area, but does not develop operator mission doses, the EPU assessment develops the estimated radiation levels following EPU. There are no acceptance criteria for this case. The licensing basis for such cases is the availability of radiation dose rate information such that the licensee can factor this information into any post-accident access planning.

Scaling Evaluations

The impact of the EPU on the post-LOCA gamma radiation dose rates developed in the 1979 Design Review Report and utilized to determine operator exposure during vital area access is evaluated by comparing the gamma source terms, based on the original core inventory used to develop the post-LOCA dose rates, to the gamma source terms, based on the EPU core inventory. This approach takes into consideration that a) the post-LOCA operator mission requirements, including the task description and required time / duration for access is not impacted by the EPU, and b) EPU does not impact the operation and layout / arrangement of plant radioactive systems.

Theoretically, following EPU, the post-LOCA environmental gamma dose rates and the operator dose per identified mission should increase by approximately 19% (1811 MWt / 1520 MWt). However, because the EPU analyzed core reflects: a) operation with an 18-month fuel cycle, b) more advanced fuel burnup modeling/libraries than used in the original analyses, and c) a 4% margin to address uncertainty in fuel management schemes following EPU, the calculated EPU scaling factor values deviate from the core power ratio.

The EPU assessment is essentially a two-step process. The first develops a bounding EPU dose rate scaling factor vs time, and the second multiplies the pre-EPU personnel dose /dose rates at target areas identified in the licensing basis by the bounding EPU scaling factor.

The pre-EPU and the EPU core inventories are utilized 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 gamma dose rates is estimated by ratioing the gamma energy release rates weighted by dose rates, as a function of time, for the EPU analyzed core power level, to the corresponding weighted source terms based on the pre-EPU analyzed core power level. To address the fact that the vital access locations are outside containment, the "unshielded" values include the shielding effect of a pipe wall thickness associated with a 2-inch nominal diameter pipe. This ensures that the results are not skewed by photons at energies less than 25 kev which will be substantially attenuated by any piping sources.

To evaluate the factor impact of the EPU on post-LOCA gamma dose rates (vs time) in areas that are "shielded," the pre-EPU as well as the 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 are 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 rate scaling factors vary with source, time, as well as shielding, to cover all types of analysis models/assessments, the maximum dose rate scaling factor with respect to time developed from the above assessments is used for all source/receptor combinations, with or without shields, for the time period identified in the vital access assessment.

Location-Specific Analyses

The dose to the operator while performing the function of throttling service water flow to the component cooling water heat exchangers prior to initiation of the containment recirculation phase is estimated at EPU conditions using TID source terms. The earliest access time is expected to be about ten minutes after the accident. The only radiation source at that time is the airborne source inside containment. The source term assumptions for the containment airborne source are consistent with that used in the 1979 Design Review Report, i.e., 100% of the core noble gases and 25% halogens. No credit is taken for any removal mechanisms other than decay.

Access Routes to Target Areas

Ginna is updating the planned access routes to the target areas from that described in Table4-1 of the 1979 Design Review Report. The new access route follows a path from the north side of the control room to the south entrance of the auxiliary building at El. 271 ft utilizing a path east of the containment outside any structures or buildings. Access between the floors of the auxiliary building will be via the east stairway.

Results

The bounding EPU dose / dose rate scaling factors applicable to the post-LOCA vital area access documented in the 1979 Design Review Report are presented below.

EPU Dose and Dose Rate Scaling Factors for Vital Area Access					
Applicability	**1 hour**	**1 day**	**10 days**	**30 day**	**6 months**
Areas G, H, & I	1.29	1.38	1.31	1.32	1.51

LR Table 2.10.1-1 presents the vital area access dose estimates following the EPU. The table demonstrates that the EPU post-LOCA vital area operator dose estimates remain within the regulatory limit of 5 Rem whole body listed in NUREG-0737, II.B.2.

2.10.1.2.4 Normal Operation Radwaste Effluents and Annual Dose to the Public

Introduction

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials. Evaluation of the radioactive waste management systems for EPU conditions is provided in LR section 2.5.6.1 (gaseous waste system), LR section 2.5.6.2 (liquid waste system), and LR section 2.5.6.3 (solid waste system).

Liquid and gaseous radwaste systems are designed such that the plant is capable of maintaining normal operation offsite releases and doses within regulatory limits. The actual performance and operation of installed equipment, as well as reporting of actual offsite releases and doses, is controlled by the requirements of the Offsite Dose Calculation Manual.

There are no specific limits associated with generation of solid radwaste other than those associated with transportation. However, onsite storage of radwaste may result in increased public exposure at the site boundary which is controlled by Federal regulations.

Core uprate will increase the activity level of radioactive isotopes in the reactor and secondary coolant and steam. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems where they are located prior to discharge. As the activity levels in the coolants and steam are increased, the activity level of radwaste inputs, and subsequent environmental releases, are proportionately increased.

Description of Analyses and Evaluations

The methodology used in the EPU evaluation is to demonstrate, using scaling techniques, continued compliance with the annual dose limits to an individual in an unrestricted area set by 10CFR20, 10CFR50, Appendix I and 40CFR190 resulting from radioactive gaseous and liquid effluents released to the environment following EPU. Note that limits on dose to the public resulting from normal operation are addressed in 10CFR20, 10CFR50 Appendix I as well as 40CFR190, however, 10CFR50 Appendix I is the most limiting. 10CFR20 does have a release rate criteria that does not exist in 10CFR50 Appendix I, but the plant Technical Specifications and the Offsite Dose Calculation Manual control actual performance and operation of installed equipment and releases thus maintaining compliance with that aspect of 10CFR20. In addition, if the projected increase in offsite doses due to radioactive gaseous and liquid effluents either approach or exceed 10CFR50, Appendix I guidelines, then the Radiological Effluent Technical Specifications and the Offsite Dose Calculation Manual needs to be examined in order to determine continued compliance with 40CFR190.

The EPU does not change existing radioactive waste systems (gaseous and liquid) design, plant operating procedures or waste inputs as defined by NUREG-0017, Revision 1. Therefore, a comparison of releases can be made based on pre-EPU vs EPU inventories / radioactivity concentrations in the reactor coolant and secondary coolant / steam. As a result, the impact of the EPU on radwaste releases and Appendix I doses can be estimated using scaling techniques.

Scaling techniques based on NUREG-0017, Revision 1 methodology were utilized to assess the impact of EPU on radioactive gaseous and liquid effluents at Ginna. Use of the adjustment factors presented in NUREG-0017, Revision 1 allows development of coolant activity scaling factors to address EPU.

The conservatively performed EPU analysis utilized the plant core power operating history during the years 1999 to 2003, the reported gaseous and liquid effluent and dose data during that period, NUREG-0017, Revision 1, equations and assumptions and conservative methodology to estimate the impact of operation at the analyzed EPU core power level. The results were then compared to the comparable data from current operation on radioactive gaseous and liquid effluents and the consequent normal operation off-site doses.

The licensed reactor core power level during the 1999 to 2003 time frame was 1520 MWt. For the EPU condition, the system parameters utilized in the EPU analysis reflected the flow rates and coolant masses at an analyzed NSSS power level of 1817 MWt and a core power level of 1811 MWt. For the pre-EPU condition, the evaluation utilized offsite doses based on an average 5 year set of organ and whole body doses calculated from effluent reports for the years 1999 through 2003 including the associated average annual core power level extrapolated to 100 percent availability at the licensed power level.

Using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-EPU and EPU conditions, the maximum potential percentage increase in coolant activity levels due to the EPU, for each chemical group identified in NUREG-0017, was estimated.

To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-EPU conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the EPU and to demonstrate that the estimated off-site doses following EPU, although increased, will continue to remain below the regulatory limits.

The impact of EPU on solid radwaste generation was qualitatively addressed based on NUREG-0017, Revision 1, methodology, engineering judgment and the understanding of radwaste and affected plant system operation on the generation of solid radwaste.

The EPU assessment concluded the following:

a. <u>Expected Reactor Coolant Source Terms</u>

Based on a comparison of base vs. EPU input parameters, and the methodology outlined in NUREG-0017, Revision 1, the maximum expected increase in the reactor coolant source is approximately 18% for noble gases and 19% for other long half-life activity Considering the accuracy and error bounds of the operational data utilized in NUREG-0017, Revision 1, this percentage is well within the uncertainty of the existing NUREG-0017, Revision 1, based expected reactor coolant isotopic inventory used for radwaste effluent analyses.

b. Liquid Effluents

As discussed above, there is a maximum 19% increase in the radioactivity content of the liquid releases as input activities are based on long-term reactor coolant activity. Tritium releases in liquid effluents are assumed to increase approximately 19% (corresponding to the effective core power uprate percentage) since the analysis is identifying changes in an existing facility's power rating without changing its mode of operation.

c. Gaseous Effluents

For all noble gases, there will be a bounding maximum 19% increase of radioactivity content in effluent releases due to the effective core uprate percentage increase

Tritium releases in the gaseous effluents increase in proportion to their increased production (19%), which is directly related to core power and is allocated in this analysis in the same ratio as pre-EPU releases.

The impact of the EPU on iodine releases is approximated by the effective power level increase with the calculated increase in the reactor coolant I-131 of 19%.

For particulates, the methodology of NUREG-0017 specifies the release rate per year per unit per building ventilation system. This is not dependent on power level within the range of applicability. Particulates released via the turbine building from main steam leaks and air ejector exhaust are generally considered to be a small fraction of total particulate releases. Thus, minimal change would be expected for the EPU operations. However, a conservative approach is dictated by the fact that the annual effluent release reports do not delineate the "source" of particulates or iodines released. In addition at Ginna, tritium is included in the category with iodines and particulates (radionuclides with half-lives greater than 8 days). On the secondary side, moisture carryover is a major factor in determining the non-volatile activity in the steam. The multiplier applicable to the particulates released via the turbine building due to main steam leaks and air ejector exhaust is higher than the percentage of the EPU (primarily due to a conservatively estimated 6.7 fold increase in moisture carryover due to the EPU, coupled with a 19% increase in coolant concentration), however the contribution of particulates to the "Iodine and Particulate" category was insignificant compared to the dose contribution from iodine or tritium. For tritium and iodine, the latter had the greater increase due to moisture carryover resulting in a 29.1% increase in steam activity while the tritium increase was bounded by the increase in power. Thus the scaling factor for the entire "particulate and iodine" category was conservatively estimated at 29.1%.

d. Estimated Impact on Effluent Doses

LR Table 2.10.1-2 shows that, based on operating history, the maximum estimated dose due to liquid and gaseous radwaste effluents following EPU is significantly below the 10CFR50, Appendix I limits.

e. Solid Radioactive Waste

For Ginna, the volume of solid waste would not be expected to increase proportionally because the power uprate neither appreciably impacts installed equipment performance, nor does it require drastic changes in system operation or maintenance. Only minor, if any, changes in waste generation volume are expected. However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity bounded by the 19% maximum increase.

Thus, while the total long-lived activity contained in the waste following EPU is expected to be bounded by approximately 21%, i.e., 19% / 0.895 (average capacity factor during the five year evaluation period), over that currently stored on site, the increase in the overall volume of waste generation resulting from the EPU is expected to be minor.

It would also be expected that, in the long-term, the direct shine dose (the pre-EPU annual direct shine dose ranges from 7.9 mrem to 10.1 mrem during the five year period evaluated, as compared to the regulatory limit established by 40CFR190 which is 25 mrem/yr), would increase by approximately 21% as a) current waste decays and its contribution decreases, b) the radwaste is routinely moved offsite for disposal, and c) waste generated post-EPU enters into storage. As the impact on direct shine doses is cumulative from wastes generated over the plant lifetime and stored onsite, procedures and controls in the Offsite Dose Calculation Manual serve to monitor and control this component of the off-site dose and would limit, through administrative and storage controls, the offsite dose to ensure continued compliance with the 40CFR190 direct shine dose limits.

Results

As discussed previously, Ginna Station is committed to meeting the requirements of 40CFR190, 10CFR20 and 10CFR50, Appendix I. However, 10CFR50 Appendix I is the most limiting.

10CFR20 does have a release rate criteria that does not exist in 10CFR50 Appendix I, but the plant Technical Specifications and the Offsite Dose Calculation Manual control actual performance and operation of installed equipment and releases, thus maintaining compliance with that aspect of 10CFR20.

If the normal operation doses due to radioactive gaseous and liquid effluents either approach or exceed 10CFR50, Appendix I guidelines, then the Radiological Effluent Technical Specifications and the Offsite Dose Calculation Manual ensure continued compliance with 40CFR190.

The EPU has no significant impact on the expected annual radwaste effluent doses (i.e., this analysis demonstrates that the estimated doses following EPU will remain a small percentage of allowable Appendix I doses - see Table 2.10.1-2). It is therefore concluded that following EPU, the liquid and gaseous radwaste effluent treatment systems, in conjunction with the procedures and controls provided by the Offsite Dose Calculation Manual, will remain capable of maintaining normal operation offsite doses within the regulatory requirements.

2.10.1.2.5 Ensuring that Occupational and Public Radiation Exposures are ALARA

Introduction

The Radiation Protection Program at Ginna Station ensures that internal and external radiation exposures to station personnel, contractor personnel and the general population resulting from station operation will be within applicable limits and will be ALARA, as described in Ginna UFSAR section 12.1.

Implementation of the overall requirements of 10CFR50, Appendix I relative to utilization of radwaste treatment equipment to ensure that radioactive discharges and public exposure are ALARA are formalized in the Technical Specification requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual.

Description of Analyses and Evaluations

As noted in Ginna UFSAR section 12.1, ALARA procedures currently in place govern all activities in restricted areas at Ginna Station. Design features credited to support Ginna's commitment to ALARA operator exposures include shielding which is provided to reduce levels of radiation, ventilation which is arranged to control the flow of potentially contaminated air, an installed radiation monitoring system which is used to measure levels of radiation in potentially occupied areas and measure airborne radioactivity throughout the plant and respiratory protective equipment which is used as prescribed by the Radiation Protection Program.

Compliance with the requirements of the Offsite Dose Calculation Manual ensures that radioactive discharges and public exposure are ALARA.

The EPU assessments documented in LR section 2.10.1.2.1, LR section 2.10.1.2.2, LR section 2.10.1.2.3, and LR section 2.10.1.2.4 demonstrate that the dose limits imposed by regulatory requirements are met following EPU. The EPU does not impact the effectiveness of the design features credited to support Ginna's commitment to ALARA operator exposures. The intent of the ALARA procedures remain unchanged, specifically, a) the allowable limits on operator and public exposure and b) the intent to keep operator and public exposure at a minimum.

Results

It is concluded that no additional steps are necessary to ensure that dose increases are maintained ALARA.

2.10.1.3 Conclusion

The Ginna staff has assessed the effects of the proposed EPU on radiation source terms and plant radiation levels, the associated impact on shielding adequacy, radiation monitoring setpoints, post-accident vital area accessibility and normal operation radwaste effluents. The Ginna staff concludes that the evaluation adequately accounts for the effects on the proposed EPU on occupational and public radiation doses such that no additional steps are required to ensure that radiation doses will be maintained ALARA. Based on this, the Ginna staff concludes that the occupational and public radiation dose controls will continue to meet the Ginna Station licensing basis with respect to the requirements of GDC-19; 10CFR20; 10CFR50, Appendix I; 40CFR190 and NUREG-0737, II.B.2. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to radiation protection and ensuring that occupational and public radiation exposures will be maintained ALARA.

Table 2.10.1-1
EPU VITAL AREA ACCESS DOSE SUMMARY

Area	Location	Time After Accident	Occupancy	Dose In Area	Access Dose To and From (rem)	EPU Total
A	Hydrogen Recombiner Control Panel	Note 4				
B-F	Post Accident Sampling Access Requirements	Note 5				
G	Radwaste Control Panel	10 day	2 min	2.7rem	Negl.	2.7 rem
H	Safeguards Bus 16	1 day	----	3.3R/hr	Negl.	3.3 R/hr
I	Safeguards Bus 14	1 day	----	0.1R/hr	Negl.	0.1 R/hr
J	Post Accident Sampling Access Requirement	Note 5				
K, L, & M	HVAC Filter Change-Out (Aux. Bldg., Spent Fuel Pool, Control Room)	Note 5				
Note 3	Throttle SW to CCW HX	10 mins	10 mins	0.9rem	0.6 rem	1.5rem

<u>Notes</u>

1. Areas H and I consider Access at 1 day and beyond. Area G is based on Access at least 10 days after the accident.
2. The 1 day post LOCA EPU scaling factor is 1.38. The 10 day post-Accident EPU scaling factor is 1.31.
3. Access requirement identified subsequent to issuance of the 1979 Design Review Report
4. Requirement to maintain hydrogen recombiners removed.
5. Short-term post-accident access is not required or anticipated. Long-term sampling activities or filter changes would encounter greatly reduced dose rates, and employ dose reduction efforts such as temporary shielding as necessary based on existing conditions.

TABLE 2.10.1-2 Estimated Annual EPU Doses to the Public due to Normal Operation Gaseous and Liquid Radwaste Effluents				
Type of Dose	**Appendix I Design Objectives**	**Base Case 100% Capacity Pre-EPU case**	**Scaled Doses (EPU Case)**	**Percentage of Appendix I Design Objectives for EPU Case**
Liquid Effluents				
Dose to total body from all pathways	3 mrem/yr	3.16E-03 mrem/yr	3.77E-3 mrem/yr	0.126%
Dose to any organ from all pathways	10 mrem/yr	3.37E-3 mrem/yr	4.01E-3 mrem/yr	0.040%
Gaseous Effluents				
Gamma Dose in Air	10 mrad/yr	Included with the value reported for the total body of an individual in the Annual Radioactive Release Report	19% increase	Bounded by the percentage of the Appendix I Design Objective reported for the EPU Case for the total body of an individual
Beta Dose in Air	20 mrad/yr	Included with the value reported for the skin of an individual in the Annual Radioactive Release Report	19% increase	Bounded by the percentage of the Appendix I Design Objective reported for the EPU Case for the skin of an individual
Dose to total body of an individual	5 mrem/yr	7.06E-03 mrem/yr	8.41E-03 mrem/yr	0.168%
Dose to skin of an individual	15 mrem/yr	9.25E-03 mrem/yr	1.10E-02 mrem/yr	0.073%
Radioiodines and Particulates Released to the Atmosphere				
Dose to any organ from all pathways	15 mrem/yr	1.76E-02 mrem/yr	2.27E-02 mrem/yr	0.151%

2.11 Human Factors

2.11.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The Ginna Nuclear Power Plant, LLC's (Ginna) staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The Ginna staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU.

The NRC's acceptance criteria for human factors are based on GDC-19, 10CFR50.120, 10CFR55, and the guidance in GL 82-33. Specific review criteria are contained in SRP sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

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 the Ginna Station design relative to conformance to:

GDC -19 is described in UFSAR section 3.1.2.2.10, Control Room. As described in this UFSAR section, the Ginna Station control room contains the controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions. It is capable of continuous occupancy under all operating and accident conditions, within specified dose limits. In the unlikely event the control room should become inaccessible; provisions have been made so that operators can shutdown and maintain the plant in a safe condition by means of controls located outside the control room.

10CFR50.120 is described in UFSAR section 13.1.3.1, Qualifications of Plant Staff, and section 13.2.2.2, Replacement and Retraining of Unlicensed Personnel. As described in these UFSAR sections, the qualifications and training of the indicated personnel who provide offsite and onsite support to Ginna Station is based on a systems approach to training and is in accordance with 10CFR50.120.

10CFR55 is described in UFSAR sections 13.1.3.1, Qualifications of Plant Staff, and

13.2.2.1, Licensed Operator Replacement and Requalification Training. As discussed in these UFSAR sections, licensed operators are qualified in accordance with 10CFR55. Their training is based on a systems approach to training. The program was accredited by the National Nuclear Accrediting Board in February 1987.

NRC Generic Letter (GL) 82-33, Supplement 1 to NUREG-0737 Requirements for Emergency Response Capability, is discussed in UFSAR section 13.5.1.2, Preparation of Procedures. As discussed in this UFSAR section, the Administrative Controls section of Ginna Station's Technical Specifications require that written procedures be established, implemented, and maintained for activities which include those addressed in GL 82-33.

In addition to the UFSAR sections detailed above, Human Performance and Human Factors related elements are discussed in UFSAR sections 7.5.1.2, Control Room; 7.7.6.3, Plant Process Computer and Safety Assessment System and 15.0.9.2, Interpretation of Operator Action Times.

2.11.1.1 Technical Evaluation

Introduction

Human Factors Engineering and Human Performance initiatives are foundational characteristics that help ensure that the plant operators can effectively and safely operate the facility as well as mitigate emergency conditions. When initiating a plant change, the Change Impact Evaluation (CIE) prompts completion of a Human Factors review checklist for changes that may impact the Control Room layout (alarms, indication, appearance or performance). In addition, plant operations staff has been represented and participated in EPU planning and modification development studies. Operating Experience in the form of lessons learned from prior major modifications (e.g. Steam Generator replacement with reduced Tave, implementation of Improved Standard Technical Specifications and implementation of Westinghouse ERG based Emergency Operating Procedures) has provided valuable insights with respect to the human factors issues associated with major changes in operating methods. To ensure that changes associated with EPU do not introduce any unanticipated consequences a careful review of the effects of those changes on Human Performance was performed using a standard set of questions developed by the NRC for the review of human factors. These standard NRC questions are further discussed in this LR section.

Description of Analysis and Evaluations

The NRC has developed a standard set of questions for the review of the human factors area. The Ginna staff has addressed these questions. The following are the NRC staff's questions and the Ginna staff's responses.

1. <u>Changes in Emergency and Abnormal Operating Procedures</u>

Describe how the proposed EPU will change the plant emergency and abnormal operating procedures.

Ginna Response:

The existing Emergency and Abnormal procedure set will continue to provide adequate guidance to cover the spectrum of anticipated events. The following procedure changes are intended to enhance operator response times and to incorporate physical plant changes resulting from EPU. In addition to the more significant items listed below, minor changes (typically setpoints) have been identified for several Emergency, Abnormal and other Operating procedures.

Changes in Emergency and Abnormal Procedures:

a. To ensure meeting assumed Operator action timelines for specific Accident Analysis Scenarios, the E-0 automatic action verification steps are being streamlined to expedite diagnosis and plant stabilization. This is consistent with a recent Westinghouse Owners Group initiative.
b. Functional Restoration (FR) procedure FR-H.1 will be changed to provide earlier initiation of Standby Auxiliary Feedwater System to mitigate consequences of a high energy line break in the Intermediate Building resulting in loss of all normal Auxiliary Feedwater Pumps. (reduced time to dryout issue)
c. Standby Auxiliary Feed flow requirements will increase for certain events. This will be reflected in appropriate procedures.
d. The E-0 Main Feedwater isolation step is being changed to incorporate installation of the new Main Feed Isolation Valves (MFIVs). This will reduce the time required to complete the MFW isolation verification step. (Main Feedwater Regulating Valves will no longer be required to be manually closed before SI reset)
e. The offnormal procedures for response to Appendix R fire scenarios will be revised to enhance operator response times. The EPU power level will result in an increase in decay heat following trip, reducing time to S/G dryout if feed is not restored. Additionally, changes to the RCS Tavg and pressurizer (PRZR) level program will reduce the time available for restoration of charging flow.
f. Guidance will be added to appropriate Appendix R procedures to initiate SI injection if charging is not adequate to restore PRZR level. This will provide some additional flexibility and risk reduction since, currently, only the "A" charging pump is Appendix R protected.
g. Appendix R procedures will be modified to provide a contingency to ensure capability to cooldown the pressurizer at a rate adequate to support the water solid S/G cooldown method when RHR is not available.

h. Some minor modifications being considered for Appendix R local operating stations will result in reducing plant risk and operator response times (backup air supply for charging pump speed control, relocation of DC control power transfer switch for the "A" charging pump, local control of Turbine Driven AFW pump discharge valve, MOV-3996).
i. The Emergency Plan may require minor modifications for account for additional decay heat, potential source term changes, and verification of severe accident management guideline (SAMG) effectiveness.
j. Setpoints related to changing BOP and generation parameters and increased decay heat will be reviewed and revised throughout the Emergency, Abnormal and Operating procedure sets.

Conclusion:

The anticipated changes to the Emergency/Abnormal procedures do not alter basic mitigation strategies and will be adequately implemented by the normal procedure change process and operator training program.

2. Changes to Operator Actions Sensitive to Power Uprate

Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU.

Identify and describe operator actions that will involve additional response time or will have reduced time available. The response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes.

Ginna Response:

Changes to operator actions sensitive to power uprate include the following:

a. The time allowed for concurrent initiation of hot and cold leg recirculation to minimize boron precipitation for large LOCA will be reduced.
b. Reduction in PRZR level no-load setpoint will require increased emphasis on RCS temperature stabilization post trip to prevent letdown isolation.
c. HP turbine replacement may affect turbine startup process.
d. For Appendix R scenarios requiring water solid cool-down, the increased decay heat associated with EPU will require installation of 2 spool pieces from the steam header to the blowdown tank. (pre-EPU, only 1 required).
e. As a result of reduced time to S/G dryout (~35 minutes at uprate) procedure guidance for initiation of Standby Auxiliary Feedwater will change to ensure adequate heat sink for events where restoration of feed is delayed.
f. In conjunction with EPU, Relaxed Axial Offset Control (RAOC) will be implemented. This will alter the requirements for control of axial core power.

Conclusion:

There have previously been significant programmatic changes to Operator actions such as implementation of symptom based Emergency Operating Procedures and changes related to steam generator replacement and implementation of Improved Technical Specifications, all of which were successfully accomplished using the normal plant change/training processes. The changes in Operator actions related to EPU are less significant, and established change processes will provide an adequate implementation strategy. The changes do not significantly impact normal Operator actions or off-normal event mitigation strategies. The changes will be appropriately proceduralized and the Operators will receive formal classroom and simulator training for their implementation.

3. <u>Changes to Control Room Controls, Displays and Alarms</u>

Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably.

Ginna Response:

Changes to Control Room Controls and Displays will not be extensive and will generally include controls for two valves and expanding scales for identified instrumentation. There will also be changes to several control board and computer alarms and limited changes to plant control systems.

Below is a summary of the significant changes identified:

a. The following instrument loops are affected by EPU (calibration range, scaling or transmitter changes):

  - MFW flow scale
  - Main Steam flow scale
  - MFW pump suction flow
  - SAFW pump flow
  - First stage pressure range
  - RCS ΔT setpoint changes

b. Several Alarm Response (AR) procedures will require revision as a result of setpoint changes and changes in plant response to transients:

  - MFW pump NPSH setpoint, condensate bypass valve
  - SF/FF high flow alarms
  - SAFW flow alarms
  - AMSAC alarm inputs
  - condensate pump low pressure alarm

- condensate storage tank minimum level setpoint changes
- MFW pump low suction pressure opening condensate bypass valve

c. Some PPCS computer setpoints will be changed and new points will be added for the following parameters:

   - MFIV air accumulator pressure alarms
   - MFW and Main Steam system alarms
   - RCS delta-T alarm and protection
   - RCS Tavg
   - PRZR level
   - First stage pressure
   - other various alarm changes

d. Changes to controls and control systems:

   - MFIV switches and indicating lights will be added to the Control Board
   - Steam Dump deadband and modulating setpoints
   - Control rod speed program (power mismatch) in AUTO
   - Condensate pump auto start setpoint
   - Condensate heater bypass opening setpoints (MFWP suction press & NPSH)
   - Time delay for condensate bypass valve opening
   - PRZR level program
   - RCS Tavg program
   - Rod Bank sequencing program (Possibly unaffected)

e. There are no planned changes of analog to digital displays or controls. New data acquisition may be accomplished using digital technology.

f. There is minimal application of zone banding on the Control Board. The EPU will not impact any of the zone bands currently identified on the instrumentation.

Conclusion:

The operators will be provided detailed training related to the EPU modifications and resulting control board and procedure changes. Operators are provided station modification review packages as well as classroom and simulator training where appropriate. The initial plant startup of the uprated plant will be implemented as a SIPE (Significant Infrequently Performed Evolution) and will be controlled by the Power Ascension Testing Plan.

4. <u>Changes on the Safety Parameter Display System</u>

Describe any changes to the safety parameter display system resulting from the proposed EPU. How will the operators know of the changes?

Ginna Response:

Changes to the Safety Parameter Display System parameters not discussed previously:

- RCS subcooling margin will be reduced
- Condensate Storage Tank minimum level will increase
- Critical Safety Function status trees will be reviewed and revised as necessary for related changes to setpoints and decision points

Conclusion:

These changes will be addressed by the normal processes, Operations involvement in the modification process, procedure change reviews and operator training program modification training.

5. <u>Changes to the Operator Training Program and the Control Room Simulator</u>

Describe any changes to the operator training program and the plant referenced control room simulator resulting from the proposed EPU, and provide the implementation schedule for making the changes.

Ginna Response

The existing Licensed/Non-Licensed Operator training programs employ the Systematic Approach to Training (SAT) process which has provisions for ensuring that adequate training is provided for significant plant modifications prior to implementation. Training will focus on Technical Specification changes, procedure changes and EPU modifications. Training will be initiated during training cycle 2005-5, (8/1/05). Training for the 3 remaining cycles this year will focus on a general overview of the Uprate modifications and then training on specific topics such as Relaxed Axial Offset Control (RAOC), the new HP turbine and other topics. Portions of training cycles 2006-01 through 2006-04 will focus on the overall Plant Uprate. Comprehensive training of the entire modification scope will begin during cycle 2006-05 and will include classroom and simulator training and testing on the EPU modifications. The operators will be able to demonstrate understanding of the integrated plant response on the simulator. Additional Just In Time (JIT) startup training will be provided to the Operators during the 2006 Refueling Outage prior to the Uprate plant initial startup. This JIT training will also cover the startup testing plan both in classroom and on the simulator as necessary.

Plant uprate modifications will be reviewed to determine impact on the simulator. Changes to the simulator modeling will be made to a separate simulator load, on a schedule established to meet the operator training program requirements. The simulator load for current plant configuration will remain unchanged and available for operator training. Status of the simulator configuration will be controlled through the established training process. The Control Board hardware changes, addition of the Main Feed Isolation Valves and associated indications and replacement of indications with revised scaling, will also be scheduled to accommodate the training program requirements.

Additionally, some operators will be involved in the continuing modification review process, providing operational input and gaining knowledge of the required plant changes. Many of the procedural changes especially to the Emergency/Abnormal Operating Procedures and other offnormal procedures will be reviewed and validated by Operations personnel. This will be another process for exposing additional operators to the EPU changes and related bases. These activities will help provide a solid foundation for operator understanding and interaction during the formal EPU training sessions.

Conclusion:

EPU results in a significant number of plant modifications which will generate changes to Technical Specifications, Operations, Maintenance and Testing procedures, training simulator and training lesson plans. The Ginna Systematic Approach to Training (SAT) process has, in the past, been extremely effective in training plant personnel on significant changes including Steam Generator Replacement and associated modifications and implementation of Improved Technical Specifications. Both of these major upgrades were successfully implemented with few issues of concern identified. Training for implementation of the EPU modifications will be accomplished in accordance with this proven process.

Results

The results of the EPU Human Factors review show that changes to plant procedures, when prepared in accordance with the current procedure change control process, will not alter the basic mitigation strategies with which the operators are familiar. Changes associated with instrument scaling and setpoints will not introduce a level of complexity that would lead to misunderstanding the parameter. Operator training will provide effective reinforcement of procedure and plant physical changes as well as build proficiency with the required operator action changes.

2.11.1.2 Conclusion

The Ginna staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that Ginna has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The Ginna staff further concludes that Ginna will continue to meet the Ginna current licensing basis with respect to the requirements of GDC-19, 10CFR50.120, and 10CFR55 following implementation of the proposed EPU. Therefore, the Ginna staff finds the proposed EPU acceptable with respect to the human factors aspects of the required system changes.

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

2.12.1.1 Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The Ginna Nuclear Power Plant, LLC (Ginna) review included an evaluation of:

- plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance,
- transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and
- the test program's conformance with applicable regulations.

The NRC's acceptance criteria for the proposed EPU test program are based on 10CFR50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Ginna Current Licensing Basis

The initial startup test program at the R. E. Ginna Nuclear Power Plant is described in UFSAR section 14.0. The test program was performed in order to ensure the safe and efficient operation of the plant up to its initial rating of 1300 MWt. The reactor was shown to be stable at all power levels up to 1300 MWt with induced disturbances to the reactor system. Perturbations to the secondary system were 10% load swings, 50% load reductions, and 100% turbine trip. Control rods were used for a dynamic rod drop test, ejected and dropped rod worth measurements, and a xenon oscillation test. Maximum and/or minimum values of critical reactor system parameters during plant transient tests were within allowable limits. In addition, core thermal-hydraulic limits were not exceeded for steady-state or transient situations.

The startup and power testing program results substantiated design predictions. The core thermal and hydraulic performance showed that the core operated within the specified thermal and hydraulic limits. Reactor system stability measurements were within applicable criteria. Control rod reactivity worth measurements and rod insertion scram times were satisfactory.

The results of the preoperational testing program and the operational and transient tests for operation up to 1300 MWt were reported to the NRC in the Technical Supplement Accompanying Application to Increase Power, February 1971. The staff reviewed and reported on these results in the Safety Evaluation issued by letter dated January 20, 1972.

An amendment to the operating license was issued on March 1, 1972, which authorized an increase in the plant output from 1300 to 1520 MWt. A diverse and thorough testing program was used in the power escalation performed from March 8 to April 14, 1972.

The program consisted of a number of tests and measurements at power levels of 1300, 1380, 1455, and 1520 MWt. At each of these power levels, in-core flux maps, delta T measurements, containment radiation surveys, and primary coolant activity measurements were performed. Additional flux maps were obtained at 1455 MWt to calibrate the axial offset monitoring. The flux maps, delta T measurements, and the containment radiation surveys all showed very good agreement with predictions.

The response of system components to increases in core power output was studied. The reactor was operated for a short period at 1520 MWt and performed satisfactorily. Core physics parameters agreed well with design data and there was considerable margin to core safety limits. Core instrumentation continued to accurately reflect the behavior of the core.

A detailed discussion of the uprating test program is included in UFSAR section 14.6.2. Rochester Gas and Electric Corporation reported the results of the test program to the AEC in a letter dated August 14, 1972.

2.12.1.2 Technical Evaluation

2.12.1.2.1. Introduction

Ginna is currently proposing an Extended Power Uprate (EPU) to increase core thermal power to 1775 MWt. This uprate involves changes to the plant configuration to accommodate the higher reactor power limit as well as the larger steam and feedwater flows commensurate with the power increase. As a result of these changes, testing is required to ensure that the plant can be operated safely in its uprated condition.

2.12.1.2.2. Background

The proposed EPU at Ginna Station will result in the reactor operating at a new core thermal power of 1775 MWt. The current licensed core thermal power is 1520 MWt. Ginna has significant operating experience, over 33 years, at its current operating condition. Ginna is a Westinghouse two-loop design. The proposed EPU power level has been successfully achieved by a similar Westinghouse two-loop design plant, Kewaunee, with no adverse affects.

In a PWR, the biggest change in system operating parameters occurs in the secondary side where mass flow is increased commensurate with the uprate. Minor changes also occur in primary side temperatures to provide additional heat transfer in the steam generators. At Ginna, the main steam and condensate/feedwater flows will increase by approximately 17%, but the main steam system pressure will be similar to current operation due to increasing reactor coolant T_{avg}.

In order to accommodate this new thermal power, changes in plant operating parameters have to occur. However, it has been found that the fundamental operating characteristics of an uprated plant remain consistent with the operating characteristics prior to the uprate, and also consistent with other similar units that have been uprated. This means that pre-uprate plant operating experience and industry operating experience provide valuable insight to the viability of a plant uprate. This operating experience has been incorporated into the proposed test plan.

Several plant modifications are required to support power operation at the proposed uprated core thermal power. Post-modification testing of these modifications will be performed to ensure proper installation. A list of the significant plant modifications and the post-modification testing currently planned for these modifications is provided in Table 2.12-5. Additionally, system surveillance tests will be performed as required to verify that the modifications meet applicable performance criteria. Integrated plant analyses were performed to define the performance criteria of the various plant modifications necessary to accommodate the uprated power. The results of these analyses are used, in part, in lieu of large transient testing to verify that the plant systems are capable of performing safely in the uprated condition.

The EPU testing program will also draw on the results of the original startup and test program and applicable industry experience as a means of ensuring safe operation at the new core thermal power level. Comparison will be made between this original data, recent operating data and the data that will be gathered during the uprate testing to ensure that the results are reasonable. Additionally, Ginna has over 33 years of operating experience at the current licensed power level such that system interactions are all well known. Kewaunee has uprated to a core thermal power (1772 MWt) that is nearly identical to the Ginna EPU power level (1775 MWt) and has operated successfully at the new power. Ginna has established and maintained close communication with Kewaunee throughout the EPU project in order to benefit from their power uprate experience.

In addition to Kewaunee, Ginna has benefited from industry operating experience in power uprate implementation from several industry sources, including INPO. Ginna developed an industry operating experience database pertaining to power uprate and has used this database in the formulation of plans for system inspections during the 2005 refueling outage, design of EPU modifications, determination of control system settings and setpoints, and development of post-modification and power ascension test plans. For example, Ginna has learned valuable lessons from the industry regarding vibration and vibration monitoring, iso-phase bus duct cooling and air flow, turbine controls, feed/condensate/drain system flows and pressure drops, feedwater heater performance and reactor control system setpoints.

Finally, several transient and load change tests are proposed to ensure that the system dynamic behavior is satisfactory. This testing will also ensure that no new thermal hydraulic phenomena or adverse system interactions are created by the proposed uprate. Additional large plant transient tests are being considered for verification of integrated plant performance.

In summary, the proposed EPU testing program is comprised of a mixture of power ascension monitoring, post-modification testing, analytical evaluation, and transient testing, to ensure that the plant can operate safely at its new uprated core thermal power. The following sections describe the proposed Ginna Power Ascension Testing Program and clearly demonstrate that the proposed testing program contains all of the necessary elements to assure safe operation at the uprated power level.

2.12.1.2.3. Proposed Power Ascension Test Plan

2.12.1.2.3.1. General Discussion

The development of the power uprate test program is based on review of similar test programs performed at other plants, the outputs of various system and integrated plant analyses performed in support of the EPU, and the Ginna Technical Supplement Accompanying Application to Increase Power, dated February 1971. Additionally, Chapter 14, section 14.6.2 of the Ginna FSAR, describing the test methodology used during power ascension from 1300 to 1520 MWt was also reviewed.

Prior to the commencement of power ascension testing, the Test Program will require the completion of numerous activities, which include:

- Review and revision of applicable plant operating procedures, administrative procedures, surveillance test procedures, calibration procedures, chemical and radiological procedures and other similar procedures.
- Review and revision of computer software programs have been as required to support the power uprate test program and the new EPU power level.
- Incorporation of applicable plant instrumentation setpoint changes and recalibration of instrumentation as required.
- Implementation and successful post-modification testing of all required plant modifications.
- Review of Temporary Modification logs and GL91-18 (Operable but Degraded) conditions to assure there is no impact on the ability of the effected equipment to support uprate, and that uprate will not have an adverse impact on any existing plant condition.

Additionally, commitments which were the result of the EPU Updated Safety Analysis Report, EPU License Amendment, the NRC EPU Safety Evaluation Reports (SER), and any other actions associated with the Ginna EPU implementation, will be verified as being closed, included in the Power Ascension Testing Program, or evaluated as not impacting power ascension.

The EPU Power Ascension Test Program will be developed to verify the following:

- Plant systems and equipment affected by EPU are operating within design limits.
- Nuclear fuel thermal limits are maintained within expected margins and the core is operating as designed.
- Steam generator water level control is stable with adequate control margin to allow for anticipated transients.
- Reactor control systems are stable and within acceptable limits.
- MSR and feedwater heater drains and level control are stable.
- System radiation levels are acceptable and stable.
- General area and local environmental conditions are acceptable.

2.12.1.2.3.2. EPU Power Ascension Test Plan and Test Conditions

Performance in accordance with expectations based upon analyses and operating experience of similar equipment will be established. Acceptance criteria will be established for each plant parameter determined to be included in the "monitored parameter list". This list will be populated by utilizing industry operating experience as well as consultation with Ginna plant engineering personnel, and industry experts at vendors with significant power uprate testing experience.

During the EPU start-up, power will be increased in a slow and deliberate manner, stopping at pre-determined power levels for steady-state data gathering and formal parameter evaluation. These pre-determined power levels are referred to as Test Conditions. The typical post-refueling power plateaus will be used until the current (pre-EPU) full power condition is attained at approximately 85% of the EPU power level (1520 MWt), with additional equipment and plant transient testing performed at 25% and 50% of the EPU power level as discussed later to verify expected component, system and integrated plant performance. A summary of the Power Ascension Test Plan is provided in Table 2.12-1.

Prior to exceeding the current licensed core thermal power of 1520 MWt, the steady-state data gathered at the pre-determined power plateaus and transient data gathered during the specified transient tests at low power, as well as observations of the slow, but dynamic power increases between the power plateaus, will allow verification of the performance of the EPU modifications. In particular, by comparison of the plant data with pre-determined acceptance criteria, the test plan will provide assurance that unintended interactions between the various modifications have not occurred such that integrated plant performance is adversely affected.

Once at approximately 85% of EPU power (1520 MWt), power will be slowly and deliberately increased through 5 additional Test Conditions, each differing by approximately 3% of the EPU rated thermal power. Again, both dynamic performance during the ascension and steady-state performance for each Test Condition will be monitored, documented and evaluated against pre-determined acceptance criteria.

Following each increase in power level, test data will be evaluated against its performance acceptance criteria (i.e., design predictions or limits). If the test data satisfies the acceptance criteria then system and component performance will be considered to have complied with their design requirements.

In addition to the steady-state parameter data gathered and evaluated at each test condition, and the dynamic parameter response data gathered and evaluated during the ascension between test conditions, several transient tests will also be performed. These tests are listed and described in Section 2.12.1.2.3 and tabulated below. These transient tests will provide additional confidence in the validity of the analytical models and assumptions used in the analysis of plant modifications and integrated plant response to transients. Transient test data will be compared against predictions provided by the same analytical models used in design verification for EPU. Any significant differences between predictions and test data will be evaluated and reconciled before proceeding with the power ascension.

Specifically, hydraulic interactions between the new feed pumps and modified feed regulating valves, as well as the impact of the higher main feed flow and the associated increased piping pressure loss will be evaluated. Individual control systems such as steam generator level control and moisture separator and feedwater heater drain level control will be optimized for the new conditions as required. It is anticipated that the proposed transient tests will adequately identify any unanticipated adverse system interactions and allow them to be corrected in a timely fashion prior to full power operation at the uprated conditions.

Table 2.12-1 provides a summary of the Power Ascension Test Plan.

Table 2.12-1
Ginna Extended Power Uprate Power Ascension Test Plan

Test / Modification	Test Description	Prior To Start up	Rated Thermal Power - % of 1775 MWt (Allowance +0%, -5%)																		(Allowance +0%, -1%)				
			0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	88	91	94	97	100
Nuclear Steam Supply System (NSSS) Data Record	Data Collection						X		X				X					X		X	X	X	X	X	X
Balance of Plant Data Record	Data Collection						X		X				X					X		X	X	X	X	X	X
Transient Data Record	Data Collection						X		X				X					X		X	X	X	X	X	X
Core Map	Power Distribution and COLR Parameters								X				X					X		X	X	X	X	X	X
NSSS Calorimetric and Power Range Channel Adjustment	Verify Thermal Power and Adjust Nuclear Instrumentation						X		X				X					X		X	X	X	X	X	X
Reactor Coolant System (RCS) Flow Measurement	Elbow Tap ΔP and/or Calorimetric	X																		X					X
Incore-Excore Axial Offset Calibrations	Calibrate Excore Instrumentation to Incore Axial Offset												X[1]							X[2]					
Load Changes	10% Ramp to								X																X

[1] Incore flux map for data acquisition will be performed at 50% of 1775 MWt but actual channel calibrations will be performed during subsequent power ascension.
[2] Incore flux map for data acquisition will be performed at 85% of 1775 MWt and channel calibrations will be completed prior to exceeding 90% of 1775 MWt in accordance with Ginna Technical Specification SR 3.3.1.6.

Table 2.12-1
Ginna Extended Power Uprate Power Ascension Test Plan

Test / Modification	Test Description	Prior To Start up	Rated Thermal Power - % of 1775 MWt (Allowance +0%, -5%)																	(Allowance +0%, -1%)				
			0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	88	91	94	97	100
	Verify System Response																								
Turbine Trip	OST Turbine to Verify System Response						X																		
Turbine Stop Valve, Governor Valve, and Intercept Valve Testing.	Standard turbine valve tests w/post-modification tests												X												
Steam Generator Level Feedwater Flow Dynamic Testing	Manually inserted level setpoint step-changes in the steam generator.								X																X
Vibration Monitoring	Monitor vibration in Plant Piping and Rotating Equipment								X				X							X	X	X	X	X	X
Plant Radiation Surveys	Verify Expected Dose Rates																			X					X
Plant Temperature Surveys	Verify Expected Temperatures																			X					X

2.12.1.2.3.3. Acceptance Criteria

The acceptance criteria for the Ginna power ascension test program will be established as discussed in Regulatory Guide 1.68. Level 1 acceptance criteria are values for process parameters assigned in the design of the plant that are safety significant. If a Level 1 criterion is not satisfied, the power ascension will be stopped and the plant will be placed in a condition that is judged to be safe based upon prior testing. The power escalation test procedure and Technical Specifications will provide direction for actions to be taken to assure the plant is safe and stable. Resolution of the issue that resulted in exceeding the Level 1 criterion must be resolved by equipment changes or through engineering evaluation, as appropriate. Following resolution, the applicable test portion must be repeated to verify that the Level 1 requirement is satisfied. A description of the problem must be included in the report documenting successful completion of the test.

Level 2 acceptance criteria are values that relate to plant functions or parameters that are not safety significant. If Level 2 criteria are not met, the Power Ascension Test Plan may continue. Investigation of the issue that resulted in exceeding the Level 2 criterion may continue in parallel with the power escalation. These investigations would be handled by existing plant processes and procedures.

For the Ginna Power Ascension Test Plan specific Level 1 & 2 acceptance criteria will be established and incorporated into the Power Ascension Test Procedure.

2.12.1.2.3.4. Vibration Monitoring

A Piping and Equipment Vibration Monitoring Program will be established to ensure that any steady state flow induced piping vibrations following EPU implementation are not detrimental to the plant, piping, pipe supports or connected equipment.

Observed piping vibrations will be evaluated to ensure that damage will not result. The predominant way of assessing these vibrations is to monitor the piping during the plant heat up and power ascension. The methodology to be used for monitoring and evaluating this vibration will be in accordance with ASME OMa-S/G-2004.

The scope of the Piping and Equipment Vibration Monitoring Program includes any accessible lines that will experience an increase in their process flow rates. Any branch lines attached to these lines (experiencing increased process flows) will also be monitored as experience has shown that branch lines are susceptible to vibration-induced damage. The scope of the Piping and Equipment Vibration Monitoring Program includes the following systems:

- Main Steam, including Reheater Inlet
- Feedwater
- Condensate
- Extraction Steam
- Heater Drains
- Moisture Separator Drains

- Turbine Gland Steam and Drains
- Pumps
- Motors
- Main Turbine

The program scope will also include any lines or equipment within the monitored systems that have been modified or otherwise identified through the Ginna action report system as having already experienced vibration issues.

The piping and equipment within the scope of the vibration monitoring program will be observed at several different plant operating conditions. The first observations will be conducted prior to the shutdown in which the EPU will be implemented. These initial observations and the observations at 30%, 50% and 85% EPU rated power while in the Power Ascension Test Procedure will establish the baseline piping vibration level for further comparison.

Subsequent observations will take place at each EPU Test Condition, as described in Section 2.12.1.2.3.1 above. By comparing the observed pipe vibrations / displacements at various power levels with previously established Level 1 and Level 2 acceptance Criteria, potentially adverse pipe vibrations will be identified, evaluated and resolved prior to failure.

2.12.1.2.3.5. Transient and Dynamic Tests

The following transient or dynamic testing is proposed to support the Ginna power uprate. These tests are proposed to verify that no new adverse system interactions or thermal hydraulic phenomena have been introduced to plant systems as a result of the EPU or the associated modifications. Additional large plant transient tests are being considered for verification of integrated plant performance.

Turbine Overspeed Trip from 20% EPU Power

Once a steady state power of approximately 30% EPU rated power has been reached, a planned turbine overspeed trip will be performed. Power will be held near 30% with the main generator synchronized for approximately 8 hours to ensure that turbine metal temperatures are above the Fracture Appearance Transition Temperature (FATT). Reactor power will then be reduced to approximately 20% of EPU power and the generator taken off line. With the reactor at approximately 20% of EPU power and the majority of that power being routed to the main condenser via the steam dump valves, the turbine will be accelerated until its speed causes an actuation of the mechanical overspeed protection system. This test will verify proper performance of the overspeed setpoint and mechanism. It will also verify proper operation of the turbine valves, including the new turbine governor valves, and verify expected plant performance subsequent to the turbine trip. Performance of plant control systems such as steam dumps, and pressurizer pressure and level control will be monitored in response to the transient.

10% Load Change at 30% EPU Power

Once a steady-state power of 30% EPU rated power has been reached, power will be decreased at 1% per minute from 30% to 20% EPU power. The dynamic behavior of the various plant control systems will be observed and evaluated against Level 1 and Level 2 acceptance criteria to ensure that the combination of increased power and changes to the plant configuration (EPU modifications) have not resulted in an unacceptable aggregate impact. Once acceptable dynamic performance has been verified and documented, power will be ramped back to 30% at 1% per minute. Again, plant response will be observed during the power increase and evaluated against the appropriate Level 1 and Level 2 acceptance criteria.

10% Load Change at 100% EPU Power

Once a steady-state power of 100% EPU rated power has been maintained for 72 hours in order to meet fuel pre-conditioning requirements, and plant performance data has been gathered and reviewed, power will be decreased at 1% per minute from 100% to 90% power. The dynamic behavior of the various plant control systems will be observed and evaluated against Level 1 and Level 2 acceptance criteria to ensure that the combination of increased power and changes to the plant configuration (EPU modifications) have not resulted in an unacceptable aggregate impact. Once acceptable dynamic performance has been verified and documented, power will be ramped back to 100% EPU power at 1% per minute. Again, plant responses will be observed and evaluated against the appropriate Level 1 and Level 2 acceptance criteria.

Turbine Stop Valve, Governor Valve, and Intercept Valve Testing

During the power ascension at approximately 50% of EPU rated power, the station turbine valve test procedure will be performed. This test will validate dynamic performance of new governor valve design, which was modified for EPU, to ensure adequate transient response. It will also verify acceptable dynamic performance of the new HP turbine rotor during changes in individual arc steam flows.

Steam Generator Level / Feedwater Flow Dynamic Testing

During the power ascension at 30% and at 100% EPU rated power the steam generator water level control and feedwater flow control system will be tested by introduction of step changes in level setpoint. Behavior of steam generator level and feedwater flow parameters will be monitored as well as control system outputs to ensure that system operation is stable. Digital feedwater control system deadband, delay, etc., shall be small enough that steady state limit cycles shall not produce significant steam generator narrow range water level variations.

Table 2.12-2 provides a summary of the large plant transient tests that will be incorporated in the Ginna EPU Power Ascension Test Plan.

Table 2.12-2 Large Plant Transient Tests in Ginna EPU Power Ascension Test Plan		
Proposed Test	**Description**	**Expectation**
Turbine Overspeed Trip from 20% EPU Power	The turbine will be taken to approximately 30% power, ramped back to no-load, with the reactor at approximately 20% power, and automatically tripped as speed exceeds the mechanical overspeed trip setpoint unloaded.	This test will verify proper operation of the overspeed mechanism for the new HP turbine, proper operation of the new turbine control valves and exercise the steam dump system.
10% Load Change at new 30% and 100% EPU Power	Fast power ramps limited by station license conditions and fuel pre-conditioning considerations.	These ramps will test NSSS and BOP control system operation, to ensure that no unanticipated aggregate effects have been produced by interaction of the plant modifications.
Turbine Stop Valve, Governor Valve, and Intercept Valve Testing at 50% EPU Power	Standard turbine valve testing augmented by post-modification tests associated with HP Turbine and Governor Valve Replacement.	Validate dynamic performance of new governor valve design to ensure adequate transient response. Verify acceptable dynamic performance of the new HP turbine rotor during changes in individual arc steam flows.
Steam Generator Level / Feedwater Flow Dynamic testing at 30% and 100% EPU Power	Manually inserted level setpoint step-changes in the steam generator. Both up-going and down-going setpoint changes of different magnitudes will be inserted.	Verify SG level control system response and acceptability of over-shoot, damping and steady-state limit cycling at the new licensed power level. Verify acceptable operation of the digital feedwater control system.

2.12.1.2.4. Comparison of Proposed EPU Test Program to the Initial Plant Test Program

The following table (Table 2.12-3) provides a comparison of the original plant start-up testing to the proposed Power Ascension Test Program. The table lists all tests performed during original power ascension regardless of power level at which they were performed. Included in the table are descriptions of the original test, listings of the original power level at which the test was performed, whether the test will be replicated as part of the Power Ascension Test Program, and the justification for why it is not performed (if it is not performed). Table 2.12-4 specifically lists the large plant transient tests performed in the original startup test program and justification if not performed for EPU.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.5.1 **14.6.1.1.1**	**Safety Injection Test** Verify the proper operation of the steam line isolation sequence and the operation of the motor-driven and steam-driven feedwater pumps by their respective safety signals from the A and B logic trains.	NA (vessel internals not installed)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the logic trains for the safety injection system was not changed as a part of this EPU; therefore, the proper sequencing and operation of the safeguard system will perform as originally tested. New SLI setpoints will be verified by calibration tests prior to startup. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 4.5.1 **14.6.1.1.2**	**Accumulator Blowdown Test** This test determined the magnitude of pipe displacement and stress resulting from reaction of the fluid blowdown, the amount of water forced back through the reactor coolant pump and into the low portion of piping between the steam generator and pump suction.	NA (vessel internals not installed)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the reactor pressure will be very similar to the current operating pressures; therefore, the reaction components will be very similar to those obtained during initial testing. Additionally, blowdown of the accumulators is verified by regular surveillance testing.
SU 4.5.1 **14.6.1.1.3**	**Safety Injection Flow Test** This test verified the safety injection pumps and RHR pumps flow and head characteristics.	NA (vessel internals not installed)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the safety injection pumps and the RHR pumps are not impacted by the power uprate. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.5.1 **14.6.1.1.4**	**Containment Spray System Test** The pump for this system could not be tested without flooding the containment; however, the piping and nozzles were air tested and were satisfactory.	NA	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the EPU did not alter this system and it was shown to be satisfactory during start-up testing. The spray ring operation is verified by an air flow test periodically as required by regular surveillance testing.
SU 4.5.1 **14.6.1.1.5**	**Residual Heat Removal System Test** This test demonstrated that the Residual Heat Removal System met all performance and design requirements.	Hot Shutdown (350°F, 415 psig)	Yes	--	This test was originally performed at less than 80% power. Specifically, testing of the RHR system determined that the system met or exceeded its design requirements. It has been shown by analysis that the RHR system capabilities are adequate for the power uprate condition and that the power uprate has no adverse affect on this system. There are no modifications planned to the RHR system. Therefore, this test is not required to be performed at the uprated power conditions. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 9.8.2 **14.6.1.1.6**	<u>Safeguards Systems Operational Test</u> This test ensured that all safeguard systems were operationally checked before criticality. Systems tested were: steam line isolation, safety injection, containment spray, containment isolation, and containment ventilation isolation.	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the safeguards systems valves and motors were not actuated for this test, but rather the actuating devices of the components such as relays, controllers, etc., were monitored for operation. The components associated with the new MFIV will be functionally tested prior to startup. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 9.8.2 **14.6.1.1.7**	<u>Emergency Diesel Generator Test</u> This test verified the air capacity needed to crank the engines for 45 seconds. It also verified that the diesel could be placed on line within 10 seconds.	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the diesel start time, load time, and capacity were validated by this test. These requirements do not change as a result of the power uprate. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 9.8.2 **14.6.1.1.8**	**Direct Current Battery Test** The battery system was tested in two basic ways. First, the battery charger input voltage was varied by 10%, but the output dc voltage did not vary by more than 1%. The second test showed that the battery could sustain a discharge rate of 131 amps for 8 hours while not lowering the output voltage below 105V.	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, negligible new DC loads were added as part of this EPU; therefore, the original test remains adequate and the results bound uprate conditions. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 4.1.16 **14.6.1.2.1**	**Reactor Coolant System Pressure Comparison Test** This test verified the calibration of the primary coolant system pressure instrumentation at actual system pressures.	Cold Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the pressure of the primary coolant system will be the same as the current operating pressures; therefore, the test parameters will remain essentially the same.
SU 4.1.14 **14.6.1.2.2**	**Resistance Temperature Detector Cross Calibration** This procedure was used to determine the isothermal corrections for reactor coolant resistance temperature detectors and in-core thermocouples.	Hot shutdown (0 power)	Yes	--	The EPU will marginally raise the reactor coolant temperature; however, the calibration for the RTDs and thermocouples will be in the existing temperature range. This calibration is performed as part of normal reactor start-up.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.15.1 **14.6.1.2.3**	**<u>Steam Generator Manual Control and Level Instrumentation Test</u>** This was a functional test of the steam generators, condensate system, feedwater system, auxiliary feedwater system, and the instrumentation of these systems. This test stroked all valves in the condensate, feedwater, and auxiliary feedwater systems with final position of the valves in the normal operating position. The condensate and feedwater pumps were started and flow measurements versus feedwater bypass valve position were taken. The automatic start of the preferred auxiliary feedwater pumps was verified by tripping the main feedwater pumps.	Shutdown to Hot Shutdown (0 Power)	Yes	--	This test performed a functional test of the feedwater and condensate system. These systems have been proven to be functional based on previous operating experience. During power ascension, important BOP parameters will be monitored from 0% power through EPU 100% power to verify proper operation of the systems, including the effects of modifications. This is in addition to specific post-modification testing that will verify correct installation of the modifications. The automatic start feature of the auxiliary feedwater pumps is verified periodically by surveillance testing and does not need to be preformed in a special test. Various analyses have been performed to verify that the modified system is capable of supporting operation at the uprated power condition. The flow requirement for preferred AFW does not change as a result of EPU. Standby AFW flow requirements do change and will be verified by pump head testing and analyses. As a result, this test is not required to be performed as part of power ascension testing.
SU 4.11 **14.6.1.2.4**	**<u>Rod Position Indication System Test</u>** Satisfactory performance of the rod position indicating system and for each control rod and each control rod bank.	Hot shutdown (0 power)	Yes	Yes	This test was originally performed at less than 80% power. The power uprate fuel assemblies will result in the rods to sit approximately 3 inches higher than original design. This requires modification of the rod position indication software or placing spacers beneath appropriate coil stacks. These modifications will require testing of the RPI system.

Table 2.12-3
Comparison of Proposed EPU Tests to Original Startup Tests

SU Test # FSAR Section	Test Description	Original Power Ascension Test Power Level	Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)	Test Plan For EPU (yes/no)	Evaluation/Justification
SU 4.10.1 **14.6.1.2.5**	**Rod Stepping Test** This test verified that the rod control systems satisfactorily perform the stepping function for each rod under hot and cold conditions.	Hot shutdown (0 power)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the rod control system has performed its intended function during all phases of plant operation. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 4.10.2 **14.6.1.2.6**	**Rod Cluster Control Assembly Drop Time and Partial Length Rods Operational Tests** This test determined the drop time of each full length rod cluster control assembly under a number of reactor coolant system operating conditions.	Cold to hot shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the parameters of concern for this test are not altered by EPU. Additionally, rod drop time testing will be conducted as required by the current surveillance testing.
SU 4.13.1 **14.6.1.2.7**	**In-Core Thermocouples Test** This test provided a functional checkout and demonstration of the in-core thermocouple readout system.	Hot shutdown (0 power)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. The slight increase in core outlet temperature is within the range of the instrumentation. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the in-core thermocouple readouts are not adversely impacted by the uprate. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.13.2 **14.6.1.2.8**	**Movable In-Core Detector System Test** This test provided a functional demonstration of the in-core flux mapping system.	NA	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the In-Core Detector System is used during normal plant operation and has proven itself to be reliable. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 4.2.7 **14.6.1.2.9**	**Reactor Makeup Blender and Boric Acid Transfer Pumps Operational Test** This test provided information on the operational characteristics of this system.	Hot shutdown (0 power)		No	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, this system has successfully added boron as needed since the plant was started up.
SU 4.2.3 **14.6.1.2.10**	**Pressurizer Level Control Test** This test verified the setpoints of the pressurizer level control system and determined how the system responded to system level and Tavg variation.	Hot shutdown (0 power)	--	No	The level setpoints are changed by the EPU. The new setpoints will be verified by instrument calibration checks prior to startup. In addition, performance of the level control system with changes in power level will be verified during power escalation and transient tests. No additional test of the system is required.
SU 4.1.3 **14.6.1.2.11**	**Pressurizer Pressure Control Test** This test checked the response, stability, and general control characteristics of the pressurizer control system.	Hot shutdown (0 power)	--	No	Pressurizer pressure control system setpoints will not be altered by EPU. Proper operation of the pressure control system will be verified during power escalation and transient tests. No additional test of the system is required.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.9.2 **14.6.1.2.12**	**Steam Dump Test** This test optimized the settings of the steam dump controller. Part of the test could not be performed until mode 1 was achieved.	Hot shutdown (0 power)	--	No	The steam dump setpoints are changed by the EPU. The new setpoints will be verified by instrument calibration checks prior to startup. In addition, performance of the steam dump system will be verified during transient tests. No additional test of the system is required.
SU 4.7 **14.6.1.2.13**	**Radiation Monitoring System Operational Test** This test provided an operational test of the radiation monitoring system.	NA	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Additionally, the operation of these systems is verified by regular surveillance testing.
SU 4.1.15 **14.6.1.2.14**	**Reactor Coolant System Flow Measurement Test** This test provided a means of obtaining the necessary data to inter-relate pump input power, elbow tap pressure, and steam generator delta P as an accurate measurement of flow.	Hot shutdown (0 power)	--	Yes	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the flow rate though the reactor coolant system will change a negligible amount as a result of EPU. However, the test will be performed as part of the EPU test plan in order to validate the calorimetric flow measurement test.
SU 4.8 **14.6.1.2.15**	**Nuclear Instrumentation Test** This test provided a functional demonstration of the nuclear instrumentation system. Each of the 12 drawers (one for each channel) were functionally operated and calibrated by simulating a detector signal to the first element after the detector in a channel.	NA	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, this test provided a functional demonstration of the system only. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.1.4 **14.6.1.3.1**	**<u>Pressurizer Safety Valve Test</u>** This test was performed to verify the proper settings of the Pressurizer Safety Valves.	Hot shutdown (0 power)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the requirements and settings for the pressurizer safety valves do not change as a result of EPU.
SU 4.32.2 **14.6.1.3.2**	**<u>Main Steam Safety Valve Test</u>** This test was performed to verify the proper settings of the Main Steam Safety Valves.	Hot shutdown (0 power)	Yes	--	The setpoints for these relief valves are verified regularly by in-situ testing, similar to the start-up test. The setpoints for the valves are not changed to accommodate power uprate. Additionally, the safety valve flow capability bounds the required flow to protect the system. Therefore, this test is not required for uprate.
SU 4.6.4 **14.6.1.4.1**	**<u>Liquid Waste Concentration Demonstration Test</u>** This test demonstrated the process of drumming concentrated waste from the waste evaporator feed tank. This test also included testing the use of the drums, shields, vacuum pump, and manipulating tools.	NA	--	--	This test was originally performed at less than 80% power. The power uprate impacts these systems by increasing the amount of activity processed through them. However, the basic function of the system is not impacted and the capacity of the system remains acceptable. Thus the EPU does not invalidate the test as originally performed and this test is not required to be performed at the uprated power conditions.
SU4.6.3 **14.6.1.4.2**	**<u>Waste Disposal System Gaseous Waste Test</u>** This test was a functional test of the waste gas system to ensure that the system could adequately process or vent the gaseous waste emanating from the vent header.	NA	--	--	This test was originally performed at less than 80% power. The power uprate impacts these systems by increasing the amount of activity processed through them. However, the basic function of the system is not impacted and the capacity of the system remains acceptable. Thus the EPU does not invalidate the test as originally performed and this test is not required to be performed at the uprated power conditions.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.6.2 **14.6.1.4.3**	**<u>Liquid Waste Processing Test</u>** This test functionally checked the waste disposal system including the waste evaporator, and to demonstrate that the liquid waste disposal system can adequately of the liquid waste in a safe and reliable manner. It should be noted that this system presently operates at 1.5 gpm versus the 2.0 gpm that the system was designed for.	NA	--	--	This test was originally performed at less than 80% power. The power uprate impacts these systems by increasing the amount of activity processed through them. However, the basic function of the system is not impacted and the capacity of the system remains acceptable. Thus the EPU does not invalidate the test as originally performed and this test is not required to be performed at the uprated power conditions.
SU 4.1.7 **14.6.1.5.1**	**<u>Reactor Vessel Internals Measurement Test</u>** The intent of this test was to obtain experimental data on the reactor vessel internals movement during the startup test program.	Hot shutdown (0 power)	--	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the reactor vessel internals are not modified by EPU, and the RCS flow rate is not changing significantly. The new fuel assembly is a standard Westinghouse design that has received flow tests in the factory. Therefore no additional reactor vessel internals testing is required.
SU 4.1.8 **14.6.1.5.2**	**<u>Reactor Coolant System Vibration Test</u>** This test was performed to verify that the vibrations of the reactor coolant pumps and the reactor coolant system piping and equipment were within acceptable limits.	Hot shutdown (0 power)	Yes	--	The reactor coolant pumps will be monitored by installed plant systems for vibration during power ascension to verify that the vibration levels are acceptable. No special vibration monitoring for the RCS is anticipated since there are no changes to RCS flow rate and minor changes in RCS temperatures.

Table 2.12-3
Comparison of Proposed EPU Tests to Original Startup Tests

SU Test # FSAR Section	Test Description	Original Power Ascension Test Power Level	Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)	Test Plan For EPU (yes/no)	Evaluation/Justification
SU 4.1.5 **14.6.1.5.3**	**<u>Preoperational Reactor Coolant System Leakage Test</u>** This test was necessary to satisfy the Technical Specifications that leakage from the reactor coolant system did not exceed 10 gpm from known sources or 1 gpm from unknown sources.	Hot shutdown (0 power)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the RCS pressure is unchanged and the RCS temperature changes slightly as a result of uprate. RCS leakage is monitored during power operation and verified during the normal start-up sequence.
SU 4.1.7 **14.6.1.5.4**	**<u>Reactor Coolant System Thermal Expansion Test</u>** The major objective of this test was to verify that the reactor coolant system could expand unrestrained during the system heatup from the cold condition to operating conditions, and also to establish reference data for the expansion or reactor system components which could be used for future evaluations. This test established basepoint measurements at various points around the reactor coolant system.	Hot shutdown (0 power)	--	--	The RCS temperature increases as a result of EPU, however, the percentage of temperature increase when compared to the magnitude of the RCS temperature is small such that the additional thermal expansion is negligible. Therefore, this test is not required for uprate.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 8.3 **14.6.1.5.5**	**<u>Flow Coastdown Test</u>** In order to make a comparison with the design curve, total core flow was determined by averaging the individual flow loops. The time to 50% flow for the two-loop coastdown was predicted at 12.3 seconds while the plant was found to take more than 13.5 seconds. It was concluded that the plant coastdown rate is consistent and conservative with respect to the FSAR value such that departure from nucleate boiling is prevented. Although the core flow for the one-loop loss of flow fell faster than predicted, the two-loop coastdown is the limiting case and it is in agreement with the FSAR design value.	Hot shutdown (0 power)	--	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the RCS flow is not changed by the EPU, there is no modification to the RCPs and there is no impact to the coastdown times.
SU 8.4 **14.6.1.5.6**	**<u>Natural Circulation Test</u>** This test was conducted to determine if natural circulation occurs and if it is adequate for decay heat removal.	Hot Standby 2% Power	--	--	This test was originally performed at less than 80% power. The adequacy of natural circulation was confirmed to be adequate to remove decay heat at the uprated condition via analyses. Therefore, power uprate has no adverse affect on this parameter and does not invalidate the test as originally performed. Specifically, the slight changes in the RCS temperature will have no impact on the ability to achieve natural circulation so this test remains valid. Therefore, this test is not required to be performed at the uprated power conditions. See additional justification for not performing this test in Section 2.12.1.2.7.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 4.30.3 **14.6.1.6.1**	<u>**Backfeed from the 115 Kilovolt Grid Test**</u> The purpose of this test was to ensure that power can be fed back from the 115 kV grid through the main and auxiliary transformers to the station auxiliaries, in the event of an extended outage of the station auxiliary transformer as specified in the original FSAR.	Shutdown	--	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Specifically, the test conclusively proved that the backfeed could be done and the system is still in this configuration; therefore, the test does not need to be performed. Additionally, load flow analysis has confirmed the capacity of the backfeed from the grid. Therefore, this test is not required to be performed at the uprated power conditions.
SU 4.40.1 **14.6.1.6.2**	<u>**Blackout Test Without Safety Injection**</u> This test was performed to verify the ability of the diesel generators to supply emergency power to the 480 Volt buses in the event that normal outside power is lost.	Shutdown	--	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, all testing needed to prove that the system would perform as needed have been completed successfully and do not need to be repeated for this EPU. EPU modifications have negligible effect on diesel generator loading.
SU 4.32.3 **14.6.1.6.3**	<u>**Main Steam Isolation Valve Test**</u> This test demonstrated that the Main Steam Isolation Valves would close if a high-high containment pressure signal was received. The required valve closure time of 5 seconds was met because they were observed to close in 1 second.	Hot shutdown (0 power)	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, these valves close with flow such that the higher main steam flow at uprated conditions has no adverse impact on this system. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3
Comparison of Proposed EPU Tests to Original Startup Tests

SU Test # FSAR Section	Test Description	Original Power Ascension Test Power Level	Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)	Test Plan For EPU (yes/no)	Evaluation/Justification
SU 4.20 **14.6.1.6.4**	**Fire Service Water Test** This test was functional test of the fire protection system to verify the design criteria of the booster, and diesel-driven and motor-driven fire pumps, and all fire detecting devices.	NA	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the system tests were successful and the EPU will not affect the Fire Service Water System or any of its components or controllers.
SU 9.8.3 **14.6.1.6.5**	**Electrical System Logic Test** The purpose of this test procedure was to specify the operations necessary to operationally test the following systems: Turbine and generator protection, Emergency power system logic, Rod stop, Turbine load reduction	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the logic of the system was not modified as part of the EPU. This test does not need to be performed to verify the logic since this was done during the startup testing of the plant. The actual tripping of circuit breakers, closing of valves, and starting of the diesel generators was not demonstrated in the test, but rather the activating devices, relays, controllers, etc., were monitored with the final action blocked. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU9.8.1 **14.6.1.6.6**	**Reactor Trip System (RTS) Operational Test** This test was to provide the operations necessary to operationally check out the reactor trips of the Reactor trip System (RTS). It was first demonstrated that the reactor trip breakers would open automatically and then for the remainder of the test, the trip breakers were prevented from opening and the devices that actually tripped the breakers were monitored for performance.	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the logic of the Reactor Trip System will not be changed as a part of this EPU and the test does not need to be repeated since the initial testing had satisfactory results. New reactor trip setpoints for EPU will be verified by instrument calibration tests. Additionally, the operation of these systems is verified by regular surveillance testing.
SU1W-2.1 **14.6.1.6.7**	**Reactor Coolant System Hydro Test** The function of this test was to verify the integrity and leak-tightness of the reactor coolant system and the high pressure portions of the auxiliary systems at 3105 psig (1.25 times the design pressure)	Hot shutdown (0 power)	--	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the pressure of the Reactor Coolant System will be the same as the current operating pressure and the design pressure of the system is not changing; therefore, the hydro-test does not need to be performed again.
NA **14.6.1.6.8**	**Ventilation Systems Test** Several tests were written and performed on the various ventilation systems of Ginna Station.	Shutdown	--	--	The EPU did not modify the ventilation system and the testing/balancing that was performed during startup is still valid; therefore, testing of the ventilation system will not be performed. Monitoring of general area temperatures, particularly those areas where new equipment is installed, will be performed as part of the power ascension test procedure to confirm that the ventilation system continues to perform its intended function.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
NA **14.6.1.6.9**	**<u>Preoperational Containment vessel Leak Rate Test</u>** The purpose of this test was to establish the degree of leak-tightness of the reactor containment building, penetrations, and isolation valves at the design pressure of 60 psig and establish a reference test for subsequent retests at 35 psig.	Shutdown	--	--	The EPU did not change or alter any of the penetrations and the original test results are still valid. The allowable leakage of 0.2%/day was defined by the design basis accident in accordance with 10CFR100. The test results were satisfactory and the test does not need to be repeated. Containment design pressure is not changing. Additionally, containment leakage is verified by regular surveillance testing.
NA **14.6.1.6.10**	**<u>Structural Integrity Test</u>** The purpose of this test was to functionally verify the strains and displacements of the cylinder wall, the discontinuity of the dome and cylinder wall, and the dome.	Shutdown	--	--	The containment structure was subjected to 69 psig (115% of design pressure of 60 psig) and found to have greater structural stiffness than anticipated. This test does not have to be repeated because the EPU did not modify the containment structure or penetrations in any way.
SU 9.7 **14.6.1.6.11**	**<u>Reactor Trip System (RTS) Operation Time Response Test</u>** This test was used to determine the response time from the time the plant protection parameters reach their setpoints until the tripping time of the reactor trip breakers.	Shutdown	Yes	--	This test was originally performed at less than 80% power. The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the trip response time limits as specified in Chapter 14 of the FSAR were proven conservative by this test. Since the trip logic of the RTS was not modified as part of this EPU, the response time for tripping the reactor breakers will be the same and do not require retesting. Additionally, the operation of these systems is verified by regular surveillance testing.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 8.5 **14.6.1.7.6**	<u>**Operational Dynamic Rod Drop Test**</u> The purpose of this test were as follows: (A) Demonstrate the operation of power range rod drop detection circuits and to provide a basis for the optimum adjustments of setpoints. (B) Demonstrate the operation of turbine runback controller and blocking of automatic rod withdrawal. (C) Evaluate the plant transient response following a dropped rod and demonstrate the adequacy of the dropped rod recovery procedure.	Mode 1 50% Power		No	The dropped rod recovery procedure was proven adequate and in subsequent testing, the turbine runback controller performed as designed. This system has been fully tested and found to be satisfactory and the EPU will not affect this system so testing again is not necessary.
SU 9.3.1, 9.3.3 **14.6.1.7.7**	<u>**Delta T Zero Alignment and Delta T Channel Span Adjustment Tests**</u> The delta T zero power alignment test provided instructions for the zero alignment for all four delta T channels. The delta T channel span adjustment test provided a curve of amplifier output versus plant load.	Hot shutdown (0 power) to 75% power		Yes	The 17% power increase for the EPU will mean that the 75% calorimetric value for the existing power rating will not be valid for the new 75% power rating under the EPU. The amplifiers for each protection channel will have to be span adjusted for the actual power level to provide an output as dictated by the linear curve of amplifier output versus plant load. These tests are performed as part of normal plant start-up.
SU 9.3 **14.6.1.7.8**	<u>**Nuclear Instrumentation Calibration and Reactor Coolant System Flow Confirmation**</u> The purpose of this procedure was to specify the requirements for obtaining data for nuclear instrument calibration and reactor coolant system flow confirmation and to check the performance of the nuclear instruments.	Mode 1 Shutdown to Full Power		Yes	Nuclear instrumentation calibration is performed as part of normal reactor start-up. The flow confirmation test is not impacted by EPU, but a calorimetric flow test will be performed at 85% and 100% EPU power.

Table 2.12-3 Comparison of Proposed EPU Tests to Original Startup Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup, Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 9.4 **14.6.1.7.9**	**Ex-Core In-Core Calibration Test** The function of this test was to establish a relationship between in-core and ex-core generated axial offset and delta flux.	Mode 1		Yes	The results of this test are used to calibrate the upper and lower detector channels and align the axial offset signals to the delta T setpoints. The higher power allowed by the EPU will require the recalibration of the upper and lower detector channels. This test is performed as part of normal reactor start-up.

Table 2.12-4 Transient Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 8.1.1 **14.6.1.7.1**	<u>**Ten Percent Load Swing Test at Thirty Percent Power**</u> The purpose of this test was to introduce a 10% load decrease and verify the nuclear plant transient response including automatic systems performance and then introduce a 10% increase in load and verify the response and performance as the system returned to 30% power.	Mode 1	--	Yes	This test was originally performed at less than 80% power. Specifically, this test verified the adequacy of various plant systems to respond to load swings. This test was later repeated at 100% power. This load swing test will be performed at both 30% and 100% of the post-EPU power level. The rate of load rejection and load increase will be 1% per minute.
SU 8.2.1 **14.6.1.7.2**	<u>**Generator Trip Test**</u> This test was to verify the ability of the automatic control system and the secondary plant to sustain interaction between systems and accommodate a net electrical load loss from below 50% power.	Mode 1 (110 Mwe)	--	No	This test was originally performed at less than 80% power and was done to validate the plant response at less than 100% power prior to the 100% plant trip test. The power uprate does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. See Section 2.12.1.2.7 for additional justification for not performing this test.
SU 8.1.2 **14.6.1.7.3**	<u>**Ten Percent Load Swing Test at Seventy-Five Percent Power Level**</u> The purpose of this test was to introduce a 10% load decrease and verify the nuclear plant transient response including automatic systems performance and then introduce a 10% increase in load and verify the response and performance as the system returned to 75% power.	Mode 1 (348 Mwe)	--	No	This test was originally performed at less than 80% power. The power uprate does not invalidate the test as originally performed. Specifically, this test condition is enveloped by the 10% load swings at 100% power that will be performed at the new post-EPU 100% power condition. Also, this test would be difficult to perform due to fuel pre-conditioning.

Table 2.12-4 Transient Tests					
SU Test # FSAR Section	**Test Description**	**Original Power Ascension Test Power Level**	**Normal Startup Surveillance, or Low Power Physics Testing (Yes/no)**	**Test Plan For EPU (yes/no)**	**Evaluation/Justification**
SU 8.6.1 **14.6.1.7.4**	**Fifty Percent Load Reduction from Seventy Five Percent Power Level Test** The purpose of this test is to verify the ability of the automatic control system and the ability of the secondary plant to sustain a 50% load rejection from 75% of full power and the interaction between the systems.	Mode 1	--	No	This test was originally performed at less than 80% power. The power uprate does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the testing and analyses performed for power uprate will validate that the plant systems remain capable of performing to accommodate this transient. See Section 2.12.1.2.7 for additional justification for not performing this test.
SU 8.1.3 SU 8.2.1 **14.6.1.7.5**	**One Hundred Percent Power Level Transient Tests** Ten percent and 50% load swing tests were performed at the 100% power level and were identical to the same load swings at 75% power. The results of these tests were satisfactory and similar to those at the 75% power level. Another 100% power test required the operator to push the manual turbine trip button on the main control board and monitor the results. This test was successful.	Mode 1	--	Yes (Partial**)	The analyses performed to support EPU indicate that the plant response to these transients is not adversely impacted as a result of EPU. See Section 2.12.1.2.7 for additional justification for not performing this test. Additionally, certain planned transient tests will be performed that adequately assess the ability of the modified plant systems to safely perform their intended function at the EPU power. See section 2.12.1.2.3 for further information. **Note: 50% load swing test and 100% turbine trip tests will not be performed.

2.12.1.2.5. Post Modification Testing Requirements

Plant modifications will be implemented at Ginna in order to achieve the EPU rated power. Plant modifications are controlled by administrative procedures. These procedures provide configuration control, installation instructions, and testing requirements. Post modification testing verifies satisfactory performance of the modification in accordance with the design documentation. The performance of post modification testing is addressed by existing programmatic controls. A review of the EPU plant modification performance criteria has been performed, although the final design of the modifications is not complete. Functional and operational post modification testing will be performed for each modification to verify satisfactory installation and performance. The post-modification tests that are considered in the verification of plant integrated response to transients are listed in Table 2.12-5.

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
Moisture Separator Reheater (MSR) Safety Valve Capacity Increase	Increase the safety valve setpoint of one existing valve and replacing five others with a rupture disc type relieving device.	No	No	Valves will be modified, set, certified, and tagged at vendor. ASME B31.1 Initial Service Leak Test required for header mods and new valve welds and joints	No startup testing required beyond in-service leak test.
Heater Drains System Modifications	Modify the drain systems as follows to accommodate the increased flow rates: 1. Replace the Feedwater heater 1A/B normal vent system orifices. 2. Add 1" vent line from each reheater 4th pass drain level control tank to the scavenging steam near the moisture separator cycle steam inlet. 3. Add a disengagement chamber (enlarged section of pipe, vented back to reheater head) for the 2nd pass reheater at some a place below the reheater. 4. Replace 8" heater drain tank emergency drain valve with larger capacity 10" valve.	No	No	Channel Calibration. Air Operated Valve (AOV) Testing	Heater and MSR drain tank levels will be monitored for stability during power ascension.
Feedwater Isolation Valve New Operators	Add an air operator to the two existing manual valves to provide less than 30 seconds closure time.	Yes	Yes	Closure Stroke Time Test AOV Diagnostic Baseline ASME Class 2 Pressure Test Position Indication Verification Functional Test of Low pressure switch and low pressure alarm Continuity check of all wiring Test operation of all controls and interlocks	None beyond Post-Mod Tests
Condensate Storage Tank Volume Increase	Raise the condensate storage tank overflow to provide additional tank capacity.	Yes	Yes	In Service Leak Tests of Welds	None beyond in service leak test

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
Stand-by Auxiliary Feedwater Flow Capacity Increase	Replace internal valve trim for additional flow capacity.	Yes	Yes	Channel Calibration Air Operated Valve (AOV) Testing ASME XI i.a.w. section IWV	System Capacity increase verified by test prior to exceeding 1520 MWt.
High Pressure Turbine Modifications	Replace the High Pressure rotor, new inner cylinder, and modify the inlet sleeves. Replacing turbine control valves with new design.	No	No	120% Rotor Factory Test Overspeed Trip Test Vibration Monitoring Thermal Performance Test	Turbine Overspeed Test during EPU startup. Valve Testing will be performed during power ascension.
Condensate Cooler Replacement	Replace the existing cooler to provide additional capacity for the Generator Hydrogen Coolers.	No	No	Initial Service Leak Test of Piping Heat Exchanger Performance Test	Monitoring during power ascension.
Feedwater Regulating Valve Trim Change	Replace internal valve trim for additional flow capacity.	Yes	Yes	Channel Calibration Air Operated Valve (AOV) Testing Stroke Time Testing	SG Level and FW Flow Control dynamic testing for verification of acceptable EPU performance
Main Feedwater Pump Motor HP Increase	Replace the existing two Feedwater pump motors (4500 hp) with 5500 hp motors	Yes	Yes	Pre-Operation Electrical Tests Perform set point testing on overcurrent protective relay Perform continuity check Check motor rotation	SG Level and FW Flow Control dynamic testing for verification of acceptable EPU performance
Main Feedwater Pump Impeller Replacement	Install new impellers into existing pump casings to increase Feedwater flow capacity.	Yes	Yes	Pump Performance Test Vibration Monitoring	SG Level and FW Flow Control dynamic testing for verification of acceptable EPU performance

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
Condensate Booster Pump	Replace the existing pumps with a new pump physically similar (externally) to the existing except that the new impellers are larger in width and diameter. Pump casings will be machined to fit the new impellers. Pump motors will be increased form 350 hp to 500 hp.	No	No	Pump Performance Test Vibration Monitoring Pre-Operation Electrical Tests on Motor Check pump and motor rotation	Monitor feed pump suction pressure and general performance to EPU conditions.
I/C Scaling & Setpoint Changes	NSSS Instrument Modifications : **1. 3.5 second time delay on T_{hot} - 4 channels.** **2. Steam dump setting changes.** **3. Control rod power mismatch gain setting changes.** **4. T_{avg} program setting changes.** **5. Pressurizer level program setting changes.** **6. OTΔT fΔI amplifier module replacements.** Balance of Plant Modifications: Transmitter Replacements: 1. Feed pump suction flow FT-2004 & 2005. **2. Standby aux feed pump discharge flow FT-4080 & 4085. (235 gpm or 250 gpm)** **3. SG feedwater flow FT-466,467,476,477, 500 & 503.** Transmitter/Instrument Calibrations: 4. Heater drain pump discharge flow FT-2003. **5. Feedwater pump low suction pressure switches (PS-2010 & PS-2011) replacement.** **6. 1st Stage Turbine Pressure (PT-485 & 486) rescale.** **7. 10 to 15 second time delay and setting**	Yes – Those **BOLDED**	Yes	Channel Calibration.	All physical instrument and setpoint changes will be validated as correct and functioning as expected prior to exceeding 1520 MWt. Performance of equipment affected by listed I&C changes will be monitored during EPU power ascension.

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
	change to low pressure feedwater heater by-pass valve V3959. **8. Generator protection and voltage regulator setting changes.** **9. Condensate pump auto-start and alarm setting changes.**				
Isophase Bus Duct Cooling Capacity Increase	Replace 7.5 HP assemblies with 25 HP assemblies Provide interlocks to prevent operation of both fan assemblies. Install additional instrumentation	No	No	Pre-Operation Electrical Tests Perform set point testing on overcurrent protective relay Perform continuity check Check motor rotation	Monitor Isophase Bus Duct temperature during EPU power ascension.
Oilstatic Cable Forced Oil Cooler Capacity	Install temperature data acquisition system and additional temperature wells and ambient temperature probes. Modifications to the pumping plant system. Modifications to the cathodic protection system. Addition of heat tracing to the above ground cross connect piping.	No	No	Pre-Operation Electrical Tests Perform continuity check	Monitor temperature during EPU power ascension.
Main Transformer Bushing Replacement	1. Replace existing (3) 3000A high voltage bushings with new 3500A bushings. 2. Replace or modify existing (3) high voltage bushing adaptors. 3. Install (5) new cooling assembly units. 4. Replace condition monitor including remote monitoring capability. 5. Refurbish transformers.	No	No	Testing at completion of re-assembly to include: Power Factor – Doble Method Excitation – Doble Method Turns Ratio Megger Oil Dielectric Testing Dissolved Gas Analysis Furan Testing Hydro Test Fire Suppression System	Monitor temperature during EPU power ascension. **COMPLETED IN 2005**

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
				Operational Spray System Test **COMPLETED IN 2005**	
Generator Instrumentation	Installation of the following three generator monitoring instrumentation systems: 1. Fiber Optic Vibration Monitoring System to be integrated into the Plant Process Computer system. 2. Flux Probe and cables. 3. Partial Discharge (PD) Monitoring Instrumentation.	No	No	Channel Calibration. Pre-Operation Electrical Tests Perform continuity check **COMPLETED IN 2005**	Monitor installed generator instruments during EPU power ascension. **COMPLETED IN 2005**
Replace MS and FW Snubbers and Rods	Replace five snubbers and six hanging rods in various BOP locations in the Turbine Building.	No	No	ISI and NDE of welds and joints.	Vibration monitoring of MS and FW piping.
MRPI Spacer or Programming Change	MRPI Coil Stack Spacer/Software Change to Address New Fuel Assembly Height	No	No	Post-modification test of rod position indication.	Rod drop testing as part of normal startup tests.

Table 2.12-5 Post Modification Testing					
Modification Title	**Modification Description**	**Potential Impact on Transient Response**	**Modeled in Transient Analysis**	**Post Modification Test**	**EPU Startup Testing**
Main Turbine Gland Sealing Steam Spillover	Modify gland sealing steam spillover line to dump excess gland sealing steam to condenser.	No	No	ISI and NDE of welds and joints.	Monitoring of gland sealing steam performance during power ascension.

2.12.1.2.6. Transient Analytical Methodology

Condition I Initiating Events

Analyses and evaluations have been performed for the Condition I operating transients to assess the aggregate impact of the equipment modifications and setpoint changes for EPU conditions. These analyses and evaluations used the same principal computer code (i.e., LOFTRAN) that has been used in control system analyses for Ginna Station at current power conditions. The LOFTRAN computer code is described in WCAP-7907-P-A (LOFTRAN Code Description, April 1984). The code has been approved by the NRC and has been used for many years by Westinghouse for accident evaluations for Safety Analysis Reports, and for control system performance and equipment sizing studies.

LOFTRAN has been used in the analysis of Condition I initiating events on Ginna Station as well as on other Westinghouse designed nuclear power plants. The NRC Safety Evaluation Report (SER) included in WCAP-7907-P-A describes the LOFTRAN verification process performed by Westinghouse for transients including reactor trip from 100% power, 100% load reduction, and step load changes. The verification process consisted of comparison of LOFTRAN results to actual plant data and to other similar thermal-hydraulic programs. The LOFTRAN verification process also included analysis of the R. E. Ginna steam generator tube rupture (SGTR) event, where comparison of the LOFTRAN results to available plant data demonstrated the ability of LOFTRAN to analyze the SGTR event.

The NRC SER included in WCAP-7907-P-A concludes that the data comparisons and the results comparisons provided by Westinghouse demonstrate the ability of LOFTRAN to analyze the types of events for which it has been used in licensing safety analysis. In conjunction with its extensive use for many years, it has been used in evaluation of Condition I operating transients at many Westinghouse designed nuclear power plants including other similar Westinghouse designed 2-loop nuclear power plants currently operating at approximately 1775 MWt NSSS power.

The LOFTRAN computer code was used to analyze the following Condition I initiating events and Condition II turbine trip transient at Ginna Station at EPU conditions:

- Step load increase of 10% of full power from 90% to 100% power
- Step load decrease of 10% of full power from 100% to 90% power
- Large load rejection of 50% of full power from 100% power
- Turbine trip without reactor trip initiated from P-9 setpoint (49% power)
- Turbine trip from 100% power

Based on these limiting analyses run with LOFTRAN, the ramp load increase and decrease of 5% of full power per minute between 15% to 100% power was evaluated as being acceptable at the EPU conditions.

The LOFTRAN analysis inputs and models were updated as appropriate to incorporate the applicable EPU equipment modifications and setpoint changes as well as the EPU operating conditions. The analyses results showed that the plant responses to Condition I initiating events satisfied acceptance criteria and that the NSSS control system responses were stable. Furthermore, plant responses to Condition I initiating events were shown to have acceptable margins to reactor trip and engineered safety features actuation. The results of the analyses performed for Condition I initiating events at EPU conditions are reported in LR section 2.4.2. The plant responses to Condition I initiating events at EPU conditions are consistent with their characteristic responses based on operational and analytical experience on Ginna at the current power conditions as well as operational and analytical experience on other similar Westinghouse designed 2-loop nuclear power plants (Kewaunee) currently operating at approximately the same NSSS power.

Condition II, III, and IV Initiating Events

Analyses and evaluations have been performed for the Condition II, III, and IV operating transients to assess the aggregate impact of the equipment modifications and setpoint changes for EPU conditions. Analysis inputs and models were updated as appropriate to incorporate the EPU equipment modifications and setpoint changes as well as the EPU operating conditions. These analyses results showed that the plant responses to Condition II, III, and IV initiating events satisfied acceptance criteria. The results of the analyses performed for Condition II, III, and IV initiating events at EPU conditions are reported in LR section 2.8.

The principal computer codes that were used in the analyses and evaluations for Condition II, III, and IV initiating events for Ginna Station at EPU conditions are, with two exceptions, the same as those used for the Ginna Station analyses of record for current power conditions. The two exceptions are for the Condition IV large break LOCA where the NRC-approved Westinghouse BELOCA methodology and WCOBRA/TRAC computer code were used, and for the Condition II Non-LOCA transients where the NRC-approved RETRAN computer code and VIPRE computer code was used. All of these computer codes are generically NRC-approved and are used in the analyses of Condition II, III, and IV initiating events for other Westinghouse designed nuclear power plants. Furthermore, analytical and operational experience at these other Westinghouse designed plants supports the use of these computer codes for assessing the aggregate impact of the EPU equipment modifications and setpoint changes in combination with the EPU operating conditions on Ginna Station.

The dynamic plant responses to Condition II, III, and IV initiating events at EPU conditions with the EPU equipment modifications and setpoint changes are consistent with their characteristic responses based on operational and analytical experience at other similar Westinghouse designed 2-loop nuclear power plants (Kewaunee) currently operating at approximately the same core thermal power.

Natural Circulation

Natural circulation capability for the Ginna EPU is evaluated using the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) methodology. This method is used to estimate flow rates and core delta temperatures using core hydraulic resistance coefficients. Further discussion of this methodology appears in Section 2.12.1.2.7 below.

2.12.1.2.7. Justification for Exception to Transient Testing

Ginna has reviewed the recommendations of draft Standard Review Plan (SRP) 14.2.1 for the EPU testing programs. As a result of this review, and a review of the original Ginna startup test program and recommendations from the NSSS vendor, Ginna concludes that no large load transient tests need to be performed as part of the EPU test program. However, due to the number of modifications made to the Balance of Plant systems to accommodate EPU power levels, Ginna is proposing several smaller transient tests as previously described. This section discusses the justification for not performing the large transient tests.

Original Startup Transient Tests

The following transient tests were performed during the initial power ascension test program:

Transient Test	Startup Test #
+/-10% Step Load Test at 30% Power	14.6.1.2.1
Net Electrical Load Loss from Below a 50% Power Level	14.6.1.2.2
+/-10% Step Load Test at 75% Power Level	14.6.1.2.3
50% Load Reduction from 75% Power Level	14.6.1.2.4
100% Power Level Transient Tests	14.6.1.2.5
Natural Circulation Test	14.6.1.5.6

Justification for Exception - General

Ginna is being modified to allow for operation at the process conditions associated with 1775 MWt core power level. The LOFTRAN computer code was used to evaluate plant response to Condition I and II initiating events at EPU conditions. The LOFTRAN computer code has been verified with respect to plant data and has been approved by the NRC for use in licensee safety analysis. The LOFTRAN verification process included comparison with plant data for transients including reactor trip from 100% power, 100% load reduction, and step load changes. The LOFTRAN verification process also included analysis of the Ginna steam generator tube rupture (SGTR) event, where comparison of the LOFTRAN results to available plant data demonstrated the ability of LOFTRAN to analyze the SGTR event. The code has been used by Westinghouse for accident evaluations for Safety Analysis Reports and for control system performance and equipment sizing studies. The application of the LOFTRAN computer code to Ginna considers any limitations included in NRC approval of the code along with plant-specific operating parameters and system configurations.

The LOFTRAN computer code has been used for Ginna Station for many years at the original and current power levels. In addition to its use on Ginna, it has also been used in evaluation of Condition I and II operating transients at many Westinghouse designed nuclear power plants including other similar Westinghouse designed 2-loop nuclear power plants. This use of LOFTRAN for analysis in a wide variety of different Westinghouse plants for various types of transients – both licensing/design basis analyses and for plant problem troubleshooting – has shown that this computer code can acceptably be used to predict the plant response, thereby negating the need to perform plant transient testing to validate the predicted code responses to large plant transients.

The LOFTRAN analysis inputs and models were updated as appropriate to incorporate EPU-related changes to parameter and setpoint values. Bounding inputs for design parameters were used as described in LR section 1.1. Analyses and evaluations were then performed for the NSSS control systems at EPU conditions. The NSSS control systems include the reactor (rod) control system, reactor coolant temperature (T_{avg}) control system, pressurizer level control system, pressurizer pressure control system, steam generator level control system, and steam dump control system. NSSS control systems setpoints are being revised as required to support EPU operations. Control systems including the rod control and T_{avg} control system, pressurizer level control system, and steam dump control system will have set points changed as described in LR section 2.4.1.

NSSS control systems analyses were performed at EPU conditions for the following design basis Condition I operating transients and the Condition II turbine trip transient to demonstrate acceptable stability and set points:

1. 10% step load increase from 90% to 100% of uprated full power
2. 10% step load decrease from 100% to 90% of uprate full power
3. 50% load rejection from 100% of uprated full power
4. Turbine trip without reactor trip from P-9 setpoint (49% of uprated full power)
5. Turbine trip followed by a reactor trip from 100% of uprated full power

The NSSS control systems analyses assessed the aggregate impact of the applicable equipment modifications and setpoint changes at EPU conditions. The analyses results demonstrate that plant response to Condition I and II initiating events satisfies acceptance criteria, NSSS control systems responses are stable, and margins to reactor trip and engineered safety feature actuations are acceptable. Specifically, the performance of the rod control system and the steam dump control system is acceptable during both steady state and transient operating conditions. The results also show that sufficient operating margins exist to reactor trip and engineered safety feature (ESF) actuation set points at EPU conditions with the NSSS control systems in the automatic mode. The NSSS control systems' pressure control components (i.e., pressurizer power operated relief valves, pressurizer spray valves, pressurizer heaters, and condenser steam dump valves) satisfy sizing requirements at EPU conditions and are acceptable for the analyzed transients.

These results are consistent with experience on several similar Westinghouse-designed 2-loop nuclear power plants that use the LOFTRAN computer code for analysis of Condition I and II initiating events and operate at approximately the same NSSS power level as for Ginna Station at EPU conditions.

Other process parameter changes being made to accommodate the power increase are within the design capability of the related systems, or necessary upgrades are being installed. Therefore, no new thermal-hydraulic phenomena are introduced by either the physical modifications or the changes in operating conditions. Furthermore, the results of the analyses indicate that no new system dependencies or interactions are being introduced by the changes. However, several transient tests are proposed as part of the power ascension test program to validate these results. Additional large plant transient tests are being considered for verification of integrated plant performance.

As discussed above, the aggregate impact of the EPU equipment modifications and setpoint changes on the dynamic plant response of Ginna Station for Condition I and II initiating events at EPU conditions was assessed through analyses and evaluations. These analyses and evaluations used the LOFTRAN computer code, which has been verified and approved by the NRC. The extent of the aggregate impact of the EPU equipment modifications and setpoint changes on dynamic plant response is such that it can be adequately addressed through analyses and evaluations. It is accepted practice to use analyses and evaluations to assess the aggregate impact of these types of equipment modifications and setpoint changes on Ginna as well as on other Westinghouse designed nuclear power plants.

Therefore, performing the load transient tests identified above would not confirm any new or significant aspect of performance not already demonstrated through analysis, by previous operating experience or routinely through plant operations. The following provides a description of the load transient tests and justification for exception.

Justification for Exception - Specific

+/-10% Step Load Change Test (14.6.1.2.1 and 14.6.1.2.3)

The purpose of the +/-10% step load change test during the initial startup test program was to verify plant control system response to small but rapid load changes. The test verified the ability of the pressurizer level and pressure control system to maintain parameters within design limits and provide for stable plant operation. An analysis of a +/-10% step load change was performed using the LOFTRAN code as described in <u>LR section 2.4.2</u>. This analysis demonstrated that the Ginna plant response to +/-10% step load changes at EPU conditions is acceptable. Since the NSSS control system functional design and hardware are not impacted and the analyzed +/- 10% step load change Condition I operating transients show acceptable stability, setpoints, and margin to reactor trip and ESF actuation, the NSSS control systems are acceptable for operation at full power EPU conditions. Based on these NSSS control systems analyses, +/-10% step load change tests are not required in the Ginna EPU Power Ascension Test Plan. However, +/-10% load swing tests at 1% per minute ramp rates will be performed at 30% and at 100% EPU power levels to verify proper operation of the plant and automatic control systems.

Electrical Load Loss from Below 50% Power Test (14.6.1.7.2) and

Loss of 50% Load at 75% and 100% Power Tests (14.6.1.7.4 and .5)

The net electrical load loss from below 50% power and the loss of 50% load at high power are tests to demonstrate that the control systems act together to prevent a reactor trip and also prevent the opening of the main steam safety valves (MSSVs). In particular, the test demonstrates that the rod control, steam dump and pressurizer pressure and level control systems act together to control the NSSS response to within design limits and the reactor trip setpoints. An analysis of a 50% step load reduction from below the P-9 setpoint (49%) and from 100% EPU power were performed using the LOFTRAN code as described in <u>LR section 2.4.2</u>.

These analyses demonstrate that the Ginna plant response to 50% step load decrease at EPU conditions will not cause a reactor trip and will not cause the MSSVs to open.

There are no major hardware modifications planned for NSSS components as part of the EPU that would affect the plant transient response. Since the NSSS control system functional design and hardware are not impacted and the analyzed 50% load rejection Condition I operating transients show acceptable stability, setpoints, and margin to reactor trip and ESF actuation, the NSSS control systems are acceptable for operation at full power EPU conditions. A reactor trip, or the potential for a reactor trip, from high power level results in an unnecessary plant transient and the risk associated with such a transient, while small, should not be incurred. Based on this analysis and the avoided risk of an unnecessary plant transient, a step load reduction of 50% from below the P-9 setpoint and from 100% EPU power to verify proper operation of the plant and automatic control systems is not required in the Ginna EPU Power Ascension Test Plan.

Manual Turbine Trip from 100% Power Test (14.6.1.2.5)

The manual turbine trip from 100% power is a test to demonstrate that the control systems act together to maintain NSSS parameters within design limits post-trip and to demonstrate MSSVs do not open. In particular, the test demonstrates that the rod control, steam dump and pressurizer pressure and level control systems act together to control the NSSS response to within design limits and prevent opening of MSSVs. An analysis of a turbine trip from 100% EPU power was performed using the LOFTRAN code as described in LR section 2.4.2. This analysis demonstrates that the Ginna plant response to a turbine trip at full power EPU conditions results in acceptable response of pressurizer level and pressure, and MSSVs do not open.

There are no major hardware modifications planned for NSSS components as part of the EPU that would affect the plant transient response. Since the NSSS control system functional design and hardware are not impacted and the analyzed turbine trip from 100% EPU power Condition II operating transient shows acceptable stability, setpoints, and margin to ESF actuation, the NSSS control systems are acceptable for operation at full power EPU conditions. A reactor trip, or the potential for a reactor trip, from high power level results in an unnecessary plant transient and the risk associated with such a transient, while small, should not be incurred. Based on this analysis and the avoided risk of an unnecessary plant transient, a manual turbine trip from 100% EPU power to verify proper operation of the plant and automatic control systems is not required in the Ginna EPU Power Ascension Test Plan.

Natural Circulation Test (14.6.1.5.6)

The purpose of the natural circulation test is to demonstrate the capability of natural circulation to remove core decay heat while maintaining NSSS parameters within design limits. The test was performed at 2% power and demonstrated that natural circulation flows were adequate to remove heat and maintain NSSS parameters in an acceptable range.

To evaluate the natural circulation capability for the Ginna station EPU, the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) methodology is used to estimate flow rates and core delta temperatures using core hydraulic resistance coefficients. These equations are evaluated for several decay heat assumptions (1, 2, 3, and 4%) over a range of temperature conditions. This analysis of natural circulation cooldown to residual heat removal (RHR) cut-in conditions is described in more detail in LR section 2.8.7.2.

The estimated loop delta temperatures at selected decay heat levels are compared to the original natural circulation test results at a lower power level. The EPU calculated loop delta temperature values are multiplied by the power ratio (1311 MWt / 1811 MWt) raised to the 2/3 power to obtain calculated loop delta temperatures at the original Ginna startup power level. The resulting scaled loop delta temperatures are shown in <u>LR section 2.8.7.2</u>.

In addition, the atmospheric relief valve (ARV) capacities are estimated as function of steam generator secondary pressure that is correlated with primary system saturated temperature. After 4 hours at hot standby conditions, the plant is assumed to cool down to the RHR cut-in conditions at the maximum Emergency Operating Procedure (EOP) rate (25°F/hour).

There is close agreement between the hydraulic resistance coefficients for the Diablo Canyon and Ginna plants at the uprated conditions and the loop flow ratios are in good agreement. The calculated loop delta temperatures show the same trends and slightly higher scaled values compared to the UFSAR reported measured values. The natural circulation flow rate shows expected behavior – decreases as the decay heat decreases at a constant temperature and a decrease with temperature at a constant value of decay heat. The loop delta temperature shows expected behavior – decreases as the decay heat decreases at a constant core average temperature and increases as the core average temperature decreases at a constant value of decay heat.

For the following reasons, the Ginna EPU will not adversely impact the natural circulation cooldown capability of the plant:

- No major hardware modifications to NSSS components that could affect loop flow resistance or steam generator heat transfer are part of the EPU scope.
- Acceptable results are found for natural circulation cooling during the hot standby period for realistic residual heat rates as high as 3% of 1811 MWt. The core outlet temperatures calculated for this case (604.5°F) are bounded by those specified for full power operation for the high Tavg cases (611.8°F) (PCWG Cases 3 and 4, LR section 1, "Nuclear Steam Supply System Parameters", Table 1-1).
- The calculated loop delta temperatures are scaled and compared to the UFSAR measured values. The scaled, calculated values show the same trends as the original measurements and are slightly larger than measured, due to several conservative assumptions in the calculations. One of the conservative assumptions is that the hydraulic resistance for the reactor coolant pump (RCP) is based upon a locked-rotor K value.
- The atmospheric relief valves (ARVs) at the uprated conditions are adequate to achieve cooldown to the RHR entry point in an acceptable time period. RHR cut-in conditions can be achieved in approximately 14 hours at the maximum rate specified in plant procedure ES-0.2, which includes 4 hours in hot standby conditions.

The natural circulation behavior for Ginna is essentially unchanged for EPU conditions. Core outlet temperatures remain bounded by full power operating conditions and subcooling is adequate. Based on this analysis, a natural circulation test is not required in the Ginna EPU Power Ascension Test Plan.

2.12.1.3. Conclusion

Ginna has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. Ginna concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, Ginna finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10CFR50, Appendix B, Criterion XI. Therefore, Ginna finds the proposed EPU test program acceptable.

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

2.13.1.1 Regulatory Evaluation

The risks associated with the proposed extended power uprate (EPU) are acceptable. No "special circumstances" are created by the proposed EPU. As described in Appendix D of the Standard Review Plan (2.13.2, Reference 1) Section 19, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided to meet the deterministic requirements and regulations. The Ginna staff review covered the impact of the proposed EPU on core damage frequency and large early release frequency for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the Ginna staff review covered the quality of the risk analyses used to support the application for the proposed EPU. This included a review of the actions to address issues or weaknesses that have been raised in previous Ginna staff and industry-peer reviews of the individual plant examination and individual plant examination of external events. The NRC's risk acceptability guidelines are contained in Regulatory Guide 1.174 (2.13.2, Reference 2). Specific review guidance is contained in Matrix 13 of RS-001 (2.13.2, Reference 13) and its attachments.

Ginna Current Licensing Basis

The Ginna Station Level 1 and Level 2 PSA Model (GPSA) was initially developed in response to NRC Generic Letter 88-20 (Individual Plant Examination, or IPE). Since the original IPE submittal, the GPSA has undergone several model revisions to incorporate improvements and maintain consistency with the as-built, as-operated plant.

The GPSA Revision 5.0 update involves extensive revision of the human reliability analysis along with enhancements to thermo hydraulic analysis, fire modeling, station blackout modeling, and steam generator tube rupture modeling. In addition, the RCP seal LOCA modeling is revised to current Westinghouse standards (2.13.2, Reference 10). Overall, the GPSA is reviewed and upgraded with a goal of increased fidelity in areas related to EPU.

2.13.1.2 Technical Evaluation of EPU

This section describes the risk analysis associated with the extended power uprate. The EPU will implement/license an increase in the NSSS power level from 1526 MWt to 1781 MWt. The safety analyses that support the Station license will be performed at an NSSS power level of 1817 MWt (core power of 1811 MWt, up from 1550 MWt pre-EPU). The evaluation addresses the power uprate impacts on:

Initiating event frequencies,
Component and system reliability,
Operator response times,

Success criteria, and
Overall impact on CDF and LERF.

The evaluation uses an update to the GPSA Version 4.3 Model. Various aspects of the model development are described in the following sections of this risk evaluation. The GPSA is a Level 1 and 2 All Modes model that includes Internal Events, Fire, Internal Floods, and Shutdown.

Risk evaluation conclusions and insights are provided for each discussion topic. The extended power uprate is not a risk informed application in that deterministic calculations are used to prove acceptability. This risk evaluation is provided for information purposes and insights.

2.13.1.2.1 Ginna Level 1 Model

The GPSA uses standard small event tree/large linked fault tree Level 1 methodology. Event trees are developed for each unique class of identified internal initiating events and top logic is developed to link these functional failures to system-level failure criteria using the Computer Aided Fault Tree Analysis (CAFTA) code. Fault trees, comprised of component and human failure events, are developed for each of the systems identified in the top logic. The exceptions are the Main Feedwater (MFW) System and the Reactor Protection System. Although these systems include dependencies with other systems (for example, electric power), the hardware is modeled at a higher level with a few representative basic events (for example, MFW Pumps fail to run).

Fault tree hardware-related events are quantified with a mixture of generic data from throughout the nuclear industry and Ginna Station specific data. A nine year (January 1, 1980 through December 31, 1988) data window was originally established for quantifying component failure rates; this data window is updated as new data becomes available. GPSA incorporates a complete update of both generic and plant-specific data, the latter appending operating data from January 1, 1994 through December 31, 2000 to the 1980's data where appropriate (or in some cases, replacing the 1980's data). Licensee Event Reports and other in-house event reporting systems, especially those established to monitor component behavior to comply with the Maintenance Rule, are also reviewed to ensure completeness.

Human failure events are quantified in two phases. In the first phase, conservative screening values are assigned to all human failure events identified in the logic models prior to model quantification. In the second phase, refined values are assigned to those human failure events identified as being significant based on an evaluation of first phase quantification results.

Solution of the event trees yields "cutsets," or those combinations of events which lead to core damage (or large early release). Sensitivity and importance analyses of the final results were also performed to help identify risk significance.

2.13.1.2.1.1 Internal Events

The Ginna Station internal events PSA addresses LOCA, SGTR, LOOP, internal flooding, and transient initiators. For internal event initiators, the underlying contributors to these initiating events are reviewed to determine the potential effects of the power uprate on the initiating event frequencies, with the following results:

2.13.1.2.1.1.1 Loss of Coolant Accident (LOCA)

These frequencies (all sizes) are determined by the potential for passive pipe failures and are not related to reactor power level. The EPU does not involve changes to the reactor coolant system or interfacing system piping. As there are no substantive changes to this system, the LOCA frequencies are not effected by the EPU.

A LOCA can also occur as a result of an RCS pressure excursion that results in a stuck open PORV or primary safety relief valve. The parameters below represent the likelihood of a PORV challenge given a reactor trip. Note that the first two items are defined based on their impact on turbine overspeed events, due to the fact that they are also used in the PSA model to address turbine missile issues. Also, although the loss of load events in the second item are not large enough to cause a reactor trip, this item is expressed in terms of a fraction of the reactor trip frequency in Table 2.13-1, so that it can be combined with the reactor trip frequency to result in an overall frequency for these events.

- RCAZTRIPLL – *Probability the reactor trip was caused by a Loss of Electrical Load (loss of load, LOL) large enough to cause a turbine overspeed challenge.*

  Any loss of load large enough to cause an overspeed challenge will also cause a PORV and MSSV challenge. To cause an overspeed challenge, a significant reduction in load is required. The majority of the unit operating time is at one-hundred percent power. Further, loss of load transients are predominated by either complete or small loss of load challenges. Because a complete loss of load from full power will cause an overspeed challenge both pre/post-EPU, the probability for this parameter will increase only slightly. Small loss of loads are not considered a turbine overspeed challenge.

- RCAZTRIPPM – *Probability of a loss of load not large enough to be a turbine overspeed challenge, but large enough to cause a PORV and MSSV challenge.*

  Both pre/post-EPU, a loss of load that exceeds fifty percent of full turbine load will cause a PORV and MSSV challenge. A loss of load that is less than ten percent of full turbine load will not cause a PORV/MSSV challenge, both pre/post-EPU. Only those loss of loads that are between fifteen and forty percent of full turbine load are considered to cause a PORV/MSSV challenge post-EPU, but not pre-EPU.

- RCAZTRIPPV – *Probability of a PORV challenge due to a pressurizer level control problem that causes a reactor trip.*

  Due to the reduced free volume in the pressurizer post-EPU, there is an increase in the likelihood of a PORV challenge.

The changes to these variables are based on power history, an estimated loss of load profile, and judgment. The results are shown in Table 2.13-1.

Table 2.13-1
LOCA Adjustment

Basic Event	Event Description	Base Value	EPU Value	Percent Change
RCAZTRIPLL	Fraction of Reactor Trips caused by a large LOL	4.35E-02	4.92E-02	13%
RCAZTRIPPM	Fraction of Reactor Trips caused by a medium LOL (i.e. PORV/MSSV Challenge with no Turbine Missile)	5.65E-03	8.63E-03	53%
RCAZTRIPPV	Fraction of Reactor Trips due to Pressurizer Level Control Problem resulting in a PORV Challenge	1.00E-03	1.50E-03	50%

2.13.1.2.1.1.2 Steam Generator Tube Rupture (SGTR)

Ginna Station installed replacement steam generators (RSGs) in 1996. The RSGs have more tubes resulting in increased surface area allowing increasing power operation. The RSGs are constructed using Alloy 690 material which will reduce corrosion effects and include construction features that control vibration resonance. Extensive flow induced vibration analyses show that with the EPU power increase no adverse impact is expected. Lastly, the ASME qualification report indicates that the RSGs maintain an acceptable cumulative usage factor and that the expected Ginna design transients envelope the transients that can be expected during forty years of operation.

The increased core heat associated with EPU is expected to result in an increase in steam flow during normal operation and post-trip. As a result, the time to overfill given a SGTR is expected to take longer or remain the same depending on how steam flow out of the steam generator is controlled. For this evaluation, all steam generator overfill scenarios are considered to have the same recovery time available pre/post-EPU.

Therefore, the existing SGTR modeling is considered applicable to power uprate conditions.

2.13.1.2.1.1.3 Loss of Offsite Power (LOOP)

A System Reliability Impact Study performed to evaluate the impact of the Ginna Station EPU on the reliability of the local 115kV and New York Independent System Operator (NYISO) bulk power systems indicates that the thermal, voltage, and stability performance are not degraded by the EPU. The power uprate does not necessitate replacement or modification of switchyard breakers or disconnects. In a normal switchyard/plant configuration, all switchyard circuits and equipment will operate within design limits. Note that certain switchyard/plant configurations may require a small load reduction to maintain equipment within operational limits: this will be administratively controlled.

The generator step-up transformer (GSU) and oilstatic (oil pipe) cables connecting the GSU to the switchyard will be upgraded to handle the increased power. Although not uniquely associated with grid effects, these components directly interface with the switchyard.

LOOP frequency is correlated with switchyard and grid reliability. The EPU does not cause the switchyard equipment to operate beyond design limits. However, it is possible that the unanticipated loss of some circuits may increase load on the remaining circuits to unacceptable levels. In most cases where an overload is projected, NYISO agrees that the identified overloads can be mitigated through various techniques.

To address any unforeseen switchyard reliability issues the LOOP frequencies are increased as shown in Table 2.13-2. Note that the percentage increases are based on engineering evaluation reports. Within the reports, components with reduced operating margin are identified. Judgment is used to estimate the initiating event frequency impact due to the change in operating margin for this equipment.

Table 2.13-2
LOOP Initiator Adjustment

Basic Event	Event Description	Base Value	EPU Value	Percent Change
TI48LOSPNW	Loss of Both Circuit 751 and 767 due to Non-Weather Related Causes as Initiator	2.54E-04	2.79E-04	10%
TIGRLOSP	Loss of Offsite Power - Grid	6.44E-03	7.08E-03	10%
TILOP751NW	Loss of Offsite Circuit 751 due to Non-Weather	2.50E-01	2.75E-01	10%
TILOP767NW	Loss of Offsite Circuit 767 due to Non-Weather	5.00E-02	5.50E-02	10%
TISWLOSP	Loss of Offsite Power - Switchyard	7.87E-03	8.66E-03	10%
ACLOPRT751	Loss of Offsite Circuit 751 Following Reactor Trip	1.28E-03	1.41E-03	10%
ACLOPRT767	Loss of Offsite Circuit 767 Following Reactor Trip	1.28E-03	1.41E-03	10%
ACLOPRTALL	Loss of All Off-Site Power Following Reactor Trip	2.59E-03	2.85E-03	10%

2.13.1.2.1.1.4 Transients

A transient initiator assessment is conducted to determine what component or system changes could impact the likelihood of a reactor trip.

Instrumentation and Control Systems: For the primary/secondary process parameter changes associated with the EPU, it is expected that the RPS trip setpoints and control systems (feedwater, pressurizer pressure/level, and steam dump/bypass) are tuned to accommodate a number of operational transients and ensure proper plant response without generating a reactor trip. The settings will also allow for variations in plant power levels and diverse operating conditions.

Power Block Equipment: Increased power output will result in a loss of operating margin for the Main Generator. For full output the generator must operate at or near a 1.0 power factor. It is recognized that certain plant equipment will operate at higher power regions with less margin (that is, equipment will operate closer to trip setpoints or capacity limits).

Plant Operation: Some control room instruments will be re-scaled to meet EPU conditions. Although it is the intent of the EPU to train operators on these instrument changes, it is possible that, initially, the likelihood of operator error is higher.

Based on the above items, the "plant trip" initiator is increased as shown in Table 2.13-3. Note that the percentage increases are based on engineering evaluation reports. Judgment is used to estimate the initiating event frequency impact due to the changes to this equipment.

Table 2.13-3
Plant Trip Initiator Adjustment

Basic Event	Event Description	Base Value	EPU Value	Percent Change
TIRXTRIP	Reactor Trip	1.25E+00	1.50E+00	20%

125VDC Distribution System: Other than some minor instrumentation and control additions, there are no changes to the 125VDC system. The additional load these components present is negligible and easily within the conservatism in the system load calculations. Thus, there are no changes to the 125VDC initiating event frequencies.

4160VAC Distribution System: The most significant change to the 4kV switchgear and buses is the load increase associated with larger condensate booster pump and main feedwater pump motors. Although this will decrease the 4kV operating margins, the switchgear and buses are adequately sized to support overall plant operation. Any slight increases in the likelihood that the 4kV switchgear is lost would be bounded by the increase in LOOP frequencies already considered (see 2.13.1.2.1.1.3, Loss of Offsite Power, Table 2.13-2).

120VAC Distribution System: No new loads are being added to the 120VAC system to support plant operation at EPU. Since the system and equipment were adequately rated to support plant operation prior to EPU, the system will continue to perform its intended functions during all plant operating and accident conditions. Therefore, there are no changes to 120VAC initiating event frequencies.

Service Water: The ultimate heat sink is Lake Ontario, which provides water to the Service Water System. The Service Water System provides cooling water for heat removal from safety-related heat exchangers and supplies water from the ultimate heat sink to the Standby Auxiliary Feedwater System (SAFW) and preferred Auxiliary Feedwater System for emergency heat removal from the reactor coolant system.

Although service water heat loads will increase as a result of the EPU, the system will continue to provide the required heat removal capability at EPU conditions and in postulated accident scenarios. The flow requirements for service water as a source of AFW will also increase to support additional heat removal at EPU. This increase is considered insignificant with regard to the Service Water System design and capacity.

Post-EPU, the containment recirculation fan coolers could fail in the unlikely event a single service water pump is supporting all the coolers given an 85°F (or greater) lake temperature and a design basis LOCA. This impact is addressed in the model by considering all the recirculation fan coolers failed during any medium or large LOCA where only one service water pump is running and the lake temperature is above 70°F. In the extremely unlikely event this occurs, there are several operator actions that could be used to recover from this situation. Due to the low contribution of this issue, the model does not contain these recovery actions. For this issue, the CDF increase due to EPU are shown in Table 2.13-4.

Table 2.13-4
Impact of Single Service Water Pump with LOCA at Higher Lake Temperatures

Basic Event	Event Description	Base Value	EPU Value	EPU CDF delta Increase
SWOOFAILCRFC	CRFCs fail due to inadequate SW support	0.00	1.00	2.92E-10

As there are no substantive changes to this system the initiator frequencies remain unchanged.

Component Cooling Water: Although Component Cooling heat loads will increase as a result of the EPU, the system will continue to provide the required heat removal capability at EPU conditions and in postulated accident scenarios. Further, EPU activities do not add any new components or introduce new functions for existing components.

As there are no substantive changes to this system the initiator frequencies remain unchanged.

Instrument Air: There are no substantive changes to this system and these initiator frequencies remain unchanged.

Main Feedwater: The main feedwater system operates at increased flow rates for power uprate. To accomplish this, the Condensate Booster Pump and Main Feedwater Pump impellers and motors will be replaced to obtain higher capacity and more horsepower respectively. The EPU also requires modifications to other feedwater components (for example, the Feedwater Regulating Valves are modified to reduce frictional pressure drop).

Piping segments are analyzed, and where flow velocity or vibration exceed industry standards, the appropriate initiator is adjusted. For main feedwater, certain turbine building pipe sections exceed industry standards for flow velocity (some piping sections also exceeded standards pre-uprate). Thus, the turbine building feedline break initiator is increased. (Note that the pipe sections exceeding flow velocity standards are identified for inclusion in the Corrosion/Erosion Program. See 2.1.8, Flow-Accelerated Corrosion for a description of this program.)

Although all the components will operate within design limits, the loss of feedwater initiator is adjusted as follows to address equipment and operational concerns as shown in Table 2.13-5. Note that the percentage increases are based on engineering evaluation reports. Within the reports, components with reduced operating margin are identified. Judgment is used to estimate the initiating event frequency impact due to the change in operating margin for this equipment.

Table 2.13-5
Loss of Main Feed Initiator Adjustment

Basic Event	Event Description	Base Value	EPU Value	Percent Change
TIFWLOSS	Loss of Main Feedwater	5.44E-02	7.62E-02	40%
TIFLBOTB	Feedline Break in Turbine Building	2.20E-03	3.08E-03	40%

Main/Extraction Steam: The steam system piping also operates at increased flow rates for power uprate. Piping segments are analyzed and where flow velocity or vibration exceeds industry standards the appropriate initiator is adjusted. No main steam piping segments are identified as exceeding industry flow velocity standards. However, several extraction steam piping segments that exceed industry standards for flow velocity are identified. Note that all these sections are located in the turbine building.

To address the possibility of accelerated pipe wear leading to failure, these steamline break initiators are adjusted as shown in Table 2.13-6. (Note that the pipe sections exceeding flow velocity standards are identified for inclusion in the Corrosion/Erosion Program. See 2.1.8, Flow-Accelerated Corrosion for a description of this program.)

Note that the Steamline Break Through Steam Dump System is an instrumentation induced steamline break (not a piping failure). This frequency is increased due to the tighter instrument tolerances required for EPU as show in Table 2.13-6. The percentage increases are based on engineering evaluation reports. Within the reports, components with reduced operating margin are identified. Judgment is used to estimate the initiating event frequency impact due to the change in equipment operation.

Table 2.13-6
Main Steam Initiator Adjustment

Basic Event	Event Description	Base Value	EPU Value	Percent Change
TIOSLBSD	Steamline Break Through Steam Dump System	4.10E-03	4.51E-03	10%
TISLBOTB	Steamline Break in Turbine Building	4.55E-03	6.37E-03	40%

As discussed in 2.13.1.2.1.1.1, Loss of Coolant Accident, the increased post-EPU core power could cause a slight increase in the number of MSSV (and PORV) challenges due to loss of load at mid-power levels. This is addressed by increasing the likelihood of an MSSV challenge as shown in Table 2.13-1.

2.13.1.2.1.1.5 Flooding

Other than the initiators discussed above (pipe breaks), there are no substantive changes to other systems that might induce internal flooding. Therefore, the flooding impacts and initiator frequencies remain unchanged.

2.13.1.2.1.2 Anticipated Transients Without Scram (ATWS)

ATWS sequence initiation occurs under the same primary system conditions as existed prior to the power uprate except the post-EPU moderator temperature coefficient (MTC) is more negative throughout the cycle than the pre-EPU MTC (a risk benefit).

Although the MTC will be more negative, the relief capacity of the PORVs, MSSVs, and atmospheric relief valves (ARV) relative to thermal power is lower. Further, the AFW/MFW flow required post-EPU would be larger. As the more negative MTC will turn power more quickly, the equilibrium power level will be a lower fraction of full power post-EPU as compared to pre-EPU. For this reason, the likelihood of ATWS mitigation, given plant conditions are such that a trip should result, is considered to be approximately the same.

Overall, ATWS events are expected to increase slightly in proportion to the initiator frequency increases discussed above.

2.13.1.2.1.3 External Events

Generic Letter No. 88-20, Supplement No. 4 (2.13.2, Reference 7) requested each licensee to conduct an individual plant examination of external events for severe accident vulnerabilities. The following discussion provides a brief description of the GPSA external events modeling and the associated EPU impacts.

2.13.1.2.1.3.1 Seismic Events

Ginna Station is classified as a focused scope plant based on seismicity and locale. The IPEEE seismic evaluation was a seismic review of the plant performed to the plant's original design basis. This was accomplished by performing a seismic margins assessment of the safe shutdown equipment list with plant walkdowns in accordance with GL 87-02 (USI A-46, 2.13.2, Reference 4). Safe shutdown success paths were developed to identify the systems that must function to successfully shutdown and cool the reactor following the occurrence of a safe shutdown earthquake. All identified seismic vulnerabilities have been addressed with the exception of a reactor make-up water tank (RMWT) outlier.

As part of the Systematic Evaluation Program topic IX-3, it was determined that the failure of non-seismically qualified tanks could increase the water level in the Auxiliary Building, flooding the residual heat removal (RHR) sub-basement area. Since Ginna had alternative means of achieving safe shutdown without use of this system, plant configuration was determined to be acceptable. However, as part of IPEEE, it was determined that a means of mitigating a small break LOCA (SBLOCA) should be seismically qualified. Since the RHR pumps are required components needed to achieve SBLOCA mitigation, a seismic failure of the RMWT could adversely affect this capability. Ginna committed to perform a cost-benefit analysis of providing seismic qualification of this tank, or otherwise protecting the RHR/SBLOCA mitigation success path. Since the modification is not currently in place, the seismic risk evaluation performed for the EPU considers the failure of the RMWT and its impacts on plant equipment.

Post-EPU, existing equipment monitoring techniques (such as vibration analysis, thermography, oil analysis, radiography), preventive maintenance, and condition monitoring programs (such as Maintenance Rule and Erosion/Corrosion) will address any additional wear related changes that might effect seismic qualification of equipment and structures. Through trending, these programs will also identify deviations or increases in component failure rates. While the power uprate may require more frequent maintenance or more frequent replacement of components, the overall reliability of components can be maintained at the existing pre-uprate standards.

The increase in power level is not expected to effect equipment or structural response during an earthquake. The power uprate does not modify the safe shutdown pathways assumed in the seismic margins assessment. Thus, the seismic margins assessment is not impacted by the power increase.

A seismic event is likely to result in a LOOP. Power recovery in the short term is unlikely. Many non-safety-related systems may be lost. Given this, the risk increase due to a seismic event would be similar to a non-recovery grid loss. This non-recovered LOOP (a LOOP exceeding the twenty-four hour mission time) is complicated by the impact of the failure of the RMWT. The failure of the RMWT alone will not cause an immediate trip. Therefore, the dominant risks for seismic will be the combination of a LOOP with the failure of the RMWT. To generate a reasonable risk estimate, the fragility of the switchyard is assigned a fifty percent chance of failure at 0.2 g. The RMWT is considered failed if a seismically induced non-recoverable LOOP occurs. Using the subdivided frequencies from NUREG-1488 (2.13.2, Reference 5) coupled with the fragility of a non-recovered LOOP, the annual frequency of a non-recovered LOOP due to a seismic event is 1.15×10^{-4}. Specific data is shown in Table 2.13-7:

Table 2.13-7
Seismic Split Fractions

			Seismic split fraction values			
Fragility Parameters ------->		Am	1.2	0.1	0.2	
		Br	0.4	0.1	0.2	
		Bu	0.44	0.3	0.25	
		Bc	0.59	0.30	0.32	
IE No.	pga	IE Frq	1 Surrog	RMW	LOOP	LOOP and Earthquake Frequency
SMCP03	0.026	7.50E-03	0.00%	0.00%	0.00%	7.00E-13
SMCP05	0.051	1.70E-03	0.00%	1.24%	0.00%	1.67E-08
SMCP08	0.077	3.61E-04	0.00%	19.18%	0.14%	5.18E-07
SMCP12	0.115	1.43E-04	0.00%	67.93%	4.20%	5.99E-06
SMCP13	0.134	7.15E-05	0.01%	83.54%	10.55%	7.54E-06
SMCP15	0.153	7.15E-05	0.03%	92.18%	20.14%	1.44E-05
SMCP18	0.179	1.95E-05	0.07%	97.39%	36.45%	7.11E-06
SMCP20	0.204	1.95E-05	0.14%	99.13%	52.47%	1.02E-05
SMCP23	0.230	1.95E-05	0.27%	99.73%	66.88%	1.30E-05
SMCP24	0.242	9.75E-06	0.35%	99.84%	72.42%	7.06E-06
SMCP26	0.255	9.75E-06	0.46%	99.91%	77.60%	7.56E-06
SMCP28	0.281	7.43E-06	0.73%	99.97%	85.59%	6.36E-06
SMCP31	0.306	7.43E-06	1.08%	99.99%	90.80%	6.75E-06
SMCP33	0.332	3.51E-06	1.54%	100.00%	94.33%	3.31E-06
SMCP36	0.357	3.51E-06	2.07%	100.00%	96.48%	3.39E-06
SMCP38	0.383	3.51E-06	2.74%	100.00%	97.88%	3.44E-06
SMCP41	0.408	3.51E-06	3.48%	100.00%	98.70%	3.47E-06
SMCP43	0.434	1.69E-06	4.36%	100.00%	99.22%	1.68E-06
SMCP46	0.459	1.69E-06	5.30%	100.00%	99.53%	1.68E-06
SMCP49	0.485	1.69E-06	6.38%	100.00%	99.72%	1.69E-06
SMCP51	0.510	1.69E-06	7.51%	100.00%	99.83%	1.69E-06
SMCP55	0.548	1.03E-06	9.37%	100.00%	99.92%	1.03E-06
SMCP59	0.587	1.10E-06	11.46%	100.00%	99.96%	1.10E-06
SMCP63	0.625	1.10E-06	13.63%	100.00%	99.98%	1.10E-06
SMCP66	0.663	1.10E-06	15.92%	100.00%	99.99%	1.10E-06
SMCP74	0.740	8.92E-07	20.81%	100.00%	100.00%	8.92E-07
SMCP78	0.778	4.46E-07	23.31%	100.00%	100.00%	4.46E-07
SMCP82	0.816	4.46E-07	25.83%	100.00%	100.00%	4.46E-07
SMCP92	0.918	5.03E-07	32.62%	100.00%	100.00%	5.03E-07
SMC1P5	1.508	1.75E-06	64.96%	100.00%	100.00%	1.75E-06
Total Frequency of a non-Recoverable LOOP due to Seismic---->						*1.15E-04*

For a full understanding of the importance of the RMWT in seismic events, a sensitivity evaluation is performed both with and without the failure of the RMWT. The results are shown in Table 2.13-8:

Table 2.13-8
Seismic Risk Sensitivity

Boundary Conditions	CDF		% Change	Delta
	Pre-Uprate	Post-Uprate		
Non-Recoverable LOOP with RMW Tank Failure	6.36E-06	6.37E-06	0.2%	1.17E-08
Non-Recoverable LOOP	3.47E-07	3.48E-07	0.3%	1.17E-09

Although the fragility of Ginna equipment is not impacted by EPU, seismic events do challenge mitigation systems (for example AFW). Since the time available for mitigation response is shorter due to increased decay heat associated with EPU, seismic risk increases post-EPU.

As shown in Table 2.13-8, the additional delta increase in CDF due to seismic events without resolving the RMWT outlier is approximately 1×10^{-8}. Even considering the seismic impact, the Ginna total CDF is less than 1×10^{-4} and the EPU delta increase is less than 1×10^{-5}. Therefore, this sensitivity evaluation demonstrates that the seismic issue is not likely to impact the acceptability of EPU, even considering the RMWT outlier.

2.13.1.2.1.3.2 Fire

The fire portion of the Ginna Station Individual Plant Examination for External Events (IPEEE) response was performed using a combination of two approaches, the Electric Power Research Institute Fire Induced Vulnerability Evaluation Methodology (FIVE, 2.13.2, Reference 6), and fire PSA. The FIVE methods were used for progressive screening of most fire areas, and PSA methods were used for the analysis of non-screened areas. Two distinct analysis phases were performed.

In the first phase, fire areas were evaluated to determine if they contained Appendix R safe-shutdown components (or cables) and if a fire in that area would cause a demand for safe shutdown functions. If not, these fire areas were screened out. Fire zones, which are subsets of fire areas, were similarly screened out if they met the above criteria and had no credible potential for fire spread into other fire areas or zones. This process conformed with Phase I of the FIVE methodology.

In the second phase, all fire zones not screened out in the first phase were evaluated using more sophisticated quantitative models consistent with NUREG/CR-2300 and NUREG/CR-4840 (2.13.2, References 11 and 12). A comprehensive set of fire scenarios was developed for each zone, depending on the physical configuration of the zone and the locations of combustibles, equipment, and cables. The internal events PSA models and data were modified to incorporate the fire impacts and to include potential recovery actions.

Fire events are a significant contributor to the overall CDF at Ginna Station. For this analysis, fire related operator recovery actions are re-analyzed to ensure suitable environmental conditions exist throughout the operator travel path and at the equipment to be manipulated. In addition, various performance shaping factors, stress, and time available are adjusted to reflect fire conditions. (See 2.13.1.2.1.6.3, Operator Actions, *Degraded Case* for a discussion of fire related recovery actions.)

2.13.1.2.1.3.3 High Winds, Floods, and Other External Events

The IPEEE Other Events Analysis Screening for Ginna Station used the screening approach described in Generic Letter 88-20 (2.13.2, Reference 7). The IPEEE submittal reviewed the plant design for consistency with the acceptance criteria in terms of high winds, onsite storage of hazardous materials, and offsite developments.

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 1970's the NRC instituted the Systematic Evaluation Program (SEP) to review the designs of older operating nuclear power plants and confirm 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) (2.13.2, Reference 8). The report 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 a significant addition to safety margin. The current UFSAR incorporates the SEP review into the Current Licensing Basis.

The IPEEE found no high winds, floods, or offsite industrial facility accidents that significantly altered the estimate of either the core damage frequency or the distribution of containment release categories. The IPEEE concluded that a detailed PSA to address high winds, onsite storage of hazardous materials, and offsite developments was not required.

In the calculation of LOOP frequency, the "IPEEE events" that impact LOOP are implicitly addressed in the LOOP frequency development. The frequency of a LOOP event at Ginna, caused by the loss of the grid or the main switchyard, is based on industry LOOP experience as compiled by EPRI. The EPRI database includes LOOP events due to all causes, including "IPEEE events." Since the EPRI data includes plants from around the country, it can be assumed to encompass locations where the likelihood of an "IPEEE event" is both greater than, and less than, that for Ginna. The EPRI generic industry data is updated with Ginna-specific data using Bayesian techniques. This accounts for the Ginna-specific likelihood of "IPEEE events."

The EPU does not affect high wind, flood, or offsite industrial accident frequencies; nor does it affect applicable protective features such as missile or flood barriers. However, as with seismic events, these external events are likely to result in a LOOP. Power recovery in the short term is unlikely. In addition, some non-safety-related systems may be lost. Given this, the risk increase due to these "IPEEE events" would be similar to a non-recovery grid loss. As part of the sensitivity analysis, the sensitivity of LOOP initiating event frequencies for EPU are examined. This provides a gross estimate on the impact EPU will have on "IPEEE events." The change in CDF associated with an unrecovered LOOP is shown in Table 2.13-9.

Table 2.13-9
Non-Recovered LOOP Impact

Boundary Conditions	CDF		% Change	Delta
	Pre-Uprate	Post-Uprate		
Non-Recoverable LOOP	3.47E-07	3.48E-07	0.3%	1.17E-09

Although the ability of Ginna equipment to mitigate "other external events" is not impacted by EPU, "other external events" do challenge mitigation systems (for example AFW). Since the time available for mitigation response is shorter due to increased decay heat associated with EPU, the "other external events" risk increases post-EPU.

This sensitivity evaluation demonstrates that High Winds, Floods, and Other External Events are not likely to impact the acceptability of EPU.

2.13.1.2.1.4 Turbine Missile Generation

As stated earlier (2.13.1.2.1.1.4, Transient Initiators), the generator operating margin will be reduced post-EPU. As a result of this decreased margin, it is anticipated that the loss of load trip frequency will increase (see basic event RCAZTRIPLL in Table 2.13-1). For this trip, the steam admission valves to the turbine must quickly close to avert a turbine overspeed which could lead to a turbine blade ejection.

Post-EPU, the possibility of a turbine missile due to an increase in overspeed challenges is exacerbated by two other EPU related factors:

- An increase in operating steam temperature also raises the turbine blade operating temperature. The likelihood of blade crack formation and propagation increase at higher temperatures.
- Additional stored energy (due to higher steam temperatures) within the turbine high to low pressure stages will result in higher rotor speeds if the high pressure governor or stop valves close successfully but a low pressure stop or intercept valve fails to close on a trip.

The change in turbine overspeed missile generation likelihood is shown in Table 2.13-10. The actual change in the likelihood of a turbine missile is considered less than 7.5 percent; a bounding value is used to address the uncertainties associated with the previously mentioned EPU factors.

Table 2.13-10
Turbine Missile Likelihood Changes

Basic Event	Event Description	Base Value	EPU Value	Bounding Percent Change
TMAZTBMISSA	Probability of generating a turbine missile given a design overspeed (120%) event	5.00E-04	5.38E-04	7.50%
TMAZTBMISSB	Probability of generating a turbine missile given an intermediate overspeed event (132%) event	9.00E-03	9.68E-03	7.50%

Due to the various turbine missile trajectories and possible targets, the likelihood of core damage given a turbine missile has a great deal of uncertainty. To address this large uncertainty, there is considered a fifty percent chance of core damage given a turbine missile. This is considered a bounding likelihood. Table 2.13-11 shows the pre/post-EPU CDF associated with this issue.

Table 2.13-11
Turbine Missile Risk

Risk Contributor	CDF		% Change	Delta
	Pre-Uprate	Post-Uprate		
Turbine Missile Contribution	9.73E-07	1.30E-06	33%	3.23E-07

This conservative delta risk of turbine missiles is included in the overall CDF/LERF changes mentioned in 2.13.1.2.4.4, Level 1 and Level 2 Results, Table 2.13-20. This accounts for approximately four percent of the total delta risk associated with EPU.

2.13.1.2.1.5 Shutdown Operations Risk

The Residual Heat Removal System (see Section 2.8.4.4) evaluation shows the ability to achieve cold shutdown in a reasonable time frame exists at EPU conditions. Since the decay heat levels will be higher at power uprate conditions, all cooldown times are increased. Although it may take more time to achieve cold shutdown, the increased cooldown time is not a safety issue. The design analysis also determined that the requirements to achieve cold shutdown (200°F) with two residual heat removal/CCW trains in service or using a single train Appendix R cooldown are achievable.

At the higher thermal power of EPU, it will take either a longer amount of time to reach Mode 5 and/or more equipment. For example, to achieve Mode 5 using steam generator water solid cool down post-EPU, new spool pieces will be required. Also, it could take two ARVs to reach Mode 5 within seventy-two hours under design basis conditions. The success criteria for the PSA models are based on twenty-four hours with a trend to success. Beyond that time the recovery possibilities increase due to the availability of additional manpower and equipment resources. Therefore, these longer term issues introduced by EPU do not significantly impact

CDF or LERF.

During an outage, Ginna Station closely monitors shutdown safety parameters including decay heat removal, vital power, reactivity control, containment closure, and the RCS. There are no EPU changes to the primary system, instrumentation for reduced inventory operation, or available equipment/methods to mitigate a loss of residual heat removal cooling. In addition, existing reduced inventory procedures and administrative controls minimize the likelihood of uncovering the core while ensuring a defense-in-depth response is available if needed.

The GPSA shutdown related initiating events are Loss of Residual Heat Removal, Boron Dilution Event, Loss of Coolant Accident, and RCS Overpressurization. The power uprate does not increase the frequency of these initiators, but the increase in temperature and the increase in decay heat will decrease the time available for operators to respond to them. To address this concern, shutdown related operator action failure likelihoods are re-evaluated based on the shorter times available.

Note that the spent fuel pool design analysis shows that the additional decay heat associated with the EPU results in increased time after shutdown to commence refueling operations. The analysis also shows that on a loss of spent fuel pool cooling, makeup water is available (from either the refueling water storage tank or the CVCS hold-up tanks) to maintain the pool volume and remove decay heat. Therefore, no additional risk impacts are associated with the spent fuel pool post-EPU.

2.13.1.2.1.6 Model Attributes

This section discusses several key GPSA model attributes and how the EPU is expected to impact those attributes.

2.13.1.2.1.6.1 Functional / System Level Success Criteria

A detailed review is performed to identify the effect of the increase in thermal power level on the system success criteria credited in the GPSA model. These success criteria specify the requirements of the plant systems to address critical safety functions. These safety functions are:

REACTIVITY CONTROL

There are no changes in reactivity control methods or effectiveness. As mentioned earlier, the MTC will remain more negative throughout the post-EPU fuel cycle than the pre-EPU MTC. This improves the likelihood of successful ATWS mitigation (not credited in the analysis). Otherwise, there are no changes to the reactivity control success criteria.

RCS PRESSURE CONTROL

There are no changes in manner of operation, pressure, or components that affect pressure control success criteria.

RCS AND CORE HEAT REMOVAL

Increased decay heat, associated with higher power, degrades bleed and feed (BAF). Without charging available, two power operated relief valves are required for successful bleed and feed post-EPU (a single power operated relief valve was sufficient pre-EPU). With charging, bleed and feed is possible with a single PORV, however, time available to initiate bleed and feed is reduced (see 2.13.1.2.1.6.2, Insights from Thermo Hydraulic Analysis, Table 2.13-12).

RCS and core heat removal can be lost due to boron precipitation following a medium or large break LOCA. The EPU does not significantly increase the likelihood of boron precipitation. As a result of the EPU, the boron concentration in the refueling water storage tank (RWST) and the accumulators will be increased. This, along with other EPU related changes (increased core decay heat levels, for example), reduces the time to reach the solubility limit for boron in the core during some large and intermediate size LOCAs from twenty hours to 6.5 hours. Reaching the solubility limit within the core can result in boron precipitation on the fuel assemblies which reduces heat transfer rates and leads to core damage. To preclude this, the Ginna Emergency Procedures (EOPs) direct operators to re-establish cold leg safety injection (that is, simultaneous injection) at nineteen hours into the event. As a result of the EPU, the Ginna EOPs will be revised to instruct operators to re-establish cold leg safety injection no later than six hours after the termination of safety injection in the cold leg to prevent boric acid precipitation.

The equipment needed to provide safety injection re-initiation is unaffected by EPU. Therefore, the only potential risk increase would be from an increase in the probability of the human action for operators failing to re-initiate safety injection. A review of the Ginna EOPs indicates the longest time to reach the crucial procedural step would be approximately 105 minutes into the event. Operators currently have approximately seventeen hours to complete this step. Post-EPU, operators will have approximately four hours to complete this step. Given the long time available in both cases, the human action failure likelihood would be driven by execution rather than diagnoses. The methodology used to calculate human error failure rates in the GPSA produces no change of the failure rate for this action when the diagnoses time is over sixty minutes. Thus, the increase in boron concentration and associated reduction in time to re-initiate safety injection following a large or medium sized LOCA produces no increase in the risk of core damage.

The number of components required to support at-power RCS and core heat removal using MFW will not change post-EPU. The condensate booster pump and MFW pumps are being modified to ensure that the number of condensate booster pumps pre-EPU required to support a given power level is the same number required post-EPU.

RCS INVENTORY CONTROL

The pressurizer level control program will change with EPU. Currently, the pressurizer level varies from thirty-five to fifty percent in correlation with zero to one-hundred percent power. After EPU, the pressurizer level will vary from twenty percent to sixty percent for the corresponding zero to one-hundred percent power levels.

Although unlikely, it is possible that the higher water level could lead to more PORV/primary safety challenges. There will be less steam volume to react to pressure changes and less margin recover from uncontrolled charging. This impact is considered represented by the increase to event RCAZTRIPPV (see 2.13.1.2.1.1.1, Loss of Coolant Accident (LOCA), Table 2.13-1).

2.13.1.2.1.6.2 Insights from Thermo Hydraulic Analysis

Note: PCTRAN is used for most thermo hydraulic evaluations. For a description of this software see 2.13.1.2.5.4, Software.

One of the main impacts of the EPU is a reduction in the time available for operator response due to the increased decay heat. The GPSA includes more than one-hundred post-trip human actions; less than half of these human actions were impacted by the EPU. The types of human actions not impacted by EPU are:

- *The time to recover cooling to the RCP seals.* The time to restore cooling to the seals is not impacted by the slight increase in T_{avg}.
- *The time to trip RCPs prior to catastrophic seal failure.* The time to trip the RCPs is not impacted by the slight increase in T_{avg}.
- *The time to manually open the containment sump MOVs (swap to recirculation) prior to loss of net positive suction head to the safety injection equipment on a LOCA.* This timing is largely driven by the number of pumps running and the RWST volume. The number of pumps running depends on the time to depressurize the RCS and the time to pressurize containment. This is driven by the size of the break and the pressure of the RCS. Since the pressure of the RCS does not change, the time difference between pre/post-EPU is negligible.
- *The time available to depressurize the steam generator to minimize leakage during a SGTR.* This is driven by RCS pressure which does not change.
- *Time to align TSC batteries prior to depletion of the station batteries.* The time available for this human action is a combination of the battery depletion time and time to steam generator dry out. Although EPU does impact the time to steam generator dry out, the amount of time available is so large compared to the implementation time that there is no change in human action likelihood.
- *Operators manually secure RWST drain-down due to a hot short.* The time is based on the line size and RWST inventory. This does not change with EPU.

Although not an all inclusive list, this list does provide samples of the types of human actions not impacted by EPU. Human actions that are impacted by EPU fall into two broad categories: inventory control and decay heat removal.

INVENTORY CONTROL

The time to isolate a LOCA (for example, stuck open PORV) or align safety injection is impacted by the EPU. The impact on inventory control losses is not as severe as the impact on losses of decay heat removal. LOCA inventory losses are driven by RCS pressure. So initially the inventory losses are the same pre/post-EPU. As the LOCA progresses with no injection, the EPU will keep the pressure slightly higher and experience core damage first due to the higher decay heat. With two PORVs stuck open, the recovery time available changes from seventy-three minutes (pre-EPU) to sixty-four minutes (post-EPU). This is a twelve percent reduction in the recovery time available. The impacts on decay heat removal are typically much larger.

DECAY HEAT REMOVAL

EPU has direct effects on decay heat removal. A few important aspects are:

- The time to restore cooling on a total loss of decay heat removal varies considerably based on the amount of water in the steam generator at the time of the trip and the status of the RCPs. Although the RCPs increase the rate of heat transfer to the steam generators, the RCPs become a liability when the steam generators are not receiving cooling due to the additional heat added by the pumps. Heat from the RCPs is a noticeable fraction of the decay heat, especially later in the accident response. The emergency operating procedures direct that the RCPs be tripped during total loss of feedwater scenarios. Although the RCPs are typically tripped, the RCPs actually act to dampen the delta risk increase associated with EPU because the constant heat load from the RCPs dampens the change in overall heating of the RCS pre-EPU versus post-EPU.

- The RCS operates at a slightly higher T_{avg} post-EPU, this reduces the post-EPU recovery time.

- The steam generator operates at a lower temperature. This helps the post-EPU recovery times.

Table 2.13-12 presents some of the key total loss of decay heat removal scenarios and possible human actions. The table also lists the RCP status, timing data, and failure criteria.

Table 2.13-12
Decay Heat Time Changes

Sample Human Action	Description	Base Time (Min)	EPU Time (Min)	Reduction in Time Available	RCP Tripped?	SG Water Level at Trip (% NRWL)	Failure Criteria
AFHFDBLOWD	OP Fails to Isolate SG Blowdown Manually	14	10	31%	On lack of NPSH	17%	Half the Time until SG Dryout
AFHFDTDAFW	OP Fails to Manually Open Steam Valves to TDAFW Pump (No Fire)	25	17	31%	On lack of NPSH	17%	SG 10% WRWL
FSHFDAFWXX-1	OP Fails to Manually Align and Start TDAFW Pump given 17% NR SG Water Level Trip and RCPs Running - Fire Event	28	19	31%	On lack of NPSH	17%	SG Dryout
RCHFPX1BAF	OP Fails to Align BAF given a Single PORV and No Charging Pumps	32	15	53%	1 min after PORV Opened	17%	BAF can not be achieved with one PORV
RCHFPX4BAF	OP Fails to Align BAF given Both PORVs Open	42	28	33%	1 min after PORV Opened	17%	BAF can not be achieved with two PORVs
RCHFPX3BAF	OP Fails to Align BAF given a Single PORV and 75gpm Charging Flow	46	25	46%	1 min after PORV Opened	17%	BAF can not be achieved with one PORV and 75gpm Charging
AFHFDSTART	OP Fails to Manually Start MDAFW Pump with No Auto Start Signal	84	65	23%	On lack of NPSH	17%	Core Uncovery
DGHFDRELAY	OP Fails to Start DG After Fire in Relay Room Fails all Auto Start Logic	85	64	25%	At T=0 (i.e. LOOP)	52%	Half the Time to Core Uncovery
FSHFDAFWXX-3	OP Fails to Manually Align and Start TDAFW Pump given 52% NR SG Water Level Trip and NO RCPs Running - Fire Event	88	64	28%	At T=0 (i.e. LOOP)	52%	Time to SG Dryout
ACHFDPWR51	OP Fails to Utilize Offsite Power Circuit 751 within 1 Hour	118	97	18%	On lack of NPSH	52%	Core Uncovery

2.13.1.2.1.6.3 Operator Actions

In general, the higher decay heat level associated with power uprate reduces the total response time (time available) for the operators to complete recovery actions. For certain recovery actions, the reduced total response time can increase the human action failure likelihood or even fail the action for insufficient time.

Based on unresolved Peer Review comments (see 2.13.1.2.5.1, Model Peer Review) significant GPSA Revision 5.0 human actions were reassessed using the EPRI HRA calculator and EPRI methodology. The reassessment provides consistent and credible human action failure likelihoods with an added benefit of comparability between the pre/post-uprate results.

HUMAN ACTION METHODOLOGY

As part of the EPU a significant effort is made to update the human action analysis. All human actions contained within the GPSA are reviewed to address the impacts associated with adverse environmental conditions and accessibility limitations. For each operator action, the travel path route and relevant equipment locations are identified. Using this information, each initiating event with an environmental impact is evaluated for its effect on each operator action. Based on this evaluation, the human action is classified as normal, degraded, or failed as appropriate.

- *Normal Case* – The normal human action failure rate was used for all cases except for actions that are either failed or degraded due to impacts associated with adverse environmental conditions or accessibility limitations. Two methods are used to calculate the failure probability associated with these human actions. Older calculations were developed by contractor support from Tenera, Inc., and EQE/PLG, Inc. These calculations used Human Error Probabilities (HEPs) to support the GPSA Revision 4.3. This methodology is based on the NRC's Accident Sequence Evaluation Program (ASEP) methodology. However, the results were considered conservative, resulting in slightly higher values for many of the HEPs. Because of this conservatism, the EPRI HRA Calculator software is used for risk significant human actions in Revision 5.0.

- *Degraded Case* – An adverse environment degrades the operator ability to perform recovery actions, and consequently, increases the human action failure rate. The EPRI HRA calculator is used to determine the highest possible failure rate given the normal parameters selected for operator actions at Ginna Power Station. Considering smoke, water, degraded lighting, heat, and increased stress, the HRA calculator parameters are adjusted to reflect these conditions. Selections are made to bound all actions within the model. This includes increasing stress levels to extreme, establishing high dependency levels, selecting higher failure rates for most Performance Shaping Factors, increasing time to perform the action, and assessing the nominal number of critical steps for a local action. All these increases result in a failure probability of 0.097. Therefore, 0.10 is conservatively used for the generic degraded case. EPU actions also use this generic degraded case, except high dependency levels are changed to complete dependency with a resulting failure likelihood of 0.16. If the human action in the normal case (above) already had a complete dependency, then the degraded case human action is failed. Note: Risk significant human actions are individually degraded on a case by case basis.

- *Failed Case* – Initiating events with either failed fire suppression or a large flood volume are assumed to prevent the operator from performing recovery actions. Failed case human actions are set to one.

<u>Human Action Timing and Impact</u>

When applicable, the recovery action total response time comes from the pre-uprate and post-uprate thermo hydraulic analyses (pre-initiator operator actions are not time dependent and time available for some post-initiator operator actions did not change as a result of the power uprate).

The operator actions where total response times changed as a result of the power uprate are listed in Table 2.13-13. The table provides the basic event, its description, the total response times pre/post-EPU, and the failure likelihoods pre/post-EPU. A later table (Table 2.13-14) shows the increase in the delta risk of EPU as the human action failure likelihood changes.

Table 2.13-13 (1 of 4)
Human Action Timing and Values Pre/Post-EPU

Basic Event	Event Description	Base Time (Min)	EPU Time (Min)	Base Value	EPU Value
ACAZDLOSP1	Failure to Restore Offsite Power Within 1 Hr	60	Note 1	5.82E-01	6.41E-01
ACAZDLOSP2	Failure to Restore Offsite Power Within 2.25 Hrs	150	Note 1	4.05E-01	4.46E-01
ACAZDLOSP5	Failure to Restore Offsite Power Within 5 Hrs	420	Note 1	2.21E-01	2.43E-01
ACAZDLOSP6	Failure to Restore Offsite Power Within 6 Hrs	480	Note 1	1.82E-01	2.00E-01
ACAZLOSP10	Failure to Restore Offsite Power Within 10 Hrs	900	Note 1	9.30E-02	1.02E-01
ACHFDPWR51	OP Fails to Utilize Offsite Power Circuit 751 within 1 Hour	118	97	1.61E-02	2.90E-02
ACHFDPWR51-DF	OP Fails to Utilize Offsite Power Circuit 751 within 1 Hour during a Fire/Flood	118	97	1.00E-01	1.60E-01
ACHFDPWR67	OP Fails to Utilize Offsite Power Circuit 767 within 1 Hour	118	97	1.61E-02	2.90E-02
ACHFDPWR67-DF	OP Fails to Utilize Offsite Power Circuit 767 within 1 Hour during a Fire/Flood	118	97	1.00E-01	1.60E-01
AFHFDBLOWD	OP Fails to Isolate SG Blowdown Manually	14	10	1.00E-01	1.00E+00
AFHFDSTART	OP Fails to Manually Start MDAFW Pump with No Auto Start Signal	84	65	1.10E-05	1.60E-04
AFHFDSTART-DF	OP Fails to Manually Start MDAFW Pump with No Auto Start Signal during a CR Fire w/o CR Evac	84	65	1.60E-05	1.70E-04

Table 2.13-13 (2 of 4)
Human Action Timing and Values Pre/Post-EPU

Basic Event	Event Description	Base Time (Min)	EPU Time (Min)	Base Value	EPU Value
AFHFDSUPPL-3	OP Fails to Supply Alternate Sources of Water to AFW	60	51	5.30E-04	7.20E-04
AFHFDSUPPL-3-DF	OP Fails to Supply Alternate Sources of Water to AFW during a Fire/Flood	60	51	1.01E-03	2.60E-03
AFHFDTDAFW	OP Fails to Manually Open Steam Valves to TDAFW Pump (No Fire)	25	17	1.00E-01	3.72E-01
AXHFDCITYW	OP Fails to use City Fire Water for SAFW per ER-AFW.1	84	65	7.64E-04	3.93E-03
AXHFDCITYW-DF	OP Fails to use City Fire Water for SAFW per ER-AFW.1 during a Fire/Flood	84	65	5.90E-03	2.42E-02
AXHFPSAFWX-2	OP Fails to Align and Start SAFW Pumps C or D	84	65	7.11E-05	9.72E-04
AXHFPSAFWX-2-DF	OP Fails to Align and Start SAFW Pumps C or D CR Fire w/o CR Evac	84	65	2.43E-04	1.88E-03
AXHFPSAFWX-3	OP Fails to Align and Start SAFW Pumps C or D (during ER-FIRE.4 or .5)	84	65	1.50E-03	4.50E-03
AXHFPSAFWX-3-DF	OP Fails to Align and Start SAFW Pumps C or D during a CR Fire w/o CR Evac	84	65	1.00E-01	1.60E-01
CCHFD738AB-DF	OP Fails to Manually Open CCW MOVs 738A/B to the RHR HX when Power is Lost to the MOVs during a Fire/Flood	295	296	1.00E-01	1.60E-01
DGHFDCITYW	OP Fails to Connect City Water to DG Cooling per ER-D/G.2	85	64	1.00E-04	1.38E-03
DGHFDCITYW-DF	OP Fails to Connect City Water to DG Cooling per ER-DG and ATTACH-2.4 during a Fire/Flood	85	64	3.00E-03	7.06E-03
DGHFDRELAY	OP Fails to Start DG After Fire in Relay Room Fails all Auto Start Logic	85	64	1.00E-01	1.87E-01
FSHFDAFWXX-1	OP Fails to Manually Align and Start TDAFW Pump given 17% NR SG Water Level Trip and RCPs Running - Fire Event	28	19	3.81E-01	1.00E+00
FSHFDAFWXX-2	OP Fails to Manually Align and Start TDAFW Pump given 27% NR SG Water Level Trip and RCPs Running - Fire Event	47	34	8.59E-02	2.25E-01
FSHFDAFWXX-3	OP Fails to Manually Align and Start TDAFW Pump given 52% NR SG Water Level Trip and RCPs Running - Fire Event	88	64	2.73E-02	4.76E-02
FSHFDDGAXA	SS Fails to Strip Bus 18 Loads and Manually Close Breaker for DG A per ER-FIRE.3	85	64	3.86E-03	8.60E-03

Table 2.13-13 (3 of 4)

Human Action Timing and Values Pre/Post-EPU

Basic Event	Event Description	Base Time (Min)	EPU Time (Min)	Base Value	EPU Value
FSHFDDGAXA-DF	SS Fails to Strip Bus 18 Loads and Manually Close Breaker for DG A per ER-FIRE.3 during a Fire/Flood	85	64	1.00E-01	1.60E-01
FSHFDDGAXX	STA Fails to Start DG A per Attachment 2 to ER-FIRE.1, .2 or .5	85	64	3.23E-04	1.78E-03
FSHFDDGAXY	CRF Fails to Strip Bus 18 Loads and Manually Close Breaker for DG A per Attachment 1 to ER-FIRE	85	64	2.55E-04	2.08E-03
FSHFDDGAXZ	CO Fails to Strip Bus 14 Loads and Manually Close Breaker for DG A/SW A per Attachment 4 to ER-FIRE	85	64	8.08E-05	1.61E-03
FSHFDDGAXZ-DF	CO Fails to Strip Bus 14 Loads and Manually Close Breaker for DG A/SW A during a Fire/Flood	85	64	3.91E-03	1.05E-02
FSHFDDGBXX	STA Fails to Start DG B and SW Pump per ER-FIRE.4	85	64	6.44E-03	1.30E-02
FSHFDMDAFW	OP Fails to Manually Start MDAFW Pump with No Auto Start Signal	84	65	1.00E-01	3.58E-01
FSHFDOFFPR	OP Fails to LOCALLY Re-Close Supply Bkrs for 52/16 and/or 52/14 after Hot Short Event Transfers the Bkrs Open	85	64	1.03E-01	1.72E-01
FSHFDOFFPR-DF	OP Fails to LOCALLY Re-Close Supply Bkrs for 52/16 and/or 52/14 after Hot Short Event Transfers the Bkrs Open given a fire/flood	85	64	2.20E-01	3.67E-01
FSHFDRPORV	OP Fails to De-Energize PORV Control Circuit during a Fire/Flood	73	64	1.34E-03	3.97E-03
FSHFDRPORV-DF	OP Fails to De-Energize PORV Control Circuit during a Fire/Flood that directly degrades the action	73	64	1.00E-01	1.60E-01
IAHFDCSA04	OP Fails to Place an Air Compressor in Service per E-0 STEP 33.C OR AP-IA.1	60	46	1.63E-03	2.41E-03
IAHFDCSA04-DF	OP Fails to Place an Air Compressor in Service during a CR Fire w/o CR Evac	60	46	3.70E-03	5.60E-03
IFHFDAFWSW	OP Fails to Locally Align SW to TDAFW Suction Following CR Evac for Fires and Floods (ER-FIRE)	148	115	7.50E-03	9.50E-03
IFHFDAFWSW-DF	OP Fails to Locally Align SW to TDAFW and SAFW Suction Following CR evac for Fire	148	115	1.00E-01	1.60E-01
MFHFPMF100	OP Fails to Correctly Re-Establish MFW	84	65	9.73E-05	9.59E-04

Table 2.13-13 (4 of 4)
Human Action Timing and Values Pre/Post-EPU

Basic Event	Event Description	Base Time (Min)	EPU Time (Min)	Base Value	EPU Value
RCHFDCD0SS	OP Fails to Cooldown to RHR After SI Fails (Injection or Recirc) - SSLOCA	69	60	1.40E-03	3.60E-03
RCHFDCD0SS-DF	OP Fails to Cooldown to RHR After SI Fails (Inject/Recirc) - SSLOCA during a Fire/Flood	69	60	7.32E-03	1.20E-02
RCHFDPLOCA	OP Fails To Close PORV Block Valve (515/516) To Terminate LOCA	73	64	6.44E-03	8.63E-03
RCHFDRHRSB-DF	OP Fails to Rapidly Depressurize to RHR (or Use AFW Long-Term) during a Fire/Flood	60	60	1.00E-01	1.60E-01
RCHFPX1BAF	OP Fails to Align BAF given a Single PORV and No Charging Pumps	32	15	1.33E-02	1.00E+00
RCHFPX1BAF-DF	OP Fails to Align BAF given a Single PORV and No Charging Pumps during a Fire/Flood	32	15	4.24E-02	1.00E+00
RCHFPX3BAF	OP Fails to Align BAF given a Single PORV and 75gpm Charging Flow	46	25	1.14E-02	1.33E-02
RCHFPX3BAF-DF	OP Fails to Align BAF given a Single PORV and 75gpm Charging Flow during a Fire/Flood	46	25	3.01E-02	1.00E+00
RCHFPX4BAF	OP Fails to Align BAF given Both PORVs Open	42	28	3.83E-03	1.29E-02
RCHFPX4BAF-DF	OP Fails to Align BAF given Both PORVs Open during a Fire/Flood	42	28	1.33E-02	1.48E-02
RHHFDREC00	OP Fails to Recover RHR System before onset of boiling (< 1 Hour)	60	Note 1	5.00E-01	6.33E-01
RHHFDREC01	OP Fails to Recover RHR System before onset of boiling (1-4 Hours)	240	Note 1	1.00E-01	1.27E-01
RHHFDREC01-DF	OP Fails to Recover RHR System within 4 hrs before onset of boiling, Fire/Flood (1-4 Hours)	240	Note 1	1.00E-01	1.60E-01
RHHFDREC04	OP Fails to Recover RHR System before onset of boiling (4-12 Hours)	720	Note 1	1.00E-02	1.27E-02
RHHFDREC24	OP Fails to Recover RHR System before onset of boiling (12-24 Hours)	1440	Note 1	5.00E-03	6.33E-03
RHHFDRECXX	OP Fails to Recover RHR System before onset of boiling (> 24 Hours)	1440	Note 1	1.00E-03	1.27E-03
SWHFDSTART	OP Fails to Start a SW Pump from the CR Given a Fire or a Loss of 480VAC Bus	60	51	3.04E-04	7.70E-04
SWHFDSTART-DF	OP Fails to Start a SW Pp from the CR Given CR Fire that is suppressed (No CR Evac)	60	51	5.11E-03	5.30E-03
XXHFGNOAFW	OP Fails to Diagnose Loss of AFW	84	65	5.32E-06	1.00E-05

Note 1: These non-recovery probabilities are increased using the change in decay heat pre-EPU versus post-EPU.

The impacts of the human action deltas on CDF are shown in Table 2.13-14. For each basic event the table shows the base and EPU human action value, the EPU CDF increase, and the percent of the overall uprate CDF increase. Note: As human actions are used redundantly, the sum of the CDF contribution listed in this table is actually greater than the total human action contribution to the CDF increase associated with the EPU. Also note that human actions having a CDF increase less than 10^{-8} are not shown.

Table 2.13-14 (1 of 3)
Human Action Impacts Pre/Post-EPU

Basic Event	Event Description	Base Value	EPU Value	CDF Increase due to HA	% of Uprate Increase
RHHFDREC00	OP Fails to Recover RHR System before onset of boiling (< 1 Hour)	5.00E-01	6.33E-01	1.10E-06	14.51%
ACAZLOSP10	Failure to Restore Offsite Power Within 10 Hours	9.30E-02	1.02E-01	7.44E-07	9.79%
XXHFGNOAFW	OP Fails to Diagnose Loss of AFW	5.32E-06	1.00E-05	6.10E-07	8.03%
RHHFDREC04	OP Fails to Recover RHR System before onset of boiling (4-12 Hours)	1.00E-02	1.27E-02	4.62E-07	6.07%
FSHFDAFWXX-1	OP Fails to Manually Align and Start TDAFW Pump given 17% NR SG Water Level Trip and RCPs Running - Fire Event	3.81E-01	1.00E+00	4.59E-07	6.04%
RHHFDREC01	OP Fails to Recover RHR System before onset of boiling (1-4 Hours)	1.00E-01	1.27E-01	4.45E-07	5.86%
RRHFDRECRC-L	OP Fails to shift RHR to Sump Recirc with 2 CS Pumps Initially Running - small, med, & large LOCAs	4.39E-02	4.39E-02	3.60E-07	4.73%
AXHFPSAFWX-2	OP Manually Trips Reactor prior to Loss of MFW during ER-FIRE.1, .2, or .3 fire or flood	7.11E-05	9.72E-04	3.39E-07	4.46%
AFHFDBLOWD	OP Fails to Shift SI System to Recir during a LOCA	1.00E-01	1.00E+00	3.37E-07	4.43%
FSHFDAFWXX-2	OP Fails to Provide Cooling to TDAFW Lube Oil from Diesel Fire Pump	8.60E-02	2.25E-01	2.08E-07	2.74%
SWHFDSTART	OP Fails to Manually Open RHR Suction/Injection Valves	3.04E-04	7.70E-04	1.97E-07	2.60%
RHHFDREC24	OP Fails to Implement Emergency Boration	5.00E-03	6.33E-03	1.94E-07	2.56%
FSHFDDGAXY	CRF Fails to Strip Bus 18 Loads and Manually Close Breaker for DG A per Attachment 1 to ER-FIRE	2.55E-04	2.08E-03	1.90E-07	2.50%

Table 2.13-14 (2 of 3)
Human Action Impacts Pre/Post-EPU

Basic Event	Event Description	Base Value	EPU Value	CDF Increase due to HA	% of Uprate Increase
FSHFDDGAXZ	CO Fails to Strip Bus 14 Loads and Manually Close Breaker for DG A/SW A per Attachment 4 to ER-FIRE	8.08E-05	1.61E-03	1.53E-07	2.02%
RCHFPX4BAF	OP Fails to Align BAF given Both PORVs Open	3.83E-03	1.29E-02	1.50E-07	1.98%
RCHFPX1BAF	OP Fails to Align BAF given a Single PORV and No Charging Pumps	1.30E-02	1.00E+00	9.83E-08	1.29%
AFHFDTDAFW	OP Fails to Manually Open Steam Valves to TDAFW Pump (No Fire)	1.00E-01	3.72E-01	9.78E-08	1.29%
FSHFDOFFPR	OP Fails to LOCALLY Re-Close Supply Bkrs for 52/16 and/or 52/14 after Hot Short Event Transfers the Bkrs Open	1.03E-01	1.72E-01	8.83E-08	1.16%
DGHFDRELAY	OP Fails to Start DG After Fire in Relay Room Fails all Auto Start Logic	1.00E-01	1.87E-01	8.62E-08	1.13%
FSHFDDGBXX	STA Fails to Start DG B and SW Pump per ER-FIRE.4	6.44E-03	1.30E-02	7.83E-08	1.03%
RHHFDRECXX	OP Fails to Recover RHR System before onset of boiling (> 24 Hours)	1.00E-03	1.27E-03	7.71E-08	1.01%
AXHFDCITYW	OP Fails to use City Fire Water for SAFW per ER-AFW.1	7.64E-04	3.93E-03	5.98E-08	0.79%
FSHFDMFW01	OP Manually Trips Reactor prior to Loss of MFW during ER-FIRE.1, .2, or .3 fire or flood	5.00E-01	5.00E-01	4.55E-08	0.60%
FSHFDAFWXX-3	OP Fails to Manually Align and Start TDAFW Pump given 52% NR SG Water Level Trip and NO RCPs Running - Fire Event	2.73E-02	4.76E-02	4.52E-08	0.60%
SRHFDRECRC	OP Fails to Shift SI System to Recir during a LOCA	5.40E-03	5.40E-03	3.77E-08	0.50%
AFHFDSUPPL-3	OP Fails to Supply Alternate Sources of Water to AFW	5.32E-04	5.32E-04	2.79E-08	0.37%
RCHFDPLOCA	OP Fails To Close PORV Block Valve (515/516) To Terminate LOCA	6.44E-03	8.63E-03	2.56E-08	0.34%
FSHFDRPORV	OP Fails to De-Energize PORV Control Circuit during a Fire/Flood	1.34E-03	3.97E-03	1.96E-08	0.26%
IAHFDCSA04	OP Fails to Place an Air Compressor in Service per E-0 STEP 33.C OR AP-IA.1	1.63E-03	2.41E-03	1.89E-08	0.25%

Table 2.13-14 (3 of 3)
Human Action Impacts Pre/Post-EPU

Basic Event	Event Description	Base Value	EPU Value	CDF Increase due to HA	% of Uprate Increase
AFHFDALTTD	OP Fails to Provide Cooling to TDAFW Lube Oil from Diesel Fire Pump	3.48E-03	3.48E-03	1.80E-08	0.24%
FSHFDDGAXX	STA Fails to Start DG A per Attachment 2 to ER-FIRE.1, .2 or .5	3.23E-04	1.78E-03	1.79E-08	0.24%
RRHFDSUCTN	OP Fails to Manually Open RHR Suction/Injection Valves	1.00E+00	1.00E+00	1.77E-08	0.23%
CVHFDBORAT	OP Fails to Implement Emergency Boration	5.00E-02	5.00E-02	1.53E-08	0.20%
RCHFDCDOSS	OP Fails to Cooldown to RHR After SI Fails (Injection or Recirc) - SSLOCA	1.40E-03	3.60E-03	1.14E-08	0.15%

The most significant human action timing changes involve recoveries from loss of shutdown cooling. During reduced inventory, early in shutdown, there is little time available for recovery. Given a fixed response time, the time available for diagnoses decreases by more than what would be expected from the change in decay heat (that is, 17% = 1811 ÷ 1550).

The remaining top contributors are AFW related. The likelihood of power recovery following a LOOP is related to the likelihood of recovering offsite power. Another factor considered in the time available to recover is the amount of time it takes for core uncovery following a loss of decay heat removal. All LOOP recovery times are conservatively adjusted by the using this change in decay heat removal timing. As seen in Table 2.13-14, the adjustment of LOOP recovery times are a significant contributor to the delta risk associated with EPU. Another important contributor is the operator's ability to diagnose a loss of decay heat removal. Although this has a low likelihood of failure, it has the high consequence of failing or degrading all decay heat removal related actions. The next most important contributor among the AFW related actions is recovering turbine driven AFW pump following a control room fire with a low steam generator water level trip. The current procedures direct operators to trip the reactor on a control room fire. If main feedwater is available at the time of the trip, the steam generator will have a normal water level at the time of the trip. Although unlikely, it is possible that the fire progression would disable main feedwater before the operators trip the reactor. If operators fail to trip the reactor, the reactor protective system trips the reactor on low steam generator water level. Even prior to the uprate, the operator failure likelihood was quite high for this unlikely situation. With the time reduction due to EPU, operators will not have sufficient time to align the turbine driven AFW pump prior to steam generator dry out.

As shown in Table 2.13-14, it would take numerous variations in human action failure likelihoods to significantly affect the conclusion that EPU is acceptable.

2.13.1.2.1.6.4 Component and System Reliability

Changes in equipment service conditions and operational changes are identified and discussed in other sections of this document. When needed, components are modified or replaced to obtain the required performance and operating margins at uprate conditions. Therefore, plant systems and equipment will continue to operate within design limits.

Existing equipment monitoring techniques (such as vibration analysis, thermography, oil analysis, radiography), preventive maintenance, and condition monitoring programs (such as Maintenance Rule and Erosion/Corrosion) will identify any accelerated component wear that might result from the power uprate. Through trending, these programs will also identify deviations or potential increases in component failure rates. While the power uprate may require more frequent maintenance or more frequent replacement of components, the overall reliability of components can be appropriately maintained.

As mentioned in 2.13.1.2.1.1, Internal Events sub-sections, initiating events for systems operating with reduced margin post-EPU are adjusted. A component sensitivity evaluation is presented in 2.13.1.2.4.2, Hardware Failure Likelihood and Unavailability.

2.13.1.2.2 Level 2 Analysis

The GPSA Level 2 analysis follows the NUREG/CR-6595 approach prescribed for PWRs with a large dry containment. The Level 2 analysis is performed after Level 1 analysis has been completed using a simplified containment event tree to calculate the Large Early Release Frequency (LERF).

The simplified Level 2 evaluation calculates the LERF using the CDF accident sequences. Sequence CDF results show the status of equipment considered important for continued containment integrity and the accident scenario involved. This equipment includes containment isolation valves and containment heat removal systems with important accident scenarios being steam generator tube ruptures and interfacing system LOCAs.

The analysis results are binned into LERF end states: Core Damage With Containment Failure at Vessel Breach (Low RCS Pressure), Core Damage With Containment Failure at Vessel Breach (High RCS Pressure), Core Damage With Induced SGTR (Low RCS Pressure), and Core Damage With Containment Failure of Bypass.

The Level 2 results are provided in Section 2.13.1.2.4.5, Modifications.

2.13.1.2.3 IPE/IPEEE Vulnerabilities

As discussed in the SAMA RAIs (2.13.2, Reference 3), all vulnerabilities identified in the IPE and IPEEE have been resolved with the exception of the potential for seismically induced flooding from the RMWT and Monitor Tank. The potential impact of a seismically induced flood on EPU is addressed in Section 2.13.1.2.1.3.1, Seismic Events.

2.13.1.2.4 EPU Risk Analysis

Table 2.13-15 shows the pre/post-EPU core damage frequency, the pre/post delta CDF, and the portion of the delta CDF attributed to initiating events and human actions. The current EPU delta risk is primarily driven by human actions.

Table 2.13-15
EPU Results Considering Human Actions, Initiating Events, and Others

	Pre-EPU	Post-EPU	Delta	HA Impact	IE Impact	Other
CDF ---->	6.36E-05	7.12E-05	7.60E-06	4.81E-06	2.03E-06	7.62E-07
Percent of contribution to delta ---->				63%	27%	10%

As part of the EPU, it is prudent to minimize losses of margin that can increases risk, yet, it is impossible to anticipate all possible losses of margin. For this reason, sensitivity evaluations are used to provide further confidence that the EPU does not pose undue risk to the public.

2.13.1.2.4.1 Initiating Events

The first sensitivity evaluation focuses on initiating events that may be impacted by EPU.

- *Increased Fluid Flow* - Although it is the goal of the erosion control program to manage the impact of increased flows and maintain piping integrity, it is possible that certain piping segments will have increased wear and have a higher likelihood of failure. For this reason, the sensitivity of feed/steam line breaks are examined.
- *Electrical Stability* - Although every prudent effort is made to ensure electrical stability post-EPU, the lines from the generator to the switchyard and beyond will be more heavily loaded. For this reason, the sensitivity of line/grid loses are examined.
- *Secondary System Reliability* - Modifications will be (or have been) made to ensure that the redundancy of major components remains the same pre/post-EPU (for example, condensate booster pump upgrade). There is still a loss of margin for secondary systems. For this reason, loss of MFW and Reactor Trip sensitivity are examined.

Table 2.13-16 shows the sensitivity results for the initiating events that may be increased by EPU. The chart shows the initiating event designator, a description of the initiator, the post-EPU frequency, the Fussell-Vesely importance, and EPU CDF Delta increase given the initiator frequency were to double. This CDF increase refers to the increase in Delta CDF of EPU if the post-EPU initiating event frequency were to double. A later table (Table 2.13-17) shows the increase in delta risk associated with the EPU as the initiating events change. It should be noted that many of the initiating event frequencies were increased for the purpose of the EPU analysis as described in Section 2.13.1.2.1.1, Internal Events.

Table 2.13-16
Key Initiating Event Sensitivity Study

Initiating Event (IE)	Description	Base IE Freq	IE Freq Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
TIRXTRIP	Reactor Trip	1.25E+00	20%	4.75E-02	1.18E-06
TIGRLOSP	Loss of Offsite Power - Grid	6.44E-03	10%	4.16E-02	4.94E-07
TIFWLOSS	Loss of Main Feedwater	5.44E-02	40%	1.01E-02	2.33E-07
TISWLOSP	Loss of Offsite Power - Switchyard	7.87E-03	10%	2.75E-03	2.54E-08
TISLB0TB	Steamline Break in Turbine Building	4.55E-03	40%	1.44E-03	6.74E-08
TIFLB0TB	Feedline Break in Turbine Building	2.20E-03	40%	6.84E-04	3.22E-08
TIOSLBSD	Steamline Break Through Steam Dump System	4.10E-03	10%	1.23E-04	2.87E-09

A basic reactor trip is the most sensitive initiating event that could be impacted by EPU. There are numerous possible causes of a reactor trip (TIRXTRIP). Of these, three are potentially impacted by EPU: load rejection capability, oil-cooled line loading, and isophase bus cooling. Given this, it seems quite unlikely that the reactor trip frequency would double.

EPU affects the likelihood of a total loss of the grid (TIGRLOSP). Although this event could be impacted by EPU, it is considered the least likely to be impacted by EPU. Post-EPU, it is possible that a transmission line could exceed its design limits following a spurious breaker opening, however, it is seen as unlikely that the loss of the entire grid would result.

Less margin between normal operations and trip set points will exist post-EPU for the secondary systems. This will potentially increase the likelihood of a loss of main feedwater (MFW). Considering the efforts being made to ensure event free MFW operation, the current increase is likely to be both conservative and bounding.

The remaining initiators would have a small impact even in the radical case where the initiating event frequency doubles.

Table 2.13-17 shows the top initiators that contribute to the increase in risk associated with the EPU. Although the initiating event frequency may not change, the time available to perform the required human actions given the initiating event occurs may change. For example, the frequency of a control room fire does not change, but the operator response time available is reduced due to higher decay heat. This can cause an initiating event not directly impacted by EPU to be an important contributor to the delta risk associated with EPU. The chart shows the initiating event designator, a description of the initiator, the pre-EPU frequency, the change in frequency post-EPU, the post-EPU Fussell-Vesely importance, and CDF increase to the EPU delta risk given the initiator frequency were to double. Note that deltas less than 10^{-8} are not shown.

Table 2.13-17 (1 of 3)
Top Initiator Contributions

Initiating Event (IE)	Description	Base IE Freq	IE Freq Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
TX000RHR	Loss of RHR	5.90E-04	Same	1.60E-01	2.30E-06
TIRXTRIP	Reactor Trip	1.25E+00	20%	4.75E-02	1.18E-06
FIOCR3-1	Fire in Zone CR-3 (Scenario 1 and 2) - Suppression Fails	1.68E-03	Same	1.58E-01	6.92E-07
TIGRLOSP	Loss of Offsite Power - Grid	6.44E-03	10%	4.16E-02	4.94E-07
FIOTB1-5	Fire in Zone TB-1 (Scenario 5) - Suppression OK	3.93E-02	Same	1.58E-02	3.68E-07
FLOOORR1	Flood in Zone RR (Scenario 1)	1.12E-05	Same	4.26E-02	2.35E-07
TIFWLOSS	Loss of Main Feedwater	5.44E-02	40%	1.01E-02	2.33E-07
FIOTB2-3	Fire in Zone TB-2 (Scenario 3) - Suppression OK	3.56E-02	Same	1.60E-02	1.53E-07
FIOCR3-3	Fire in Zone CR-3 (Scenario 3) - Suppression Fails	3.37E-03	Same	3.16E-02	1.45E-07
FLOOOTB3	Steam flooding event in Turbine Building	6.73E-03	Same	5.05E-02	1.36E-07
FIIBN1-3	Fire in Zone IBN-1 (Scenario 3) - Suppression OK	6.06E-02	Same	1.02E-02	1.26E-07
FLOOOIB1	Large internal flood in the Intermediate Building	4.09E-03	Same	5.05E-03	1.13E-07
FLOOOTB7	Large flood originating in Turbine Building mezzanine level (zone TB-2)	4.81E-03	Same	4.52E-02	1.02E-07
FIASH2-1	Fire in Zone SH-2 (Scenario 1 and 2) - Suppression OK	1.69E-02	Same	6.35E-03	9.66E-08
FIOOORR7	Fire in Zone RR (Scenario 7) - Suppression OK	3.74E-03	Same	2.18E-03	8.72E-08
FIBR1A-3	Fire in Zone BR1A (Scenario 3 and 4) - Suppression Fails	6.75E-03	Same	1.99E-02	7.49E-08
TISLBOTB	Steamline Break in Turbine Building	4.55E-03	40%	1.44E-03	6.74E-08
FIOOORR6	Fire in Zone RR (Scenario 6) - Suppression OK	3.16E-02	Same	5.07E-03	6.14E-08
TIOOOCCW	Loss of Component Cooling Water	9.47E-04	Same	3.67E-03	5.84E-08
FIOOTYW2	Fire in Zone TY-W (Scenario 2) - Suppression OK	1.69E-03	Same	1.39E-02	5.48E-08
FIBR1B-3	Fire in Zone BR1B (Scenario 3 and 4) - Suppression Fails	6.76E-03	Same	4.71E-02	5.27E-08
FIOOTYE3	Fire in Zone TY-E (Scenario 3) - Suppression OK	1.59E-03	Same	1.30E-02	4.79E-08
FIATB2-1	Fire in Zone TB-2 (Scenario 1 and 2) - Suppression OK	1.15E-02	Same	5.02E-03	4.51E-08

Table 2.13-17 (2 of 3)
Top Initiator Contributions

Initiating Event (IE)	Description	Base IE Freq	IE Freq Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
FIDG1B10	Fire in Zone EDG1B-0 - Suppression OK	9.71E-05	Same	1.07E-03	3.97E-08
FIOSH2-3	Fire in Zone SH-2 (Scenario 3) - Suppression OK	7.21E-03	Same	2.48E-03	3.92E-08
TIFLBOTB	Feedline Break in Turbine Building	2.20E-03	40%	6.84E-04	3.22E-08
FIADG1B1	Fire in Zone EDG1B-1 (Scenario 1 and 2) - Suppression OK	1.37E-02	Same	2.31E-03	2.77E-08
FIAOABB1	Fire in Zone ABB (Scenario 1 and 2) - Suppression OK	1.37E-02	Same	2.15E-03	2.59E-08
TISWLOSP	Loss of Offsite Power - Switchyard	7.87E-03	10%	2.75E-03	2.54E-08
LISSLOCA	Small-Small LOCA (0-1)	8.51E-03	Same	3.56E-02	2.25E-08
TI48LOSP1	Loss of Offsite Power to 480V Buses from both offsite circuits (Initiator in all configurations)	7.53E-04	Same	3.44E-03	2.17E-08
FIOTB1-6	Fire in Zone TB-2 (Scenario 6) - Suppression OK	3.26E-02	Same	4.62E-03	1.98E-08
FIAOABM3	Fire in Zone ABM (Scenario 3 or 4) - Suppression OK	9.62E-03	Same	1.48E-03	1.82E-08
TI0000SW	Total Loss of Service Water	6.30E-05	Same	5.86E-04	1.70E-08
FI00ABO1	Fire in Zone ABO (Scenario 1 and 2) - Suppression OK	3.41E-03	Same	7.84E-04	1.49E-08
TI000DCB	Loss of Main DC Distribution Panel B (DCPDPCB03B)	1.03E-03	Same	1.12E-03	1.42E-08
TI48LOSP4	Loss of Circuit 751 followed by loss of circuit 767 within 24 hours (Initiator in either 50/50 mode)	5.15E-04	Same	2.25E-03	1.39E-08
TI48LOSP5	Loss of Circuit 767 followed by loss of circuit 751 within 24 hours (Initiator in either 50/50 mode)	5.15E-04	Same	2.25E-03	1.39E-08
TIIALOSS	Loss of Instrument Air (Freq calculated via fault tree model)	3.28E-02	Same	4.65E-03	1.37E-08
FIABR1A3	Fire in Zone BR1A (Scenario 3 and 4) - Suppression OK	6.75E-03	Same	1.02E-03	1.31E-08
FLODG1B1	Internal flood originating in Diesel Generator B room (zone EDG1B)	6.40E-04	Same	5.18E-04	1.30E-08
FL000RC1	Internal flood in Reactor Containment	4.34E-02	Same	5.68E-04	1.28E-08
FIABR1B3	Fire in Zone BR1B (Scenario 3 and 4) - Suppression OK	6.76E-03	Same	1.07E-03	1.26E-08

Table 2.13-17 (3 of 3)
Top Initiator Contributions

Initiating Event (IE)	Description	Base IE Freq	IE Freq Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
FI000AVT	Fire in Zone AVT - Screened	2.79E-02	Same	3.63E-03	1.23E-08
FI000TB3	Fire in Zone TB-3 - Suppression OK	2.86E-02	Same	8.98E-04	1.22E-08
FI00ABM5	Fire in Zone ABM (Scenario 5) - Suppression OK	6.44E-03	Same	9.55E-04	1.19E-08
FIA0ABB3	Fire in Zone ABB (Scenario 3 and 4) - Suppression OK	5.78E-03	Same	8.49E-04	1.08E-08

The likelihood of the RHR system failing is not directly impacted by EPU, but the time available to recover from those failures is significantly impacted by EPU (see RHHFDREC00 in 2.13.1.2.1.6.3, Operator Actions, Table 2.13-14).

Not only is the likelihood of a reactor trip directly impacted by EPU, but the ability to mitigate a reactor trip is degraded by the reduction in time available for mitigation actions (see 2.13.1.2.1.6.3, Operator Actions, Tables 2.13-13 and 2.13.-14).

Although fire in the control room complex is not directly impacted by EPU, the human action to align AFW in this situation is significantly impacted (see FSHFDAFWXX-1 in 2.13.1.2.1.6.3, Operator Actions, Table 2.13-14).

2.13.1.2.4.2 Hardware Failure Likelihood and Unavailability

For additional assurance of the EPU acceptability, the impacts of changes in hardware failure likelihood and unavailability are also examined. Table 2.13-18 shows the top EPU related hardware items. If the failure likelihood (or unavailability fraction) were to double, then the delta increase for EPU is shown. For example, with no modifications the increase in risk associated with EPU is 7.60×10^{-6} (see 2.13.1.2.4.5, Modifications, Table 2.13-23). If the failure likelihood of *Inverter 'B' and Constant Voltage Transformer 'B' to Instrument Bus 1C* doubles (that is, IBMMBUS1CX is 2.64×10^{-4} versus 1.32×10^{-4}), then the EPU CDF risk increase would be

$$8.69 \times 10^{-6} = 7.60 \times 10^{-6} + 1.09\text{E} \times 10^{-6}$$

Table 2.13-18 (1 of 5)
Hardware Failure Impact

Basic Event	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
IBMMBUS1CX	Failures Common to INVTB and CVTB	1.32E-04	Same	7.23E-02	1.09E-06
ACAZSEALX3	21gpm/pp RCP Seal Failure Likelihood during SBO (>42 gpm)	7.90E-01	Same	2.17E-01	8.86E-07
ACAZSEALX2	182gpm/pp RCP Seal Failure Likelihood during SBO (364 gpm)	2.07E-01	Same	7.18E-02	4.63E-07
DGDGF0001B	Diesel Generator KDG01B Fails to Run	4.22E-02	Same	7.16E-02	4.36E-07
DGDGF0001A	Diesel Generator KDG01A Fails to Run	4.22E-02	Same	9.87E-02	4.13E-07
RHCCFPUMPR_12	CCF of Pumps PAC01A and PAC01B (RHR Pumps) to Run	2.06E-05	Same	2.83E-02	3.87E-07
DGTM00001B	Diesel Generator KDG01B Unavailable due to Testing or Maintenance	1.74E-02	Same	4.26E-02	3.30E-07
DGTM00001A	Diesel Generator KDG01A Unavailable due to Testing or Maintenance	1.55E-02	Same	4.66E-02	2.78E-07
DGDGA0001B	Diesel Generator KDG01B Fails to Start	1.01E-02	Same	3.65E-02	2.65E-07
DGDGA0001A	Diesel Generator KDG01A Fails to Start	1.01E-02	Same	4.35E-02	2.61E-07
DGCCF00RUN_12	CCF of KDG01A and KDG01B to Run	1.76E-03	Same	3.74E-02	2.08E-07
CCCCFPUMPR_12	CCF of Pumps PAC02A and PAC02B (CCW Motor-Driven Pumps) to Run	6.74E-06	Same	1.12E-02	1.55E-07
DGCCFSTART_12	CCF of KDG01A and KDG01B to Start	3.25E-04	Same	2.27E-02	1.47E-07
AFTM0TDAFW	TDAFW Pump Train OOS for Maintenance	1.34E-02	Same	1.53E-02	1.42E-07
TLCCFMATWS	Mechanical Scram Failure Probability (Rods Cannot Be Inserted)	1.20E-06	Same	9.92E-03	9.13E-08
AFMM0TDAFW	Failure of TDAFW Pump Train Components (except CKV 3998 Fails to Open)	9.68E-03	Same	1.08E-02	8.84E-08
SICCFPSI1S_123	CCF of Pumps PSI01A, PSI01B and PSI01C (SI Pumps) to Start	5.37E-05	Same	1.15E-02	8.74E-08
RRMVK00700	MOV 700 Transfers Closed	7.46E-06	Same	5.75E-03	8.63E-08
RRMVK00701	MOV 701 Transfers Closed	7.46E-06	Same	5.75E-03	8.63E-08
ACLOPSHTDN	Loss of Offsite Power During 24 Hour Period When Shutdown	8.49E-05	Same	1.29E-02	8.53E-08
DGMMBRKR14	Failures of DG A Supply Breaker to Bus 14 to Close	3.43E-03	Same	9.50E-03	7.47E-08
FSTM0PFP01	Diesel-Driven Fire Service Water Pump (PFP01) Unavailable due to Maintenance	1.18E-02	Same	5.16E-03	7.29E-08

Table 2.13-18 (2 of 5)
Hardware Failure Impact

Basic Event	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
RRMVK00720	MOV 720 Transfers Closed	7.46E-06	Same	4.69E-03	7.06E-08
RRMVK00721	MOV 721 Transfers Closed	7.46E-06	Same	4.69E-03	7.06E-08
AFTMMAFSGB	MDAFW Train B to SG B OOS due to T/M	9.74E-03	Same	1.23E-03	6.53E-08
SICCFPSI1R_123	CCF of Pumps PSI01A, PSI01B and PSI01C (SI pumps) to Run	3.52E-05	Same	7.40E-03	5.69E-08
AXMMSAFWPC	Failure of SAFW Pump 1C Train	4.88E-03	Same	2.08E-03	5.46E-08
RCAZD00515	MOV 515 Is Closed Due to PORV Leakage	1.77E-02	Same	7.83E-04	5.37E-08
SWCCFPUMPR_ALL	CCF of PSW01A, PSW01B PSW01C and PSW01D (SW Pumps) to Run for 24 Hours	2.21E-06	Same	4.07E-03	5.32E-08
CCXVK00728	Manual Valve 728 Transfers Closed	2.26E-06	Same	3.74E-03	5.20E-08
RCAZD00515N	MOV 515 Is Not Closed Due to PORV Leakage	9.82E-01	Same	2.01E-02	5.10E-08
AFTMMAFSGA	MDAFW Train A to SG A OOS due to T/M	7.38E-03	Same	9.13E-04	4.90E-08
DGMMBRKR16	Failures of DG B Supply Breaker to Bus 16 to Close	3.43E-03	Same	6.47E-03	4.79E-08
AXTMSAFSGB	SAFW Train D to SG B OOS due to T/M	2.03E-02	Same	2.41E-03	4.76E-08
MSRYT03508	SG Relief Valve 3508 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03509	SG Relief Valve 3509 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03510	SG Relief Valve 3510 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03511	SG Relief Valve 3511 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03512	SG Relief Valve 3512 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03513	SG Relief Valve 3513 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03514	SG Relief Valve 3514 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
MSRYT03515	SG Relief Valve 3515 Fails to Close After Steam Release	4.04E-03	Same	6.63E-03	4.72E-08
AXTMSAFSGA	SAFW Train C to SG A OOS due to T/M	1.58E-02	Same	2.38E-03	4.61E-08
AXMMSAFWPD	Failure of SAFW Pump 1D Train	4.88E-03	Same	2.14E-03	4.49E-08
TMAZOVSPDB	Probability of an intermediate overspeed event (132%) occurring given an initiating event with the potential to cause	4.00E-04	Same	2.21E-03	4.48E-08

Table 2.13-18 (3 of 5)
Hardware Failure Impact

Basic Event	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
TMAZTBMISSB	Probability of generating a turbine missile given an intermediate overspeed event (132%) event	9.00E-03	7%	2.21E-03	4.48E-08
RCRZT00430	PORV PCV-430 Fails To Reseat After Steam Relief	5.00E-03	Same	1.39E-02	4.20E-08
IAXVK07350	Manual Valve 7350 Transfers Closed	4.66E-06	Same	2.54E-03	3.84E-08
IAXVK07370	Manual Valve 7370 Transfers Closed	4.66E-06	Same	2.54E-03	3.84E-08
MFAZCCMFW	CCF Fails All MFW Flow to Both SGs during the 24 Hour Period Following a Trip	1.00E-02	Same	1.66E-03	3.61E-08
CCMVN0738B	MOV 738B Fails to Open	3.90E-03	Same	4.53E-03	3.57E-08
RCRZT0431C	PORV PCV-431C Fails To Reseat After Steam Relief	5.00E-03	Same	1.35E-02	3.26E-08
CCPPJ_COMM	Pipe Rupture in the Common CCW Piping	1.36E-05	Same	4.11E-03	3.25E-08
AXCCFSAFWS_12	CCF of PSF01A and PSF01B (SAFW Motor-Driven Pumps) to Start	2.34E-04	Same	1.35E-03	3.00E-08
SIMMINJECB	Valve Failures in SI Pump B Injection Line to Loop A Cold Leg	3.19E-03	Same	3.93E-03	2.92E-08
SIMMINJECA	Valve Failures in SI Pump A Injection Line to Loop B Cold Leg	2.24E-03	Same	3.31E-03	2.52E-08
RCAZD00516	MOV 516 is Closed Due To PORV Leakage	1.77E-02	Same	3.72E-04	2.45E-08
AXMPAPSF1A	SAFW Motor-Driven Pump 1C Fails to Start	2.65E-03	Same	1.02E-03	2.42E-08
IBMMBUS01C	120 VAC Instrument Bus C (IBPDPCBCB) Bus Faults	2.69E-06	Same	1.46E-03	2.19E-08
DGTM0OUT1A	DG KDG01A Unavailable due to T/M (Outage)	1.62E-01	Same	3.09E-03	2.16E-08
AXMVD9701A	MOV 9701A Fails to Throttle Flow	2.49E-03	Same	9.37E-04	2.14E-08
AXMPAPSF1B	SAFW Motor-Driven Pump 1D Fails to Start	2.65E-03	Same	1.05E-03	2.10E-08
DGTM0OUT1B	DG KDG01B Unavailable due to T/M (Outage)	1.62E-01	Same	3.09E-03	2.09E-08
MSTM003411N	ARV 3411 NOT in T/M	9.75E-01	Same	2.55E-03	2.00E-08
AXMVD9701B	MOV 9701B Fails to Throttle Flow	2.49E-03	Same	9.82E-04	1.98E-08
MSTM003410N	ARV 3410 NOT in T/M	9.93E-01	Same	2.60E-03	1.98E-08
MSRVC03410	Air-Operated Valve 3410 (ARV) Fails to Close	1.69E-03	Same	2.57E-03	1.95E-08

Table 2.13-18 (4 of 5)
Hardware Failure Impact

Basic Event	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
MSRVC03411	Air-Operated Valve 3411 (ARV) Fails to Close	1.69E-03	Same	2.52E-03	1.92E-08
DCMMAIN1B	Failure of Circuit E76 (To Main DC Distribution Panel B)	2.62E-05	Same	8.15E-04	1.88E-08
FSDGFPFP01	Diesel Driven Fire Service Water Pump (PFP01) Fails to Run	5.64E-03	Same	1.37E-03	1.80E-08
SITMTRAINA	SI Train A Discharge Valves Unavailable due to T/M	3.43E-03	Same	2.50E-03	1.77E-08
ACB2FBUS14	Local Fault On 480 VAC Bus 14	1.46E-05	Same	2.70E-03	1.73E-08
RHTM00000A	RHR Train A OOS for Maintenance	7.00E-03	Same	4.95E-03	1.73E-08
AFMMSGAINJ	Failure of AFW Injection Line to SG A	2.65E-03	Same	3.02E-04	1.70E-08
AFMMSGBINJ	Failure of AFW Injection Line to SG B	2.65E-03	Same	3.02E-04	1.70E-08
RHTM00000B	RHR Train B OOS for Maintenance	8.65E-03	Same	5.37E-03	1.66E-08
RWMM896A/B	MOV 896A or 896B Transfers Closed (Fails CS and SI from RWST)	1.04E-05	Same	2.08E-03	1.64E-08
ACB2FBUS16	Local Faults On 480 VAC Bus 16	1.46E-05	Same	2.42E-03	1.58E-08
SITMTRAINB	SI Train B Discharge Valves Unavailable due to T/M	3.65E-03	Same	2.01E-03	1.40E-08
TI48LOSP4	Loss of Circuit 751 Followed by Loss of Circuit 767 within 24 Hours (Initiator in Either 50/50 Mode)	5.15E-04	Same	2.25E-03	1.39E-08
TI48LOSP5	Loss of Circuit 767 Followed by Loss of Circuit 751 within 24 Hours (Initiator in Either 50/50 Mode)	5.15E-04	Same	2.25E-03	1.39E-08
AFMMMDFP1A	Failure of MDAFW Pump Train A	2.15E-03	Same	2.41E-04	1.38E-08
AFMMMDFP1B	Failure of MDAFW Pump Train B	2.15E-03	Same	2.41E-04	1.38E-08
RHMVR0850A	MOV 850A Transfers Open [Injection]	8.52E-06	Same	2.80E-03	1.36E-08
RHMVR0850B	MOV 850B Transfers Open [Injection]	8.52E-06	Same	2.80E-03	1.36E-08
RHXVR1816A	Manual Valve 1816A Transfers Open	8.56E-04	Same	8.98E-04	1.34E-08
RHXVR1816B	Manual Valve 1816B Transfers Open	8.56E-04	Same	8.98E-04	1.34E-08
IAPPJHEADR	IA Piping Header Rupture	3.05E-02	Same	4.40E-03	1.31E-08
RCCCFPORVC-12	CCF of PORVs 430 and 431C to Reseat After Steam Release	1.86E-04	Same	1.29E-03	1.30E-08
FSDGAPFP01	Diesel-Driven Fire Service Water Pump (PFP01) Fails to Start (Standby)	2.50E-03	Same	9.57E-04	1.16E-08
ACCBD2BTAA	4160 VAC Bus 11A / Bus 12A Tie Breaker 52/BTA-A (BUS11A/11) Fails to Close	3.39E-03	Same	4.95E-04	1.15E-08

Table 2.13-18 (5 of 5)
Hardware Failure Impact

Basic Event	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
ACCBD5211A	4160 VAC Circuit Breaker 52/11A (BUS11A/10) Fails To Open on Demand	3.39E-03	Same	4.95E-04	1.15E-08
DGMMBRKR18	Failures of DG A Supply Breaker to Bus 18 to Close	3.43E-03	Same	4.60E-03	1.14E-08
SWXVK04665	Manual Valve 4665 From SW Supply Header A Transfers Closed	4.44E-04	Same	3.56E-04	1.12E-08
ACCBD0PHBG	480 VAC Circuit Breaker 52/PHBG to PRZR Backup Heaters Fails to Close	3.39E-03	Same	5.66E-04	1.10E-08
DGMMBRKR17	Failures of DG B Supply Breaker to Bus 17 to Close	3.43E-03	Same	4.29E-03	1.03E-08
AXMPFPSF1A	SAFW Motor-Driven Pump 1C Fails to Run	1.44E-03	Same	4.86E-04	1.02E-08
ACCBD0PHCG	480 VAC Circuit Breaker 52/PHCG to PRZR Heater Control Cabinet Fails to Close	3.39E-03	Same	4.84E-04	1.02E-08

The inverter and constant voltage transformer failure (IBMMBUS1CX) is most important during shutdown. This provides power to HCVs 624 and 625. If these valves fail open during reduced inventory, then the operating residual heat removal pumps may not have sufficient net positive suction head. If a modification/procedure change to limit the maximum fail open position of these valves is implemented, then the impact of changes in the failure likelihood of IBMMBUS1CX is much smaller (see 2.13.1.2.4.5, Modifications).

The larger RCP seal LOCAs (for example, ACAZSEALX2) are most important during fire scenarios. A modification/procedure change to allow the use of a safety injection pump during a fire scenario will reduce the sensitivity of RCP seal LOCA likelihood to EPU.

The diesel generators and the likelihood of a post-trip LOOP are sensitive parameters that could impact EPU. Fortunately, the diesel generator performance would need to be severely degraded to affect the conclusion that the EPU is acceptable.

The remaining items are considered to have little impact on EPU acceptability.

2.13.1.2.4.3 Other Impacted Parameters

For further assurance of the EPU acceptability, the impacts of changes to other parameters shown in Table 2.13-19 are also examined. If the probability of each event were to double, then the delta CDF increase for EPU is shown.

Table 2.13-19
Other Impacted Parameters

Event Name	Description	Base Probability	Probability Increase for EPU	EPU Fussell Vesely	EPU CDF Delta
ACLOPRTALL	Loss of All Off-Site Power Following Reactor Trip	2.59E-03	10%	4.71E-02	7.84E-07
RCAZTRIPLL	Fraction of Reactor Trips caused by a large LOL	4.35E-02	13%	2.38E-02	4.73E-07
TMAZTBMISSA	Probability of a generating a turbine missile, given a design overspeed event (120%) event	5.00E-04	8%	4.60E-03	9.35E-08
TL00082DAY	Time period RCS will always overpressurize	4.05E-01	25%	4.09E-03	7.50E-08
RCAZTRIPPM	Fraction of Reactor Trips caused by a medium LOL (i.e. PORV/MSSV Challenge with no Turbine Missile)	5.65E-03	53%	1.39E-03	4.57E-08
TMAZTBMISSB	Probability of a generating a turbine missile, given an intermediate overspeed event (132%) event	9.00E-03	7%	2.21E-03	4.48E-08
ACLOPRT767	Loss of Offsite Circuit 767 Following Reactor Trip	1.28E-03	10%	9.27E-05	4.39E-09
ACLOPRT751	Loss of Offsite Circuit 751 Following Reactor Trip	1.28E-03	10%	1.01E-04	3.94E-09
SW00FAILCRFC	CRFCs fail due to inadequate SW support	0.00E+00	100%	1.00E+00	2.92E-10
RCAZTRIPPV	Fraction of Reactor Trips due to Pressurizer Level Control Problem resulting in a PORV Challenge	1.00E-03	50%	1.96E-06	1.40E-10

A sensitive parameter is the likelihood of a LOOP following a reactor trip. Although it is expected that a trip at EPU conditions will increase the grid perturbations, it is not expected that the LOOP likelihood will significantly increase beyond the estimated impact.

The percentage of reactor trip that will cause a PORV challenge is a sensitive parameter. Fortunately, the increase used for this parameter is likely conservative (see 2.13.1.2.1.1.1, Loss of Coolant Accident, Table 2.13-1). The main delta arises for the mid-range power levels where pre-EPU the PORVs/MSSVs would not have been challenged, but post-EPU, the PORVs/MSSVs would be challenged. Considering the small amount of time spent at these power levels and the increased amount of recovery time available due to the lower power levels, this parameter value is likely to be much lower.

As the MSSV, ARV, PORV, and PSRV capacities remain the same, the relative steam removal per megawatt thermal power is reduced. This alone would cause the ATWS equilibrium power levels to require more equipment. As the MTC is more negative over the cycle, this impact is significantly reduced. Because ATWS could be impacted, an ATWS related parameter is examined to provide assurance that the current modeling addresses this issue. As shown in the table, the ATWS parameter (TL00082DAY) is not sensitive to EPU.

2.13.1.2.4.4 Level 1 and Level 2 Results

Level 1 and Level 2 results for all models are presented in Table 2.13-20. Note that the quantification results are based on *existing* Ginna Station equipment and procedures:

Table 2.13-20
Pre/Post-Uprate Internal Events and Internal Flood CDF

Model	Pre-Uprate		Post-Uprate		Change	
	CDF	LERF	CDF	LERF	CDF	LERF
Internal	1.30E-05	1.27E-06	1.51E-05	1.51E-06	16%	19%
Internal Flood	1.17E-05	5.10E-07	1.23E-05	5.45E-07	5%	7%
Fire	2.83E-05	2.76E-06	3.07E-05	2.89E-06	8%	5%
Shutdown	1.07E-05	3.46E-07	1.30E-05	4.04E-07	21%	17%
Total	6.36E-05	4.88E-06	7.12E-05	5.35E-06	12%	10%

The above results reflect quantification of the GPSA at a 1×10^{-10} truncation limit. This value provides sufficient resolution for the risk calculations used for EPU. This is based on truncation sensitivity analysis results that indicate a total estimated difference of approximately four percent for CDF and seven percent for LERF between the 1×10^{-10} truncation and the extrapolated no truncation quantification results. This indicates that 1×10^{-10} is an appropriate truncation limit for the EPU risk analysis.

2.13.1.2.4.5 Modifications

The EPU modeling provides several insights with regard to plant modifications and operational improvements that could reduce risk. A review of these resulted in five potential changes that are both risk and cost beneficial. These changes are listed below:

- <u>*Optimize use of the safety injection pumps (SI) During Fires*</u>
  Certain control room complex fires can result in a single charging pump being the only RCS injection capability for inventory control. If that pump is out of service or fails, there is no proceduralized injection alternative. At least one safety injection pump will be able to be powered from the unaffected power supply and could be locally started to provide injection capability. New procedure steps that direct the operator to secure power to the safety injection pump suction and recirculation MOVs (to preclude a "hot short" failing the valves closed), manually open the MOVs if necessary, and locally start a safety injection pump, would provide RCS injection if no charging pump was available.

- <u>*Mechanically limit RHR HCV-624 and HCV-625 (SDAOV)*</u>
  These air operated hand control valves (HCVs) control flow out of the residual heat removal

heat exchangers. On a loss of electrical or compressed air support these HCVs fail open. If the HCVs fail open during shutdown mid-loop operations, the residual heat removal pumps could lose adequate net positive suction head, cavitate, and (ultimately) fail. A procedure change to mechanically limit the open position of the HCVs and avoid a loss of net positive suction head could significantly reduce shutdown risk. -

- *Provide backup air to the charging pumps (BK-IA-CHG)*
  Ginna Station uses a variable speed charging pump design. Speed control is accomplished via an air operated actuator. The present design results in the charging pump going to low speed on a loss of compressed air support. At low speed, charging flow is not sufficient to support bleed and feed mitigation. A modification to install local air bottles would supply the speed controllers for more than one hour: This would allow operators to bleed and feed when needed and provide time to restore (or align alternative) compressed air.

- *Relocate charging pump control power disconnect*
  This modification would save operator time in taking local control of the charging pumps during recovery from a fire. Injecting RCS sooner would prevent the pressurizer depletion in some accident scenarios.

- *Install local controls for the turbine AFW pump discharge MOV*
  The current fire response procedure directs the operator to manually control the AFW discharge AOVs before starting the pump: This is a time consuming task that involves pinning the valve shaft (which is not readily accessible). The modification would allow the operator to control the AFW pump discharge MOV from the remote shutdown panel.

A sensitivity study is performed for these modifications. The results are shown in Table 2.13-21. For each case (including a base "no modification" case) the CDF and LERF are shown with the modification(s) in effect for the case. Cases are presented both pre/post-uprate.

(Note that case SI-AOV-IC represents the combined SI, SDAOV, and BK-IA-CHG cases. Sensitivities for the charging pump control power relocation and local controls for the turbine driven AFW pump discharge MOV are not performed as these modifications have very small risk improvements.)

Table 2.13-21
Modification Sensitivity Study

Case	Pre or Post Uprate	CDF	LERF	Optimize SI Pump in Fire	Limit RHR AOVs	Back-Up Air to Charging	Charging Control Power Relocation	TD AFW Pump Discharge MOV
Base	Pre	6.36E-05	4.88E-06	No	No	No	No	No
SI	Pre	5.63E-05	4.27E-06	Yes	No	No	No	No
SDAOV	Pre	5.94E-05	4.86E-06	No	Yes	No	No	No
BK-IA-CHG	Pre	6.34E-05	4.74E-06	No	No	Yes	No	No
SI-AOV-IC	Pre	5.20E-05	4.11E-06	Yes	Yes	Yes	No	No
Base	Post	7.12E-05	5.35E-06	No	No	No	No	No
SI	Post	6.40E-05	4.73E-06	Yes	No	No	No	No
SDAOV	Post	6.59E-05	5.32E-06	No	Yes	No	No	No
BK-IA-CHG	Post	7.10E-05	5.20E-06	No	No	Yes	No	No
SI-AOV-IC	Post	5.85E-05	4.56E-06	Yes	Yes	Yes	No	No

Table 2.13-22 shows risk improvement associated with the above modifications post-EPU.

Table 2.13-22
Modification Risk Improvements

Modifications	Improvements for EPU	
	CDF	LERF
SI	7.20E-06	6.20E-07
SDAOV	5.30E-06	3.00E-08
BK-IA-CHG	2.00E-07	1.50E-07
SI-AOV-IC	1.27E-05	7.90E-07

Table 2.13-23 shows the delta pre/post-EPU for the modifications above.

Table 2.13-23
Pre and Post Uprate Risk Deltas

Modifications	Uprate Deltas	
	CDF	LERF
Base	7.60E-06	4.70E-07
SI *	7.70E-06	4.60E-07
SDAOV	6.50E-06	4.60E-07
BK-IA-CHG	7.60E-06	4.60E-07
SI-AOV-IC	6.50E-06	4.50E-07

* The CAFTA reports used to generate the CDF and LERF risks displayed in Table 2.13-21 have only three digits. As a result, the accuracy of the numbers is $\pm 1\times10^{-7}$ for CDF and $\pm 1\times10^{-8}$

for LERF. Due to the three digit resolution, the ΔCDF and ΔLERF displayed in Table 2.13-23 have an accuracy of $\pm2\times10^{-7}$ and $\pm2\times10^{-8}$ respectively. Although the delta risk of SI appears larger, it is expected that with more accuracy the delta risk would actually be slightly smaller.

2.13.1.2.4.6 Risk Analysis Conclusion

Although there is a risk increase associated with EPU (see 2.13.1.2.4.5, Modifications, Table 2.13-21), this risk increase will likely be completely off-set through the implementation of three modifications/procedure changes:

- Credit the use of 1A Safety Injection Pump in the ER-Fire procedures,
- Limit the amount the residual heat removal outlet valves can open on the loss of support, and
- Install an air back-up system to ensure charging can be used at maximum speed for at least one hour on the loss of the normal instrument air supply.

Until implemented, the exact benefit of these modification/procedure changes is uncertain. The risk evaluations are done considering that the issue is completely resolved. Fortunately, even if the modifications/procedure changes achieve only ninety percent of the maximum benefit, the risk increase associated with EPU can be completely offset. The nature of the modification/procedure changes is such that achieving near the full benefit is a reasonable expectation. These modification/procedure changes give further confidence in the acceptability of EPU.

While the EPU is not a risk-informed application, the risk increase due to EPU meets more rigorous standard of a risk-informed application. The risk increase is less than the 1×10^{-5} CDF and 1×10^{-6} LERF Category II criteria discussed in Regulatory Guide 1.174 (2.13.2, Reference 2). This would be considered a small change in risk.

2.13.1.2.5 Quality of Ginna PSA

The Ginna Station Level 1 and Level 2 PSA Model was initially developed in response to NRC Generic Letter 88-20 (Individual Plant Examination, or IPE). Since the original IPE submittal, the PSA has undergone several model revisions to incorporate improvements and maintain consistency with the as-built, as-operated plant.

The GPSA Revision 5.0 update involves extensive revision of the human reliability analysis, along with enhancements to thermo hydraulic analysis, fire modeling, station blackout modeling, and steam generator tube rupture modeling. In addition, the RCP seal LOCA modeling is revised to current Westinghouse standards (2.13.2, Reference 10). Overall, the GPSA is reviewed and upgraded with a goal of increased fidelity in areas related to EPU.

2.13.1.2.5.1 Model Peer Review

In May 2002, the Westinghouse Owners Group performed a Peer Review of the GPSA Revision 4.1. The peer review final report was issued in December, 2002 (2.13.2, Reference 9). The GPSA received a grade of full 3 for one of the technical elements, and a grade of 3 with contingencies for all other technical elements. The contingencies were due to outstanding facts and observations (F&O's) identified during the peer review. Since the completion of the review, the majority of the peer review F&O's (all six A-level and thirty-three of thirty-five B-level) have been addressed. A list of all A and B level F&O's, and their resolution, is provided below. The two remaining peer review comments that are not fully addressed (F&O's AS-13 and DE-01) are evaluated to ensure that they do not effect the ability of the model to estimate the risk impact of power uprate, as discussed in the resolution of each of these items.

Finally, engineering calculations document the development of all major elements of the initial and updated versions of the model. These calculations have been independently reviewed and are retained as quality records.

The following is a list of Level A and B observations and resolutions:

INITIATING EVENTS IE-01   LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The LOSP is divided into two parts- loss of the grid and loss of the switchyard. The grid loss includes severe weather and grid disturbances. The process of derivation of grid loss frequency uses generic NSAC data and removes any weather or grid related failure that is not applicable to the Ginna site. But no operating time is removed from the denominator.

The mean value of the generic prior for grid loss is 7.8E-3 with an error factor of 15. When Bayesian updated with the site specific data of zero failures in 14 years, the result is 2.63E-3.

Ginna uses a moment matching Bayesian update code which can produce non-conservative results, particularly when updating with zero failures. The final value of 2.63E-3 is not supported by the generic data or the plant specific data. The low value of 2.63E-3 results from the use of the specific Bayesian update combined with the selection of the error factor. If the EF of the prior is changed to 5, the result is in the range of 7E-3.

There are 3 observations:

1) the elimination of events not specific to Ginna is not appropriate for development of a prior, unless the operating hours of the (non-applicable) plants is also reduced.

2) the choice of an error factor of 15 for the prior biases the posterior.

3) the use of a moment matching Bayesian update code yields answers that can not be supported by the existing data. (Also see related F&O IE-07)

Resolution: The current PSA analysis includes all severe weather phenomena, whether or not feasible at the Ginna site, in the LOOP calculation and uses a non-moment matching Bayesian update process. *Revision:* 4.2

INITIATING EVENTS IE-03 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The initiating event with the highest contribution to CDF in the internal events model is loss of service water (TI0000SW). The dominant contributors to this initiator are commonalties related to the intake and/or screenhouse (SWCXXSUCTI), and failures of the traveling screens. The frequency for event SWCXXXSUCTI is based on engineering judgment. The basis for this frequency is presented in Section 3.4.2.8, in the form of a review of historical records/events related to SW intake. These event provide a good starting point for a quantitative assessment, but as one of the top scenarios contributing to CDF from internal events, the analysis should be strengthened.

The fault tree logic used to quantify the traveling screen failure contribution to TI0000SW should also be revised. The current model combines the independent failure of three traveling screens with the fraction of time the screens are needed. This fraction is, with no documented basis, assumed to be 1.0E-3, equivalent to about 8 hours per year. Also, a common mode failure should be modeled for the failure of the three screens. The failure rates used for the screens in the PSA is based on all periods of operation. The use of the 1.0E-03 fraction in combination with the failure rate requires that the failure rate be developed under the condition of high stress. Therefore, the current failure rate may not applicable during the 8 hours during the year when the screens are assumed to be needed.

Resolution: Enhanced the PSA analysis for common cause failure (CCF) of the service water (SW) pumps and loss of all SW due to loss of the intake structure for the loss of SW initiator. In addition, the PSA final report has been enhanced to provide further discussion. *Revision:* 4.2

INITIATING EVENTS IE-04 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The partial loss of feedwater initiating event was subsumed and quantified as a reactor trip in the GPSA model. This is non-conservative since the model reactor trip model takes credit for recovery of main feedwater.

Resolution: New logic has been added to include specific initiating events for events where one train of MFW is lost and is unrecoverable. *Revision:* 4.3

INITIATING EVENTS IE-05 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The calculation of CCF basic events for support system initiators (i.e., loss of air, total loss of service water, loss of CCW) has some problems. The CCF events appear in the fault trees and utilize Alpha factors from NUREG/CR-5497, but the exposure times associated with the events are set to values less than 8760. For example the global CCF event for service water failure has an exposure time of 72 hours. In the CCW initiating event tree, CCF of both CCW pumps to run is "ANDed" with failure of the standby pump to start. This is a non-minimal cutset, as the CCF event alone fails both pump trains.

This miss-use of CCF events has been identified in several plant PRA peer reviews. The problem stems from the fact that when the CCF parameters from the NUREG are applied over a 8760 "mission time", the resultant system failure frequency is much, much higher than what is suggested by industry data (e.g., zero loss of CCW events in more than 2500 years of commercial operation.) But reducing the mission time for the CCF events is not an appropriate solution.

Resolution: Re-analyzed CCFs for loss of Service Water, Instrument Air/Service Air and CCW initiators. *Revision:* 4.2

INITIATING EVENTS IE-07 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The Bayesian update process for some initiating event frequencies used a moment matching technique (transformation of form lognormal to Gamma back to lognormal). This technique can cause an underestimation of the resultant frequency when the plant specific data indicates zero failures, which is the case for a number of initiators. For example, the updated GPSA frequency for grid related loss of offsite power is 2.63E-3. A more rigorous Bayesian update, without moment matching yields a result of 4.46E-3.

Resolution: Employed a non-moment matching Bayesian update process to calculate initiator frequencies for Ginna Station. *Revision:* 4.2

ACCIDENT SEQUENCE EVALUATION AS-01 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Several items were noted in review of the ATWS sequence logic in the CDF fault tree:

(1) In the ATWS fault tree logic, under Gate TL_LT (failure of long-term reactivity control), under the mechanical rod insertion failure logic, operator failure to implement emergency boration is "ANDed' with operator failure to trip rod drive MG sets (probability 1E-2). But if the rods did not insert as a result of mechanical faults, then the operator action to trip the MG sets could not be effective, and so this action should not be factored into the cutsets here. So the value of the cutsets at TL_LT should be of the order 1E-2 rather than 1E-4 as in the current model.

(2) In the ATWS fault tree, under Gate TL_KE1 (Electrical failure of RTS), failure of both reactor trip breakers is included under an AND gate, which is the common cause failure of the reactor trip breakers. Individual breaker failures are not explicitly modeled; instead, a module (Gate TLCCFBRKRF) is used to represent the effective common cause contribution of breakers RCCBV52RTA and RCCBV52RTB. That is, the failure probabilities entered for the two independent events is the square root of the assigned common cause failure probability. The common cause value assigned is the 5% lower bound value from NUREG/CR-5500 (4.6E-8), but this appears to be an optimistic interpretation of the NUREG values, with no explanation

provided. Since Reactor Trip breaker failure contribution typically dominates electrical failure contribution in other models due to common cause, additional justification should be provided as to why it is insignificant (order of E-8) in this model.

(3) In the ATWS fault tree, under Gate TL_ATWS11, there are several sequences which involve electrical failure of the RTS (as in Gate TL_KE1 noted above) ANDed with other failures and also OPERATORS FAIL TO MANUALLY INSERT RODS OR TRIP MG SETS. But the logic under Gate TL_KE1 includes Operators Fail to Trip Rod Drive MG Sets During ATWS, and the two actions, which appear to be closely related, if not identical, have different basic event identifiers. Thus, the cutsets under Gate TL_ATWS11 credit 1E-4 from operator actions, whereas it appears that a strong dependency between the two actions should be accounted for. (Ginna PSA personnel indicated that these actions had passed the HEP=1.0 screening evaluation, i.e., they had not shown up in cutsets above the truncation when their probabilities were set to 0.1.)

Resolution: The required logic changes have been made to the ATWS portion of the model. *Revision:* 4.2

ACCIDENT SEQUENCE EVALUATION AS-07 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The PSA includes credit for many recovery actions that are performed outside the control room. Accident sequence dependencies such as adverse environment, lack of access, lighting, room cooling, and availability of special tools are not explicitly addressed. As examples, operator actions AFHFDALTTD, AXHFDCITYW, and DGHFDCITYW do not discuss the performance shaping factors associated with performing local actions.

Resolution: Loss of lighting for all operator actions was incorporated in the model. Updated HRA calculations were performed using the EPRI HRA Calculator, which includes performance shaping factors for ex-control room actions. *Revision:* 4.3 and 5.0

ACCIDENT SEQUENCE EVALUATION AS-08 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The RCP seal LOCA model is appropriate. However, according to the write-up in Section 4.2.2.3.2, a 480-gpm/pump leak will result in a LOCA that is equivalent to a 1.08" break. Small-small LOCAs are considered to be < 1", and small-break LOCAs are considered to be 1" - 2". According to this definition, RCP seal LOCAs should be treated as small LOCAs, whereas transfer is made to the small-small LOCA event tree during loss of RCP seal cooling events.

Resolution: A justification for including these LOCAs in the small-small category was developed and added to Section 4.2.2.3.3 of the Final Report. *Revision:* 5.0

ACCIDENT SEQUENCE EVALUATION ***AS-10*** *LEVEL OF SIGNIFICANCE:* ***B***

Peer Review Observation: The following observations were made on the logic of event tree TL, for sequences after failure of event SG.

Event SG questions a faulted steam generator, due to several causes. Event MS asks for isolation of both steam generators in response to the faulted steam generator. Event B1 asks for steam generator cooling from 1 of 2 steam generators. Heat removal is not possible through the faulted steam generator. The fault tree logic of B1 is not sufficient to match the failures in MS and SG to prevent feeding of the faulted steam generator. Ergo, the event tree allows feeding of the faulted steam generator.

This can have an impact on LERF that is not presently included in the model.

Resolution: The fault tree model was updated to correctly address use of faulted/non-faulted steam generator. *Revision:* 4.2

ACCIDENT SEQUENCE EVALUATION ***AS-11*** *LEVEL OF SIGNIFICANCE:* ***A***

Peer Review Observation: The following observations are on the SGTR event tree and top logic:

1) The success criteria in event B1 and L1 are 1/2 SG. Thus, the ruptured steam generator can be used for heat removal. If the ruptured SG is used for heat removal, the end state of the sequences must be cold shutdown, rather than hot shutdown. The end state for success of B1 is hot shutdown.

2) Event I2 asks for closure of the ruptured SG ARV. However, there are sequences where the ruptured SG is used for heat removal. So closure of the ARV is not possible. The logic is not sufficient to capture these as failed states.

3) Event I1 asks for isolation of the ruptured SG. Failure then goes to B1, which allows heat removal with the ruptured SG. Nowhere on this path is event UH2 asked for.

This event tree is not sufficiently detailed to track the faulted and/or ruptured status of the SGs, which is needed to develop probabilities for core melt induced tube rupture.

Resolution: Fault tree model was reviewed and revised to distinguish heat removal in the intact versus ruptured and faulted steam generators. *Revision:* 4.2

ACCIDENT SEQUENCE EVALUATION ***AS-13*** *LEVEL OF SIGNIFICANCE:* ***B***

Peer Review Observation: The Ginna PSA model is a comprehensive model, which includes fire, floods, shutdown, spent fuel pool and fuel handling accidents. All these accidents are included in the same top logic fault tree. The tree is very complex (rightfully so). The tree not only includes all these initiator types, but there are many special phenomena which only pertain to a certain mode, or certain type of event. The tree makes use of AND gates and FLAGS to associate certain phenomena with certain reactor conditions.

The tree is probably difficult to print out. No print out was available for the review. The tree is difficult to review. During the review, the team found 3 (possibly 4) AND gates which should have been OR gates. This is disturbing given the short amount of time afforded to review the tree and the unfamiliarity of the reviewers with the model.

The review team believes that it is likely that there are additional mistakes in the logic structure. It is recommended that steps be taken to simply the tree for review and quality check and that a systematic review of all logic structure be performed.

Resolution: Significant reviews of the model logic and corrections of any errors have been made as a part of developing revisions 4.2, 4.3, and 5.0. In addition, the model is used on a daily basis as part of the 50.65(a)(4) program for the site. For issues associated with the EPU, corresponding parts of fault tree logic development have been checked for correctness. Additionally, cutset results have been evaluated to ensure expected cutsets are present and that cutsets make sense. Based on prior model reviews, and reviews specifically associated with the EPU, model fidelity has been assured for use in the EPU evaluation.

Note: The spent fuel pool model is for information only. The spent fuel pool does not contribute to CDF and is not considered in the EPU risk evaluation. *Revision:* 5.0

THERMAL HYDRAULIC ANALYSIS TH-02 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The SGTR event tree branches to the SBO event tree if station blackout conditions exist. In the SBO tree, top logic for HRX questions the probability of power recovery at "X" hours. For SGTR, this top event is defined as power recovery at 5 hours, based on information in Appendix B.3 of the PSA report. Appendix B.3 states that, based on MAAP run RUH2J, the time to steam generator dryout " ... following a SGTR (0.664 inch LOCA) with only one AFW pump available, ...the SG dries out at 4 hours with fuel damage at 5 hours." A check of the available information for MAAP run RUH2J (as provided in Table 4-2 of the PSA Report and in a fax of MAAP plots included in a notebook with a May 28, 1996 letter transmitting MAAP analysis results) indicates that time to TCRHOT > 1800 deg F is actually closer to 5.5 hours. But perhaps more importantly the information provided for this case indicates that credit is taken for 2 accumulators. In the SBO event tree (and the associated fault tree logic for SGTR with SBO), accumulators are not required.

There are 3 points to consider regarding the above:

(1) It is not clear that the time to core damage for the scenario modeled in the fault tree (i.e., no credit for accumulators) is applicable to the fault tree model, given the credit for accumulators. If the time to core damage were significantly shorter without accumulators, there could be a significant change in the probability for basic event ACAZDLOSP5 (currently 0.097).

(2) If point (1) were not applicable and there was no significant impact due to the credit for accumulators in the MAAP run, the time to core damage would be 5.5 hours instead of 5 hours. This is relatively important in the event/fault tree logic, because the model effectively assumes that power recovery at X hours avoids core damage. Since the supporting power recovery calcs in Appendix B.3 and B.5 use the values at 5 hours, a supporting MAAP analysis that showed core damage at 5 hours would be invalid. But if there is actually 5.5 hours to core damage (i.e., to allow time for implementing pump startup recovery actions at the 5-hours power recovery time), then the modeling assumptions would be correct. (Note that a similar comment applies to the SBO-related SLOCA recovery X-hour value; the 2.25 hours reported is a core damage time from MAAP but is used as a power recovery time in the model).

(3) The same recovery times and probabilities are used in the RCP seal LOCA model (discussed in Appendix B.4), so the extent of the impact of the error is broader than SGTR-SBO.

Resolution: The fault tree was updated to address the need for accumulators for SGTRs and small LOCAs under SBO conditions consistent with the MAAP runs referenced in PSA Final Report Appendix B. *Revision:* 4.3

THERMAL HYDRAULIC ANALYSIS TH-03 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: This observation provides some comments on interpretation and documentation of analyses that support PSA success criteria.

A relatively large number of MAAP analyses were performed in the past for transients, SGTR, SLOCAs, etc., and high level results for all the cases are summarized in Table 4-2 of the PSA Report. It is not necessarily clear from the documentation in the table, and in the limited other available analysis results information, what the various cases are supposed to demonstrate, if they actually support a modeled success criterion, etc.

An example is for SGTR. Among the sensitivity cases run are cases RUH2F and RUH2G, which vary the value of MAAP parameter VFSEP (case 2F uses a value of 0.3, case 2G uses a value of 0.7, the value used in cases 2A through 2E is not stated but a check of an available MAAP parameter file listing for Ginna showed a value of 0.6, which is near the upper end of the MAAP User Manual range of allowable values of .01 to .65).

Per the MAAP user manual, VFSEP specifies the maximum void fraction value at which natural circulation cooling can occur; for void fractions above the specified value, phases separate, and a reflux cooling heat transfer mode is used, which is less efficient. The reported results for case 2F, with the lower VFSEP, show better cooldown whereas the results for case 2G with the higher VFSEP show core damage.

Several observations are offered:

(a) It is interesting that the MAAP analyst apparently recognized a potential sensitivity of the SGTR results to the value of VFSEP and thought to check on the appropriateness of the value used and sensitivity to other values. But apparently no documentation of the conclusions or insights reached based on the sensitivity analysis is available, and the MAAP analyst(s) are no longer with RG&E. The reviewers were aware of an EPRI document (TR-100167) that indicates that the 0.6 value used for Ginna is the recommended value, and that no sensitivity analyses are needed.

(b) The results of the VFSEP sensitivities performed seem counter-intuitive in that as VFSEP is increased, such that presumably better heat transfer can occur longer, the results get worse. An explanation of what is going on in the analyses would help improve confidence in the results.

(c) There are several other cases (e.g., RUH2C and RUH2D) where it is not clear why the variations were run and for which the results of one case or the other (in this case 2D) are not clearly success and may be sensitive to the value of parameters such as VFSEP. In this particular instance, the "Result" for 2D says no core melt but RCS voiding. If this case were important to determining success criteria, its sensitivity to VFSEP could also be important. In addition, the available plots for these cases, for which it is stated that cooldown is via the intact SG, imply instead that cooldown is occurring via the ruptured SG.

Resolution: The original MAAP runs were confirmed or updated using PCTRAN. HRA event timing that effects the EPU have been confirmed or modified based on updated PCTRAN runs. *Revision:* 5.0

System Analysis SY-02 Level of Significance: B

Peer Review Observation: ES 1.3, step 5 indicates that if only one CCW pump is available (due to pump failure, lack of electric power support, etc.), then operators must isolate nonessential CCW loads and align CCW to only one RHR heat exchanger. There is a high-level operator action in the model for aligning for recirculation, and aligning CCW is part of the process of aligning for recirculation. However there is not a specific operator action for the case that a CCW pump is failed and some potential failure combinations are not being developed as cutsets. Model fidelity would be better if a specific operator action was incorporated at a lower level in the logic as an input to the specific impacted components.

Resolution: A review of this operator action indicates that the current HRA event is appropriate and no change is warranted.

System Analysis SY-03 Level of Significance: A

Peer Review Observation: Several fault tree gates were modeled as AND gates, when the logic implies they should be modeled as OR gates. Three examples are as follows:

a) Gate TL_D_CD b) Gate AF686A c) Gate AX950XZ

Resolution: The fault tree model has been updated to correct the modeling issues. *Revision:* 4.2

Data Analysis DA-01 Level of Significance: A

Peer Review Observation: The Ginna PSA uses moment matching in the Bayesian update process for developing component failure rates and initiating event frequencies. Lognormal distributions are transformed into Gamma or Beta distributions, then the update is performed, and the resultant distribution converted back to a lognormal form. This method produces good (i.e., approximately equal to more rigorous methods) posterior mean values when the plant specific data consists of a non-zero number of events. However, when the evidence consists of zero failures in "n" demands (or hours of operation), this method will consistently under-estimate the mean value of the posterior distribution. As an example, the posterior mean for "AF AV C" in Table 7-5 is listed as 1.75E-4. When this update is performed (0 failures in 884 demands) rigorously by updating the discrete lognormal probability distribution directly (using ERIN BART software), the result is 4.77E-4, nearly a factor of 3 higher than the Ginna PSA value.

Although the problem only occurs when updating with zero failures, it is noted that 184 of the 278 component failure rate updates listed in Table 7-5 involve updates with zero failures.

An additional observation regarding this method is that when updating with zero events, regardless of the number of demands, the error factor of the posterior is equal to the error factor of the prior. This is apparently another weakness of this method.

Resolution: PSA calculations now use an updated Bayesian technique which does not employ moment matching. *Revision:* 4.2

Data Analysis DA-04 Level of Significance: A

Peer Review Observation: The mean value used in the PSA for failure of the turbine driven AFW pump to start on demand was grossly under-estimated due to an error in the Bayesian update process. The wrong distribution was selected as the prior in the calculation of the subject failure rate (AF TP A).

Resolution: Used the correct prior in calculations; results included in the model. *Revision:* 4.2

Data Analysis DA-06 Level of Significance: B

Peer Review Observation: The basis for RPS control rod and reactor trip breaker CCF frequencies should be revised and/or better documented. The following information is taken from Table 7-6 of the PSA report.

Control Rod - Fails to insert mechanically	2E-7 (5th %tile from NURGG/CR-5500, T3, Rod)
Control Rod - Fails to insert electrical	1.6E-6 (mean from NURGG/CR-5500, T3, BME)
Reactor Trip Breaker - Fails to open	4.6E-8 (5th %tile from NUREG/CR-5500, T3, BME)

There is no documentation regarding the use of 5th %tile values from the source as mean values in the PSA. There is no documented basis for using the 5th %tile values. It appears that the control rod CCF failure modes above should be combined and use the mean value for ROD from Table 3 of NUREG/CR-5500. The reactor trip failure mode should use the mean value listed for BME in the same table.

Resolution: The RPS/reactor trip breaker logic has been revised and reviewed. *Revision:* 4.2

HUMAN RELIABILITY ANALYSIS HR-02 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Table 7-15 states that the screening value of 0.01 for operator action RCHFD00MRI was derived from page B-7 of WCAP-11993. However, WCAP-11993 gives the HEP for manual rod insertion (MRI) as 0.1, not 0.01

Resolution: The correct value of 0.1 has been used for this action. *Revision:* 4.2

HUMAN RELIABILITY ANALYSIS HR-04 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: All pre-accident HEPs were quantified using screening values of 3.0E-3, consisting of 0.03 for the basic HEP times 0.1 for recovery. While no one pre-accident HEP has high risk significance, using screening values for pre-accident HEPs could have an impact on the risk assessment for maintenance configurations. While it is understood that many pre-accident HEPs may be identical due to identical processes (e.g., failure to restore a component following testing usually involves an independent verification, and failure to restore a component following maintenance involves performing a post-maintenance test), plant-specific HEPs could be derived for each type of error and applied to each type of activity.

For example, the HEP for failing to restore a component after maintenance typically includes restoring the pump suction and discharge valves. Assuming that a post-work test is performed, the HEP for such a case could be calculated as 2 x 0.03 x 0.01 x 1.6 x 1.6 = 1.5E-3, where the 1.6 factors are used to convert the median HEPs to mean values. For test restoration errors, using ASEP would result in a mean HEP of 0.03 x 0.1 x 1.6 x 1.6 = 7.7E-3, which is actually higher than the screening HEP (which uses median values).

The HEP associated with miscalibrations would need to consider the recovery mechanisms available.

Resolution: The methodology for pre-initiator HEPs has been updated to address these issues, and the final report enhanced. *Revision:* 4.3

HUMAN RELIABILITY ANALYSIS HR-08 LEVEL OF SIGNIFICANCE: A

Peer Review Observation: The method and process for assessing dependent human actions does not meet the objectives of the peer review guidance. The following observations were made of the current process to identify and quantify dependent human actions:

1) A systematic search was made of the quantified cutsets for dependent HEPs in the same

sequence. The starting point for the search, however, was the set of cutsets quantified at a 1E-10 cutoff. Thus, many of the "untreated" HEP combinations could have been eliminated.

2) There is no analytical process for HEP adjustment for multiple HEPs in the same sequence. The adjustment process was to adjust the last HEP in the sequence to a value of 0.1.

3) This process resulted in adjustments for a limited number of HEP combinations. Several of the common HEP combinations found in other PRA's (AFW / MFW / F&B) were not represented.

4) Even with the correction factors, some of the HEP combinations found in the final cutsets had very low combined failure probabilities, e.g., 1.7E-6 (LISSLOCA), 1.6E-7 (FL00TB6), 7.6E-7 (TIIAWTS)

Some expected combinations that were not found are:

MFHFDMF100 * RCHFD01BAF * AXHFDSAFWX

MSHFDISOLR * RCHFDCDPPR * RCHFDCDTR2 * RCHFDCOOLD

AFHFDSUPPL * MFHFDMF100 * AXHFDSAFWX * RCHFD01BAF

Resolution: An enhanced approach for post-initiator HRA, which identifies dependencies *a priori*, not just for HEP's in dominant cutsets, was used. *Revision:* 4.2

HUMAN RELIABILITY ANALYSIS HR-09 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Event AFHFDSUPPL:

This event is discussed in the HRA notebook as if it is the refill of the CST, from other sources. The timing, and PSF seem to consider that the CST is to be refilled from hotwell or elsewhere.

However, the fault tree uses this event as input to gate AF460, which is an AND gate. The fault tree appears to consider the CST is refilled automatically from the hotwell, and this event "SUPPL" is for refill of the CST from other sources after the hotwell is depleted. If this is true, then the cues and PSFs are inappropriate.

Alternatively, gate AF460 could be an OR gate, and the PSF for this event would be correct.

Resolution: The methodology for HEP calculation and dependent HEP quantification was updated. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

Human Reliability Analysis HR-10 Level of Significance: B

Peer Review Observation: All HEPs are quantified as if they are the only HEP in the sequence, ignoring the other actions that will require time and effort. For example, in Loss of SG cooling sequences, the process of events will be:

1) Reactor trip

2) AFW fails - MFW is attempted to restore

3) MFW fails - SAFW is attempted to align

4) SAFW fails

5) Feed and Bleed attempted.

However, the timing for actions applicable to these sequences do not consider the other actions.

RCHFD01BAF uses a compelling signal cue at 9 minutes

AXHFDSAFWX uses a compelling signal cue at 10 minutes

MFHFDMF100 uses a compelling signal cue at 10 minutes

The diagnosis errors for these HEPs are:

RCHFD01BAF = .0032

AXHFDSAFWX = .00261

MFHFDMF100 = .008

(Note that these numbers imply it is more difficult to realize the need for MFW than for SAFW and F&B. In reality, MFW would be the first system for SG heat removal after AFW failed.)

A more realistic analysis would consider the compelling signal for MF to be at 9 minutes. The compelling signal for SAFW would be sometime after MFW is known to be failed and the compelling signal for Feed and bleed is when SG water level reaches the cue indicated in FRH.1 (certainly not 10 minutes).

Resolution: The methodology for HEP calculation and dependent HEP quantification was updated. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

Human Reliability Analysis HR-12 Level of Significance: B

Peer Review Observation: The diagnosis errors are calculated using either the annunciator response model or the time based crew response model. Although both these methods are used correctly, there are no criteria as to which applies in each situation. It appears that the lower probability was used when desired.

Resolution: The PSA final report was enhanced to include a discussion regarding when to use the Annunciator versus the Time Response diagnostic models in ASEP. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

Human Reliability Analysis HR-14 Level of Significance: B

Peer Review Observation: Event RRHFDCOOLX

This event has a screening recovery of 0.1 for long term RHR sequences. There is no basis for the application or probability of the event. It is assigned as a "screening value" and therefore can seemingly be assigned anywhere without justification.

RRHFDSUCTN is also a screening value used with no apparent justification.

If both of these were eliminated, internal events CDF would increase 6%.

Resolution: HEP values were calculated for these events. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

Human Reliability Analysis HR-15 Level of Significance: B

Peer Review Observation: DGHFCITYW = .0966.

This event is to align city water to the DGs in the event SW fails. The HRA analysis states the time window is 86 to 263 seconds to establish water before the DGs fail. The PRA uses 4 minutes for a diagnosis time, which is the upper bound of the time interval. In addition, if 240 seconds are used for diagnosis, this leaves only 23 seconds to align the city water.

If the true time is 86 seconds, the action cannot succeed.

If 4 minutes are allowed for diagnosis, then there is no time left for action. If 1 minute is allowed for diagnosis, the HEP is much higher.

Resolution: This event has been re-evaluated using an appropriate diagnostic time. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

HUMAN RELIABILITY ANALYSIS HR-16 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: There are 3 events in the PRA that use the same HEP. They are RCHFDCDOSS, RCHFDCDTR2, RCHFDCOOLD. These are all assigned a probability of .0307. This is based on calculation of a dependent probability for a similar event (RCHFDCDOVR), which is not used any longer. These events must be conditional on the failure to prevent SG overfill during an SGTR.

The 3 events listed above appear in many sequences that do not involve SGTR or SG overfill.

Resolution: The three HEP's discussed have been re-examined for consistency and re-quantified. In addition, EPU related HEP events were re-calculated using the EPRI HRA Calculator. *Revision:* 4.2 and 5.0

DEPENDENCIES DE-01 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The specific comments below were generated during the review. Some may be resolvable by providing available documentation that the reviewers were unaware of. However Ginna PRA staff acknowledges that the level of documentation detail is limited enough that it presents a problem to analysts outside of the group. It is therefore important that additional documentation of detail be performed.

- Need documentation of impacts of initiators on the model. For example, "Flood Scenario FL00SH1 fails components X, Y, X...."

- Discussion of propagation sources were provided but were limited and hard to follow.

- Affected components must be inferred by looking at what the flood initiator is an input to in the fault tree. A listing would facilitate review – and use of the flood analysis.

- Analytical approach first defined a flood frequency for a space based on a semi-generic data set and then apportioned it according to what was felt to be important. A clear description of a systematic approach for how this was done was not found.

- Initiating event logic for service water contains gate LSW001. Beneath LSW001, the probability of air temperatures below 30 F is given as .133 and the probability of cold lake temperatures is given as .0166. They are apparently treated as independent events but it would seem that they are highly dependent. Correction or explanation as appropriate is suggested.

- Logic below gate TL_RH3Y "ands" TL_SB_F1 (fire recoveries) and FLN700 (screenhouse recoveries). This may be correct as intended but it seemed like the gate should be an "or." Correction or explanation as appropriate is suggested.

- Flood rates were not provided, but were discussed in terms of "very large, large, etc." Specific flood rates would have been helpful in the review. Also necessary to calculate operator response times. Generally sump capacities were not discussed, however for the more significant floods this does not appear to matter.

- Suggest providing a clear listing of affected SSCs.

- Need more discussion about interaction between flood frequency apportionment in Table 7-9 and additional frequency apportionment in fault tree (i.e., screenhouse flood FL000SH1 receives a "flood size apportionment" in Table 7-9 and also an additional flood size apportionment in the model).

Resolution There are several issues discussed by this F&O. The suggested resolution of this item is to consider revising the documentation to address the issues. Two of these issues require model logic correction (bullets 5 & 6). These issues have been corrected in the model. All remaining items are solely related to documentation, and will require enhanced explanation in the final report, but have no impact on the Revision 5.0 results. *Revision: 5.0*

DEPENDENCIES DE-02 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Operator actions during floods should be reviewed.

• Instances were found where "normal" internal model operator recovery actions appeared in flood sequences (e.g. AXHFDCITYW), without a change in probability. This may be appropriate but the events should be reanalyzed to be sure, looking at staffing requirements, operator burden, cues, physical access issues, etc. For the example it is not obvious that the task of aligning city water would be as easy for a crew to accomplish during a flood as during a relatively normal trip which required extended AFW operation. If the detailed HRA analysis for this event under flooding conditions was provided, it would help support this model assumption.

• It is not clear that operator actions to isolate certain floods are being modeled at the appropriate level of detail in the model. For example, if a Service Water header fails in the aux building, or to the diesels, etc., it must be isolated and this isolation will impact what supported components receive service water and how. It is not clear that there are specific operator actions that address this.

• Event IFAZCIBFLI, "Intermediate building flood isolated before significant accumulation," appears to encompass an implicit operator action. No dependency assessment of this with other actions was noted.

• Aux. building floods of a certain size are assumed to be isolable by isolating valve 4734. It's not clear that the model *.fre file is correctly taking this pathway out when it should (it may be, but it wasn't obvious during review). In addition, it appears that the possibility of a flood in the parallel SW line (isolable by 4735) was not considered.

Resolution: A dependency analysis for flood and fire scenarios for HRA events has been performed and the model updated as appropriate to address bullets 1 and 3. Bullets 2 and 4 have been addressed by updating the model logic. *Revision: 5.0*

DEPENDENCIES DE-04 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Not clear that accident sequences were redefined for use in floods, or that existing sequences which are used were reviewed to ensure applicability. This process may have been performed but it was not clear.

Resolution: This issue has been addressed by completing the dependency analysis for flood and fire scenarios. *Revision: 5.0*

STRUCTURAL RESPONSE ST-01 LEVEL OF SIGNIFICANCE: A

Peer Review Observation: Basis for operator recovery for ISLOCA TL_LIPEN140 (failure of RHR shutdown cooling suction isolation valves and RHR suction piping) isn't clear. It would seem that failures of some suction piping sections would not be isolated by the actions proposed; some failures are apparently not isolable. Also, no modeling of sequences after a successful recovery appears to exist. If recovery is possible, there would presumably be a reactor trip with unavailability of RHR and there could even be environmental impacts on other plant systems due to the ISLOCA. This should be modeled.

Resolution: The entire ISLOCA analysis was updated. *Revision:* 4.2

QUANTIFICATION QU-05 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: At present only a parametric uncertainty analysis has been performed. Areas where additional sensitivity calculations should be performed include cases where thermal-hydraulic analyses predict only small margins for success in terms of the number of trains required or the time available for operator actions. One specific example is the impact of 1-of-2 PORVs for success in feed & bleed cooling versus 2-of-2 PORVs as contained in the actual EOPs.

Resolution: Detailed thermo hydraulic analyses have been performed for EPU related success criteria, including number of PORVs required for bleed and feed cooling, to ensure the correct success criteria are used. Further, the human action sensitivity evaluation shows the impact of changes in human action failure likelihood. As thermo hydraulic analyses changes could result in changes to human action failure likelihood, this sensitivity study also provides an indication that thermo hydraulic analysis changes are not likely to affect the conclusion that the EPU is acceptable. *Revision: 5.0*

QUANTIFICATION QU-06 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: According to Section 8.2.1, interfacing system LOCAs (ISLOCAs) were screened at 1E-7/yr, citing GL 88-20 as justification. Given the relatively high conditional CDFs for ISLOCAs, their importance to LERF, and the cumulative impact of ISLOCA sequences which may have just fallen below the truncation value, the truncation limit should be justified. In addition, the truncation limit used should consider the impact of being in a configuration that could result in a relatively high CDF.

Resolution: The entire ISLOCA analysis was updated, including detailed calculations for ISLOCAs previously screened out as well as additional scenarios. *Revision:* 4.2

<u>QUANTIFICATION QU-07 LEVEL OF SIGNIFICANCE: B</u>

Peer Review Observation: In quantification of the V-sequence frequency and any other cutsets whose frequency is proportional to X**N where X is a failure rate and N is a number of independent events in the cutset having the same failure rate, the mean frequency is not equal to the Nth power of the mean failure rate. For N=2 and the case where X is lognormally distributed,

X2 = M2 + V,

where M is the mean failure rate and V is the variance of the lognormal distribution. The problem is more complicated with N>2. When dealing with the V-sequence the failure rates are very low and the variance is very high such that the variance term dominates. When this is taken into account the Mean V-sequence frequency can easily be an order of magnitude greater than the result obtained using a mean point estimate (M2). It is not clear that this has been taken into account in the V-sequence quantification.

Resolution: Updated entire ISLOCA analysis. *Revision:* 4.2

<u>QUANTIFICATION QU-10 LEVEL OF SIGNIFICANCE: B</u>

Peer Review Observation: The dominant sequences show a station blackout cutset with CCF of all SWP fail to run: TIGRLOSP*SWCCFPUMPR_ALL. There is no similar cutset for CCF of all SWP fail to start. (SWCCFPUMPS_ALL). All pumps must restart after LOSP, so the additional cutset should be accounted for.

Resolution: Common cause failure of all service water pumps to start following a LOOP event has been included in the model. *Revision:* 4.2

<u>CONTAINMENT PERFORMANCE L2-03 LEVEL OF SIGNIFICANCE: B</u>

Peer Review Observation: There are several event probabilities and split fractions in the LERF model whose basis is not explained. The probabilities appear to be an estimate of the analyst and are not reproducible without additional documentation. These events are:

CTAZAUXBLD - AUX Building scrubbing

CTAZEARLY2- containment failed or bypassed late

CTAZLATEFT- filtered or submerged leak path

CTAZSGSMLL-SG leaks will not lead to rapid depressurization

CTAZSGTRST-SG inventory scrubs release.

The probabilities chosen for these events range from 0.01 to 0.5. The probabilities have a dramatic effect on LERF. The probabilities appear to be analyst judgment.

Resolution: Events CTAZAUXBLD, CTAZEARLY2, CTAZLATEFT, and CTAZSGTRST could not be justified and were removed from the model. CTAZSGSMLL is justified for smaller steam leaks and has been better described within the final report. *Revision:* 4.3

CONTAINMENT PERFORMANCE L2-04 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The LERF model appears to follow NUREG/CR-6595, but includes several unique features, which are not explained and not substantiated. Some of these ideas may be more advanced than the NUREG LERF model, but are not generally included in other PWR LERF models nor included in the NUREG. The purpose and basis should be explained in sufficient detail.

These items are:

1) AUX Building scrubbing of ISLOCA releases. No basis is provided to guarantee the release is through the AUX building or to establish that the HVAC system can keep up with the release if it is large.

2) scrubbing of SGTR releases.- No basis is provided to show that water will be in the SG or that the leak will be submerged.

3) fatalities from late releases- most LERF models do not discuss fatalities, but only consider LERF. No basis for the fatality split fraction was provided.

4) reduction in fatalities for early release - most LERF models do not discuss fatalities, but only consider LERF. No basis for the fatality split fraction was provided.

Resolution: The split fractions addressed here have either been removed or justified. *Revision:* 4.3

CONTAINMENT PERFORMANCE L2-05 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Emergency Action Levels are not included in the LERF model.

Resolution: The root concern of the peer review comment was that some core damage events were not considered to be LERF even though the timing of the emergency action levels as related to release time were not explicitly evaluated. To address this, all releases that are not the result of long term containment over pressurization due to a lack of containment cooling are considered a large early release. Long term containment over pressurization due to a lack of containment cooling events will not result in containment failure until well after a general emergency. A general emergency would be declared shortly after core damage occurs. *Revision: 4.3*

CONTAINMENT PERFORMANCE L2-06 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: There are eight human interactions (HIs) that are labeled "for Level 2 only." The following things were not considered in estimating the failure probabilities for these:

(1) In a post-core damage event, the radiation in certain areas of the plant could be extremely high. No assessment of the increased stress due to high radiation has been made.

(2) Once core damage occurs, operators are directed to exit the Emergency Operating Procedures (EOPs) and enter Severe Accident Management Guidelines (SAMGs). The SAMGs are not step-by-step "cookbook" procedures like the EOPs. Neither ASEP, nor THERP, nor any other HRA method is designed for this situation.

A cursory look at the HI descriptions reveals that (with the possible exception of CTHFDLOCLX), these HIs should be begun before core damage occurs. The timing implies, however, that they could be delayed until after core damage occurs.

Resolution: The fault tree has been updated and now contains only three human actions specifically related to Level 2. Two of these actions occur prior to core damage, while the third takes place in the control room where radiation levels are not an issue. *Revision: 5.0*

MAINTENANCE & UPDATE MU-01 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Element MU-4 identifies a list of information inputs which should be monitored to ensure that the PSA is kept up to date. It seems clear from discussions with Ginna PRA personnel that these inputs are being monitored but it is not clear that there is a formal requirement that they be monitored. Some elements are currently being tracked by virtue of the PRA supervisor's presence on various plant committees. A formal listing of the data sources to be monitored would better meet the requirements of sub-element MU-4.

Resolution: Ginna implemented a process to ensure the PSA matches the as-built, as-operated plant (see 2.13.1.2.5.2, Ginna PSA Maintenance and Update, for details).

MAINTENANCE & UPDATE MU-02 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: Current PRA update procedure requires notification of "process owners," i.e. owners of programs which rely on PRA products, when a significant PRA change occurs. However the risk impact of PRA changes is apparently not evaluated unless a process owner requests it.

The intent of the peer review guidance seems to be that PRA products should be evaluated whenever the PRA is changed, whether or not this is requested. This evaluation can be at a screening level if appropriate but it should be performed and documented.

Resolution: Procedures have been revised to require generation of a tracking item to track updating of risk-informed processes, if not done at time of PSA revision release.

MAINTENANCE & UPDATE MU-04 LEVEL OF SIGNIFICANCE: B

Peer Review Observation: The Ginna PSA and EOOS model update procedure (EP-3-S-0710) provides a process that requires documentation of a review of each model change request. This process is executed through the use of the EOOSCRF forms.

In general, consistent documentation of a technical review process is lacking with respect to the Ginna PSA. Although many work packages (e.g., DA-MS-99-002 and others) have signoff sheets, and are signed off by a preparer and a reviewer, technical elements of the PSA documented in the PSA have no documented review. Examples include the initiating event selection and grouping, component failure methodology and quantification, system analyses including support system dependencies, operator inputs to the human reliability analysis, and others.

Based on reviewer discussions with the Ginna PSA staff, it is apparent that additional reviews have been performed for some analyses, but it is also recognized that documented technical reviews are not being done on a consistent basis.

Resolution: Ginna implemented a process to ensure the PSA matches the as-built, as-operated plant (see 2.13.1.2.5.2, Ginna PSA Maintenance and Update, for details).

2.13.1.2.5.2 Ginna PSA Maintenance and Update

The GPSA is a living document that is updated and maintained to adequately reflect the as-built, as-operated plant. A procedurally controlled change impact evaluation process ensures that changes to the plant are reviewed for impact on the PSA. This process is integrated with the Ginna Plant Change Process, Equivalency Evaluation Process, and Setpoint Change Process such that the originator of the change and a PSA engineer determine if the change impacts the PSA. In addition, the procedure change process requires that any change, addition, or deletion of operator actions, or change to step sequence, in the Ginna Emergency/Abnormal Operating Procedures is reviewed for impact on the PSA.

Changes to the PSA are also procedurally controlled. Changes to the fault trees, databases, and the Final Report require documentation and an independent review.

2.13.1.2.5.3 Other Relevant Open Items

There are no other known open items.

2.13.1.2.5.4 Software

THERMO HYDRAULIC SOFTWARE

The timing changes for inventory control and decay heat removal losses are calculated using PCTRAN. PCTRAN is a graphical interface thermo hydraulic code. PCTRAN was benchmarked in-house against MAAP, LOFTTR2, and NOTRUMP. The most extensive benchmarking compared MAAP results for a wide array of decay heat removal and inventory control scenarios to the PCTRAN results for the same scenarios. PCTRAN is compared to LOFTTR2 using the USFAR SGTR Overfill Case. PCTRAN is compared to NOTRUMP for several LOCA cases. Further details can be found in the PCTRAN Ginna Benchmark Document.

As bleed and feed (BAF) is one of the more challenging cases for thermo hydraulic codes, Westinghouse performed an independent review of the PCTRAN modeling as it relates to BAF. Westinghouse used hand calculations to confirm that the PCTRAN results are reasonable (for details, see 2.13.2 Reference 14).

CAFTA

The Computer Aided Fault Tree Analysis (CAFTA), Version 5.1 tool is used to perform probabilistic risk analysis using a linked event tree/fault tree methodology. Details about CAFTA are available on the Electric Power Research Institute (EPRI) website.

HRA CALCULATOR

The HRA Calculator Software, Version 2.01: Human Reliability Analysis is used to determine HRA failure likelihoods. Details about the HRA calculator are available on the Electric Power Research Institute (EPRI) website.

2.13.1.2.6 Technical Evaluation Conclusion

The Ginna Station power uprate will include small increases to internal events initiator frequencies and, in general, will reduce the time available for operator recovery actions. The small increases to initiators contribute twenty-seven percent of the risk increase associated with EPU. The reduction in the time available for human actions contributes sixty-three percent of the risk increase associated with EPU. Other factors contribute the remaining ten percent of the risk increase.

No new vulnerabilities are introduced regarding fire, seismic, wind, or shutdown mitigation. Although no new vulnerabilities are introduced, the time available for operator actions decreased. This causes a risk increase for not only internal events, but for external events and shutdown operations as well.

In general, fire and shutdown risk had a larger percentage increase than internal events. Certain fire events can cause low steam generator water level trips and force operations to mitigate the accident outside the control room. Reducing the amount of response time available in scenarios where the time is already limited has a large impact. The same is true for reduced inventory shutdown scenarios.

Considering the existing plant configuration and procedures, the power uprate increases the Ginna Station internal events (including flooding) CDF by 2.7×10^{-6} per year (an increase of about eleven percent) for a power increase from the original 1550 MWt to 1811 MWt. The increase in internal events LERF is 2.8×10^{-7}.

When external events and shutdown risk are considered, the power uprate increases the Ginna Station CDF by 7.6×10^{-6} per year (an increase of about twelve percent), for a power increase from the original 1550 MWt to 1811 MWt. The increase in LERF is 4.7×10^{-7}.

These increases are small. Although this license amendment is not being requested as a risk-informed change, these risk increases meet the Regulatory Guide 1.174 Category II criteria of 1×10^{-5} for ΔCDF and 1×10^{-6} for ΔLERF. Therefore, the requested extended power uprate poses a small and acceptable risk.

To provide further confidence in the acceptability of the EPU, these modification/procedures changes will be implemented:

- Credit the use of Safety Injection Pump 1A in the ER-Fire procedures.
- Limit the amount the residual heat removal outlet valves can open on the loss of support.
- Install an air back-up system to ensure charging can be used at maximum speed for at least one hour on the loss of the normal instrument air supply.

The implementation of these modifications is likely to completely or significantly offset the risk of EPU.

The risk assessment also shows that the power uprate does not create the "special circumstances" described in Appendix D of the Standard Review Plan Chapter 19 (2.13.2, Reference 1). The power uprate does not:

- Substantially increase the likelihood of a risk significant accident that is outside of the design basis of the plant,
- Degrade multiple levels of defense,
- Reduce the availability or reliability of risk significant structures, systems, or components, or
- Introduce synergistic or cumulative changes that substantially increase CDF.

The GPSA is updated to keep it consistent with the as-built, as-operated plant and is sufficient for estimating the risk impact of power uprate. Since the Ginna Station power uprate is similar to previous uprates and involves only a small increase in risk, the power uprate is considered acceptable with respect to Appendix D of Standard Review Plan, Chapter 19.

2.13.1.3 Conclusion

The Ginna staff has reviewed the assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the potential impacts associated with the implementation of the proposed EPU are adequately modeled and/or addressed. The Ginna staff further concludes that the results of the risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of the Standard Review Plan, Chapter 19. Therefore, the Ginna staff finds the risk implications of the proposed EPU acceptable.

2.13.2 References

1. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,* Use of Probabilistic Risk Assessment in Plant-Specific Risk-Informed Decision-making: General Guidance, November 2002.

2. NRC Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Rev. 1,* November 2002.

3. Response to December 26, 2002, "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the R.E. Ginna Power Plant," January 31, 2003.

4. GL 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46*, February 19, 1987.

5. NUREG-1488, *Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains*, USNRC, April 1994.

6. EPRI TR-100370, *Fire-Induced Vulnerability Evaluation (FIVE)*, April 1992.

7. Generic Letter 88-20, Supplement 4, *Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*, NRC, June 28, 1991.

8. NUREG-0821, *Integrated Plant Safety Assessment, Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant*, December 1982.

9. *R. E. Ginna Station PSA Peer Review Report*, Westinghouse Electric Co., December 2002; *PSA Peer Review Certification Process Guidance*, SAE/RRA/101(98)-A, Westinghouse Energy Systems, 1998.

10. *RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs*, WCAP-16141, Westinghouse Electric Company, August 2003.

11. NUREG/CR-2300, *PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessment for Nuclear Power Plants*, USNRC, September, 1983.

12. NUREG/CR-4840, *Recommended Procedures for the Simplified External Event Risk Analyses for NUREG-1150*, Sandia National Laboratory, November, 1990.

13. RS-001, *Review Standard for Extended Power Uprates*, USNRC, December, 2003.

14. DAR-OA-05-6 Revision 0, *R. E. Ginna Bleed and Feed Analyses Evaluation of PCTRAN Results*, Westinghouse, April 2005.

2.14 Impact of EPU on the Renewed Plant Operating License

A license renewal application (LRA) was prepared in accordance with the requirements of 10CFR54 for the Ginna Station and was submitted to the NRC in August 2002. The NRC staff reviewed the LRA for compliance with 10CFR54. In May 2004, the *Safety Evaluation Report - Related to the License Renewal of the R. E. Ginna Nuclear Power Plant*, (SER) was issued as NUREG-1786.

The Ginna Nuclear Power Plant, LLC (Ginna) staff review focused on the effects EPU on the evaluations performed for license renewal.

The LRA and SER were reviewed to determine the impact of the EPU on license renewal. Where appropriate, each section in this Licensing Report evaluates the effect of EPU on the structure, system or component (SSCs) under review as well as evaluating the impact to the programs which manage the aging effects on those components. This section presents summary information of the results of that review, and discusses the effects of EPU on SSCs included in the LRA but not discussed in RS-001.

2.14.1 Impact of EPU on Aging Management

The LRA credited a number of existing, modified, and new aging management programs with managing the effects of aging on systems, structures, and components (SSCs) during the period of extended operation. In NUREG-1786, the NRC determined that, subject to license conditions to implement LRA commitments prior to the period of extended operation, these programs provide reasonable assurance that aging effects will be managed such that the SSC intended functions will be maintained during the license renewal period.

Sections 2.1 through 2.13 of this Licensing Report summarize the impact of the EPU on plant accident response and safety, as well as discuss the impact on the license renewal regulated events (Environmental Qualification, anticipated transient without scram, station blackout, pressurized thermal shock, and fire protection) that were the basis for license renewal scoping and screening. A review of these sections of this report has been conducted and confirms that the only additional SSCs relied upon for design basis accident mitigation or license renewal regulated event response as a result of the EPU are the Main Feed Isolation valve operators. Though considered active, these components will be added to the scope of license renewal after installation, in accordance with 10CFR50.37(b). Therefore, impact from the EPU on the license renewal scoping and screening results presented in the LRA and approved by the NRC in NUREG-1786 have been accounted for.

In addition, component-specific sections of this Licensing Report reviewed the impact of the EPU on the SSC aging assessments performed in the LRA. These sections concluded that no new aging effects will result from the EPU. Since there are no new aging effects for the SSCs, the aging management programs presented in Appendix B of the LRA that are credited with managing the effects of aging on license renewal SSCs remain valid for EPU conditions. Therefore, the aging evaluations approved by the NRC in NUREG-1786 remain valid for EPU conditions.

2.14.2 Impact of EPU on Time-Limited Aging Analyses

The time-limited aging analyses (TLAAs) for Ginna Station are presented in Section 4.0 of the LRA. The NRC concluded in NUREG-1786 that the LRA included the list of TLAAs as defined in 10CFR54.3. The staff also concluded that the TLAAs have been demonstrated to remain valid for the period of extended operation, as required by 10CFR54.21(c)(1)(I), or have been projected to the end of the period of extended operation, as required by 10CFR54.21(c)(1)(ii), or that the aging effects are adequately managed for the period of extended operation in accordance with 10CFR54.21(c)(1)(iii).

The impacts of the EPU on the TLAAs are discussed in this Licensing Report. A summary of each of the TLAA discussions is presented below.

<u>Neutron Embrittlement</u> – Section 2.1, Materials and Chemical Engineering, of this report discusses the EPU impact on reactor vessel neutron embrittlement, including analyses for upper-shelf energy, pressurized thermal shock, and pressure/temperature limits. Refer to Section 2.1 for specific conclusions.

<u>Metal Fatigue</u> – Section 2.2.6, NSSS Design Transients, of this report discusses the EPU impact on the fatigue program for the Ginna Station. The TLAAs on fatigue design will continue to be valid after the EPU by projecting that the original transient design cycles remain bounded for the 60-year operating period. The LRA identified reactor coolant pump (RCP) flywheel bore keyway fatigue crack growth as a TLAA. To resolve the TLAA, the LRA committed to continuing to perform inspections in accordance with the requirements of the ISI program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The EPU has not resulted in any change to ISI program inspection commitments for the RCPs; therefore, the commitment made in the LRA to manage fatigue crack growth for the RCP flywheel bore keyway is not impacted by the EPU. The environmental effects on fatigue are discussed separately in Section 2.14.3 (see below).

<u>Environmental Qualification</u> – The EPU impact on Environmental Qualification of electrical equipment and the impact on component qualified life is discussed in Section 2.3.1, Environmental Qualification of Electrical Equipment, of this Licensing Report. Peak post-accident pressures and temperatures remain bounded; however, post-accident radiation levels were calculated to increase. Because of margin in the qualification tests, radiation qualification for 10CFR50.49 scope equipment continues to be met.

<u>Containment Liner Stress and Fatigue, and Containment Tendon Fatigue</u> – Section 2.6, Containment Review Considerations, of this report discusses the EPU impact on containment liner stress and fatigue, as well as containment tendon fatigue. There is no change to the limiting conditions for containment and the liner; post-accident design limits continue to be met,

therefore the conclusions of the TLAAs for containment liner stress and fatigue, and containment concrete tendon fatigue as a result of the EPU remain valid.

Crane Load Cycle Limit – The LRA identified Crane Load Cycle Limit as a TLAA and stated that each of the crane estimated cycle numbers is well below the upper Design Loading Cycle limit. The LRA also stated that the average percent of the rated load lifted was well below the 50% level, relative to the design load cycles, as set forth in the design criteria. The EPU will not have an effect on the number of design cycles experienced by the cranes, nor will the rated load change; therefore, the EPU has no impact on the Crane Load Cycle Limit TLAA discussion in the LRA.

Thermal Aging Embrittlement – The LRA stated that flaw tolerance analyses have been performed to evaluate the reduction in fracture toughness due to thermal aging in cast austenitic stainless steel reactor coolant system (RCS) elbows and the RCP casing through the extended period of operation. The LRA also stated that the results demonstrate that large margins exist for postulated flaw sizes against flaw instability. LR section 2.1.6, Leak-Before-Break, presents the impact of the EPU on flaw tolerance analyses to evaluate the reduction in fracture toughness due to thermal aging of cast austenitic stainless steel RCS elbows and the RCP casings through the extended period of operation. The analyses supporting the EPU demonstrate that large margins still exist for postulated flaw sizes against flaw instability; therefore, there is no change to the conclusions in the LRA of the TLAA for thermal aging embrittlement as a result of the EPU.

2.14.3 Impact of EPU on Environmentally Assisted Fatigue Evaluations

The LRA committed to implement a fatigue monitoring program as a confirmatory program prior to the period of extended operation to ensure that design cycle limits are not exceeded. This program tracks transients and cycles for RCS components that have explicit design transient cycles to ensure that these components stay within their design basis. In addition, the LRA described evaluations performed for component locations listed in NUREG/CR-6260 that are applicable to an older vintage Westinghouse plant for the effect of the environment on the fatigue life of the components. The LRA reported that the evaluations performed confirmed that, with the environmental correction factors applied to the calculated fatigue usage factor at those component locations, all locations were found acceptable for the period of extended operation, with the exception of the pressurizer surge line. For the pressurizer surge line, the LRA committed to perform additional research prior to the period of extended operation.

Based on the information contained in the LRA and the technical responses to several requests for additional information (RAIs), the staff found in the SER that the fatigue monitoring program provides an acceptable program for monitoring the environmental fatigue usage of fatigue-sensitive locations in accordance with the requirements of 10CFR54.2(c)(1)(iii). Confirmatory Item 4.3-2 was identified for resolution of the environmental effects on pressurizer surge line fatigue. Subsequent to LRA submittal, Ginna provided a letter dated September 16, 2003 to address this issue. The information presented in this letter resolved the issue of the environmental effects on pressurizer surge line fatigue in that the limiting surge line locations developed a CUF of less than 1.0, including operation throughout the period of extended operation. The resolution of this issue was found acceptable by the staff in the SER, and Confirmatory Item 4.3-2 was closed. In its SER, the staff found that Ginna had provided an acceptable demonstration, pursuant to 10CFR54.21(c)(1), that, for the metal fatigue TLAA, the effects of aging on the intended functions will be adequately managed during the period of extended operation.

Ginna has evaluated the impact of the EPU on the environmentally assisted fatigue evaluations performed in support of the LRA and the resolution of Confirmatory Item 4.3-2. As reported in the LRA, the fatigue-sensitive component locations chosen in NUREG/CR-6260 for the older vintage Westinghouse plant were:

- Reactor vessel shell and lower head (lower shell at the core support pads)
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including hot leg and pressurizer nozzles)
- Reactor coolant piping charging system nozzle
- Reactor coolant piping safety injection nozzle
- Residual heat removal system Class 1 piping

The calculations used to support the LRA and Confirmatory Item 4.3-2 conclusions served as the basis for evaluating the impact of the EPU conditions on these conclusions. Calculations have been performed addressing the impact of the uprate conditions on the environmental fatigue evaluations of the NUREG/CR-6260 locations. Based on the results of these calculations that show the cumulative usage factors for the fatigue-sensitive component locations to be less than 1.0, all component locations were determined to be acceptable for the period of extended operation.

2.14.4 Conclusions

The Ginna Staff has reviewed the effect of EPU on the Renewed Plant Operating License. Based on this review Ginna concludes that the effects of EPU renewed operating license have been accounted for and the aging effects of the SSCs within the scope of license renewal will be adequately managed through the extended period of operation.

Appendix A

Safety Evaluation Report Compliance

A.1 Safety Evaluation Report Compliance Introduction

This Appendix is a summary of NRC-approved codes and methods used in LR section 2.8.5, "Accident and Transient Analyses" for the Ginna Extended Power Uprate. The appendix addresses compliance with the limitations, restrictions, and conditions specified in the approving safety evaluation of the applicable codes and methods (RS-001 Section 2.1 Matrix 8 Note 7).

Table A.1-1 presents an overview of the Safety Evaluation Reports (SER) by codes and methods. For each SER, the applicable report subsections and appendix subsections are listed.

Table A.1-1: Safety Evaluation Report Compliance Summary

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s)	Limitation, Restriction, Condition	Report Section	Appendix Section
1.	Non-LOCA Thermal Transients	WCAP-7908-A (Reference A.1-1) / September 30, 1986	FACTRAN	Yes	2.8.5.4.1 2.8.5.4.6	A.2
2.	Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference A.1-2) / February 11, 1999	RETRAN	Yes	2.8.5.2.4 2.8.5.4.2	A.3
3.	Non-LOCA Safety Analysis	WCAP-7907-P-A (Reference A.1-3) / July 29, 1983	LOFTRAN	Yes	2.8.5.7	A.4
4.	Neutron Kinetics	WCAP-7979-P-A (Reference A.1-4) / July 29, 1974	TWINKLE	None for Non-LOCA Transient Analysis	2.8.5.4.1 2.8.5.4.6	Not Applicable
5.	Multi-dimensional Neutronics	WCAP-10965-P-A (Reference A.1-5) / June 23, 1986	ANC	None for Non-LOCA Transient Analysis	2.8.5.4.3	Not Applicable
6.	Non-LOCA Thermal/Hydraulics	WCAP-14565-P-A (Reference A.1-6) / January 19, 1999	VIPRE	Yes	2.8.5.4.1 2.8.5.4.3	A.5
7.	Steam Generator Tube Rupture	WCAP-14882-P-A (Reference A.1-2) / February 11, 1999	RETRAN	None for Steam Generator Tube Rupture	2.8.5.6.2	A.3
8.	ASTRUM BELOCA	WCAP-16009-P-A (Reference A.1-7) / November 5, 2004	WCOBRA/TRAC	Yes	2.8.5.6.3.2	A.6
9.	App K SBLOCA	WCAP-10079-P-A, WCAP-10054-P-A (with addenda), WCAP-11145, WCAP-14710 (References A.1-8 through A.1-12) /	NOTRUMP	Yes	2.8.5.6.3.3	A.7

Table A.1-1: Safety Evaluation Report Compliance Summary						
No.	**Subject**	**Topical Report (Reference) / Date of NRC Acceptance**	**Code(s)**	**Limitation, Restriction, Condition**	**Report Section**	**Appendix Section**
		May 23, 1985				
10	LOCA Hydraulic Forces	WCAP-8708-P-A (Reference A.1-13) / June 17, 1977, WCAP-9735 Rev. 2 (Reference A.1-14)	MULTIFLEX 3.0	Yes	2.8.5.6.3.5	A.8

References

A.1-1 WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989.

A.1-2 WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel, et al., April 1999.

A.1-3 WCAP-7907-P-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984.

A.1-4 WCAP-7979-P-A, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, Jr. and R. F. Barry, January 1975.

A.1-5 WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

A.1-6 WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999.

A.1-7 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.

A.1-8 WCAP-10079-P-A, (Proprietary), and WCAP-10080-A, (Non-Proprietary), NOTRUMP - A Nodal Transient Small Break And General Network Code, Meyer, P. E., August 1985.

A.1-9 WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," N. Lee, et al., August 1985.

A.1-10 WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," C. M. Thompson, et al., July 1997.

A.1-11 WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," S. D. Rupprecht, et al., 1986.

A.1-12 WCAP-14710-P-A, "1-D Heat Conduction Model for Annular Fuel Pellets," D. J. Shimeck, May 1998.

A.1-13 WCAP-8708-P-A, (Proprietary) and WCAP-8709-A, (Nonproprietary), MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, K. Takeuchi, et al., September 1977.

A.1-14 WCAP-9735, Rev. 2, (Proprietary), and WCAP-9736, Rev. 1, (Nonproprietary), MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model, K. Takeuchi, et al., February 1998.

A.2 FACTRAN for Non-LOCA Thermal Transients

Table A.2-1: FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions
1. ***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."***   **Justification**   The FACTRAN code was used in the analyses of the following transients for Ginna: Uncontrolled RCCA Withdrawal from a Subcritical Condition (Ginna UFSAR Section 15.4.1) and RCCA Ejection (Ginna UFSAR Section 15.4.5). Initial fuel temperatures used as FACTRAN input in the RCCA Ejection analysis were calculated using the NRC-approved PAD 4.0 computer code, as described in WCAP-15063-P-A (Reference A.2-1). As indicated in WCAP-15063-P-A, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures.
2. ***"Table 2 presents the guidelines used to select initial temperatures."***   **Justification**   In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: loss of flow, locked rotor, and rod ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA ejection analysis for Ginna. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (Reference A.2-1), include uncertainties and are conservatively high. FACTRAN was not used in the loss of flow and locked rotor analyses.
3. ***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."***   **Justification**   The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA withdrawal from a subcritical condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the RCCA ejection transient, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.
4. ***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."***   **Justification**   Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Ginna {RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5). Therefore, additional justification is not required.
5. ***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."***

Table A.2-1: FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions
Justification At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Ginna (RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5)).
6. ***"Although time-<u>independent</u> mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-<u>dependent</u> mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."*** **Justification** The two transients that were analyzed with FACTRAN for Ginna (RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA withdrawal from a subcritical condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.
7. ***"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."*** **Justification** The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Ginna.

Reference

A.2-1 WCAP-15063-P-A, Revision 1 (with Errata) "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," J. P. Foster and S. Sidener, July 2000.

A.3 RETRAN for Non-LOCA Safety Analysis

Table A.3-1: RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***   <u>Justification</u>   The transients listed in Table 1 of the SER are: • Feedwater system malfunctions   • Excessive increase in steam flow   • Inadvertent opening of a steam generator relief or safety valve   • Steam line break   • Loss of external load/turbine trip   • Loss of offsite power   • Loss of normal feedwater flow   • Feedwater line rupture   • Loss of forced reactor coolant flow   • Locked reactor coolant pump rotor/sheared shaft   • Control rod cluster withdrawal at power   • Dropped control rod cluster/dropped control bank   • Inadvertent increase in coolant inventory   • Inadvertent opening of a pressurizer relief or safety valve   • Steam generator tube rupture   The transients analyzed for Ginna using RETRAN are: • Increase in feedwater flow (Ginna UFSAR Section 15.1.2)   • Steam line break (Ginna UFSAR Section 15.1.5)   • Combined steam generator atmospheric relief valve and feedwater control valve failures (Ginna UFSAR Section 15.1.6)   • Loss of external electrical load (Ginna UFSAR Section 15.2.2)   • Loss of offsite alternating current power to the station auxiliaries (Ginna UFSAR Section 15.2.5)   • Loss of normal feedwater flow (Ginna UFSAR Section 15.2.6)   • Feedwater system pipe breaks (Ginna UFSAR Section 15.2.7)   • Flow coastdown accidents (Ginna UFSAR Section 15.3.1)   • Locked rotor accident (Ginna UFSAR Section 15.3.2)   • Uncontrolled RCCA withdrawal at power (Ginna UFSAR Section 15.4.2)   • Inadvertent opening of a pressurizer safety or relief valve (Ginna UFSAR Section 15.6.1)   • Steam generator tube rupture (Ginna UFSAR Section 15.6.3)   As each transient analyzed for Ginna using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

Table A.3-1: RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions	
2.	***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."***
	<u>Justification</u> The Ginna Station consists of a single 2-loop Westinghouse-designed unit that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.
3.	***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis.."***
	<u>Justification</u> The input data used in the RETRAN analyses performed by Westinghouse came from both Constellation Generation Group and Westinghouse sources. Assurance that the RETRAN input data is conservative for Ginna is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative plant-specific input values were requested and collected from the responsible Constellation Generation Group and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference A.3-1), the safety analysis input values used in the Ginna analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Reference

A.3-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

A.4 LOFTRAN for Non-LOCA Safety Analysis

Table A.4-1: LOFTRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. ***"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:*** • ***Feedwater System Malfunction*** • ***Excessive Increase in Steam Flow*** • ***Inadvertent Opening of a Steam Generator Relief or Safety Valve*** • ***Steamline Break*** • ***Loss of External Load*** • ***Loss of Offsite Power*** • ***Loss of Normal Feedwater*** • ***Feedwater Line Rupture*** • ***Loss of Forced Reactor Coolant Flow*** • ***Locked Pump Rotor*** • ***Rod Withdrawal at Power*** • ***Rod Drop*** • ***Startup of an Inactive Pump*** • ***Inadvertent ECCS Actuation*** • ***Inadvertent Opening of a Pressurizer Relief or Safety Valve*** ***This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."*** **Justification** For Ginna, the LOFTRAN code was used in the analyses of the dropped rod transient (Ginna UFSAR Section 15.4.6) and ATWS (Ginna UFSAR Section 15.8). As each of these transients matches one of the transients listed in the SER, additional justification is not required.

A.5 VIPRE for Non-LOCA Thermal/Hydraulics

Table A.5-1: VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions, and Conditions
1. ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."***   **Justification**   The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Ginna 14x14 422V+ fuel. The use of the WRB-1 DNB correlation is based on the notification change which introduces the 14x14 422V+ mid-grid design (NPL 97-0538, CAW-97-1166). The basic change is reverting back to the larger OD fuel rod as in standard fuel but with a new Low Pressure Drop mid-grid design. The applicability of WRB-1 to the LPD mid-grid was justified under FCEP (WCAP-12488-A).   The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Ginna 422V+ fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse two-loop plants using the same fuel design.
2. ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."***   **Justification**   The core boundary conditions for the VIPRE calculations for the 422V+ fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272-P-A (Reference A.5-1).
3. ***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."***   **Justification**   As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used in the DNB analyses of 422V+ fuel for Ginna. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45, for pressures between 500 psia and 1,000 psia).

Table A.5-1: VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions, and Conditions

4. ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***

Justification

For application to Ginna safety analysis, the usage of VIPRE in the post-critical heat flux region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient,
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation,
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further ensured with the following input:

- Fuel rod input based on the maximum fuel temperature at the given power,
- The hot spot power factor was equal to or greater than the design linear heat rate,
- Uncertainties were applied to the initial operating conditions in the limiting direction.

Reference

A.5-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

A.6 Best-Estimate Large Break LOCA

The following discussion of the applicability limits and usage conditions imposed on the ASTRUM methodology used for the Large Break LOCA analysis is fashioned after the discussion in Section 13-3 of the ASTRUM topical (WCAP-16009-P-A). Only those limits and conditions which have been determined as applicable to the ASTRUM methodology (discussed in Section 13-3 of WCAP-16009-P-A and approved by the NRC in Section 4.0 of the ASTRUM SER) are addressed below.

Table A.6-1: Best-Estimate Large Break LOCA - Applicability Limits
1. "The use of the WCOBRA/TRAC EM for long term cooling licensing analyses is not covered in this review."
The WCOBRA/TRAC EM was used for the Large Break LOCA licensing analysis, but not the long term cooling analysis. As such, this applicability limit is met.
2. "Our review did not cover the use of the WCOBRA/TRAC EM for small break LOCA licensing analyses."
The WCOBRA/TRAC EM was used for the Large Break LOCA licensing analysis, but not the small break LOCA analysis. As such, this applicability limit is met.
3. "Section 2.4.4 of this SER **[for WCAP-14449-P-A]** ***discusses that ranges and biases of parameters were based on data, including UPTF and CCTF data. Of particular concern is the ranging of interfacial drag and condensation, which is based on UPTF and CCTF data. In a letter dated April 8, 1999, to assure that the 2-loop version of the methodology would not be applied for heat generation rates higher than covered by the UPTF and CCTF data, W proposed to limit the application of the UPI methodology to nominal power levels of 1980 MWt, low power region average heat generation rate of less than 6.9 kW/ft, and maximum analyzed linear heat generation rates of 17 kW/ft. We find the proposed limits are acceptable because they are consistent with the range of the UPTF and CCTF data. We also find that the use of the methodology above these values is outside the scope of our review, and would require further justification and NRC review."***
This requirement is satisfied since the analysis considers a nominal core power level of less than 1980 MWt, low power region average heat generation rate of less than 6.9 kW/ft, and maximum analyzed linear heat generation rate of less than 17 kW/ft.

Table A.6-2: Best-Estimate Large Break LOCA - Usage Conditions
1. "A recommended justification for any future time step changes (first listed item). We require that W perform this justification as recommended, and retain traceable documentation of this action in its in-house plant records."
This requirement is satisfied since all time step changes have been justified and documented in Westinghouse records.

Table A.6-2: Best-Estimate Large Break LOCA - Usage Conditions
2. *"Based on Reference 214* [A.6-1], *Attachment 7, the analysis to determine the uncertainty distributions for accumulator and SI temperatures uses plant operating data and/or plant Technical Specifications. Therefore, this analysis must be performed for each plant."*
This requirement is satisfied since the analyzed accumulator and SI temperature ranges use plant operating data.
3. *"On CQD* [A.2] *page 7-24, Westinghouse stated the fuel pellet thermal expansion model in MATPRO-11, Revision 1, Reference 176* [A.6-3], *was simplified by omitting the corrections for molten fuel and mixed oxide (Pu). In Reference 214* [A.6-1], *List II, Item 6, Westinghouse committed to resubmitting the relevant WCOBRA/TRAC models for NRC review if the code will be used to analyze US licensed plants with molten fuel or mixed oxide."*
This requirement is satisfied since the R. E. Ginna Large Break LOCA analysis does not support the use of molten fuel or mixed oxide.
4. *"Westinghouse, in Reference 214* [A.6-1], *List II, Item 8, committed to not changing the value and range of the broken loop cold leg nozzle loss coefficient for plant specific applications. Also, the values developed apply only to LBLOCA and must be justified for other applications."*
This requirement is satisfied since the range of the broken loop cold leg nozzle loss coefficient developed for LBLOCA was not changed for the R. E. Ginna Large Break LOCA analysis.
5. *"Westinghouse, in Reference 214* [A.6-1], *Attachment 9, gave additional explanation on its use of the full Method of Characteristics model for each time step in the code implementation of choked flow. In the above reference, Westinghouse committed to include the information in the CQD."*
Westinghouse satisfied this requirement by adding the necessary text to the critical flow model description in Section 4-8-2 of WCAP-12945-P-A and the ASTRUM topical report (WCAP-16009-P-A).
6. *"Westinghouse noted that the choked flow solution is implemented in the pressure solution of the code rather than in the back substitution step after solving the pressure equation. This results in a smoother pressure and flow response in the code. In Reference 214* [A.6-1], *Attachment 9, Westinghouse committed to include this information in the CQD."*
Westinghouse satisfied this requirement by adding the necessary text to the critical flow model description in Section 4-8-2 of WCAP-12495-P-A and the ASTRUM topical report (WCAP-16009-P-A).
7. *"Westinghouse, in Reference 214* [A.6-1], *List II, Item 10, committed to use the multiplier given in Reference 214* [A.6-1], *Attachment 4, to account for rod-to-rod radiation effects in the heat transfer multiplier data base."*
Westinghouse applies a correction factor to the reflood heat transfer multipliers to account for rod-to-rod radiation effects, as described on page 25-5-26 of WCAP-12945-P-A. The same correction factor is applied with ASTRUM.

References

A.6-1 N. J. Liparulo, Westinghouse, letter to USNRC Document Control Desk, "Docketing of Supplemental Information Related to WCAP-12945-P," NSA-SAI-96-156, April 30, 1996.

A.6-2 S. M. Bajorek, et. Al., WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1. "Code Qualification Document for Best Estimate LOCA Analysis," 1998.

A.6-3 D. L. Hagrman, G. A. Reymann, and R. E. Manson, MATPRO-Version 11 (Revision 1), A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, NUREG/CR-0497, Rev. 1, 1980.

A.7 NOTRUMP for Small Break LOCA

NOTRUMP SER Restriction Compliance Summary

The following table contains a synopsis of the NRC imposed Safety Evaluation Report (SER) restrictions/requirements and the Westinghouse compliance status related to these issues. Not all the items identified are clearly SER restrictions, but sometimes state the NRC's interpretation of the Westinghouse Evaluation Methodology utilized for a particular aspect of the Small Break Loss Of Coolant (LOCA) Evaluation Model.

WCAP-10054-P-A and WCAP-14710-P-A (References A.7-1 and A.7-2)
WCAP-10054-P-A is titled "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," and is dated August, 1985. The following summarizes the SER restrictions and requirements associated with this WCAP:
1. SER Wording (Page 6)
"The use of a single momentum equation implies that the inertias of the separate phases can not be treated. The model therefore would not be appropriate for situations when separate inertial effects are significant. For the small break transients, these effects are not significant."
SER Compliance
Inherent compliance due to the use of a single momentum equation.
2. SER Wording (Page 8)
"To assure the validity of this application, the bubble diameter should be on the order of 10^{-1}-2 cm. As long as steam generator tube uncovery (concurrent with a severe depressurization rate) does not occur, this option is acceptable."
SER Compliance
Westinghouse complies with this restriction for all Appendix-K licensing basis calculations. Typical Appendix-K calculations do not undergo a significant secondary side system depressurization in conjunction with steam generator tube uncovery due to the modeling methodology utilized.
3. SER Wording (Page 14)
"The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10CFR50 Appendix K for LOCA analysis at pressure above 250 psia"
SER Compliance
The original NOTRUMP model was limited to no less than 250 psia since the model, as contained in the NOTRUMP code, did not contain information below this range. Westinghouse extended the model to below 250 psia, as allowed by Appendix K paragraph I-C-2, and reported these modifications to the NRC via the 1995 annual reporting period (NSD-NRC-96-4639).
4. SER Wording (Page 16)
"Westinghouse, however, has stated that the separator models are not used in their SBLOCA analyses."
SER Compliance
Westinghouse does not model the separators in the secondary side of the steam generators for Appendix-K Small Break LOCA analyses; therefore, compliance exists.
5. SER Wording (Pages 16-17)
"Axial heat conduction is not modeled." and "Deletion of clad axial heat conduction maximizes the peak clad temperature."
SER Compliance
The Westinghouse Small Break LOCA is comprised of two computer codes, the NOTRUMP code which performs the detailed system wide thermal hydraulic calculations and the LOCTA code which performs the detailed fuel rod heatup calculations. The NOTRUMP code does not model axial conduction in the fuel rod and therefore complies. The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP-14710-P-A which supplements the original NOTRUMP documentation.
6. SER Wording (Page 17)

"...; critical heat flux, W-2, W-3, or Macbeth, or GE transient CHF (the W-2 and W-3 correlations are used for licensing evaluations);..."
SER Compliance
The information presented here indicates that the NRC apparently misstated that Westinghouse was utilizing the W-2,W-3 correlations for Critical Heat Flux (CHF) in the fuel rod heat transfer model. A review of the analyses performed by Westinghouse, including those in WCAP-11145-P-A, indicates that the Macbeth CHF correlation has been utilized for all Appendix-K analyses performed by Westinghouse. This is consistent with the slab heat transfer map as described in WCAP-10054-P-A. In addition, the Macbeth correlation is specifically called out in Appendix K I-C-4-4 as an acceptable CHF model. In a supplemental response to NRC questions (Specifically question 440.1 found in Appendix-A of WCAP-10054-P-A, Page A-10), a description of the core model describes the Macbeth as being utilized as the CHF correlation in the NOTRUMP Small Break LOCA model.
7. SER Wording (Page 21)
"The standard continuous contact model is not appropriate for vertical flow,..."
SER Compliance
The standard continuous contact flow links are not utilized when modeling vertical flow in the Appendix-K NOTRUMP Evaluation Model analyses; therefore, compliance is demonstrated.
8. SER Wording (Page 27)
"..., the hardwired choice of one fuel pin time step per coolant time step should result in sufficient accuracy."
SER Compliance
The NOTRUMP code continues to utilize only one fuel pin time step per coolant time step and therefore complies with this requirement.
9. SER Wording (Page 47)
"The code options available to the user but not applied in licensing evaluations were not reviewed."
SER Compliance
Westinghouse complies with this requirement.
10. SER Wording (Page 53)
"4. Steam Interaction with ECCS Water, a. Zero Steam Flow in the Intact Loops While Accumulators Discharge Water."
SER Compliance
Per paragraph I-D-4 Appendix-K, the following is stated: "During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption." As can be seen, the specific Appendix-K wording can be considered applicable to Large Break LOCAs only since Small Break LOCAs do not undergo a true refill/reflood period. However, the Westinghouse Small Break LOCA Evaluation Model methodology is such that for break sizes in which the intact loop seal restriction is not removed (WCAP-11145-P-A Page 2-11), steam flow through the intact loop(s) is automatically (artificially) restricted via the loop seal model. While not specifically limited to zero, the flow is drastically reduced via the application of the artificial loop seal restriction model. For breaks sizes above which the loop seal restriction is removed (typically >= 6 inch diameter breaks), this criterion is not explicitly adhered to. The implementation of the COSI condensation model into NOTRUMP (As approved by the NRC in WCAP-10054-P-A, Addendum 2, Revision 1), which is based on additional experimental documentation and improved modeling techniques, more accurately models the interaction of steam with Emergency Core Cooling Water in the cold leg region. This experimental documentation supports the more accurate modeling of steam/water interaction in the cold leg region as allowed by Appendix-K. Note

however that even with the COSI condensation model active, the accumulator injection condensation model still utilizes the conservative model as originally licensed in the NOTRUMP code.
11. SER Wording (Page 7 of enclosure 2)
"Per generic letter 83-35, compliance with Action Item II.K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs."
SER Compliance
Westinghouse submitted WCAP-11145-P-A in support of generic letter 83-35 Action Item II. K.3.31. As part of this effort, verification was provided which documented that the cold leg break location remains limiting.
WCAP-10054-P-A, Addendum 2, Revision 1 (Reference A.7-3)
WCAP- 10054-P-A, Addendum 2, Revision 1 is titled "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," and is dated July 1997. The following summarizes the SER restrictions and requirements associated with this WCAP:
1. SER Wording (Page 3)
"It is stated in Ref. 5 that the range of injection jet velocities used in the experiments brackets the corresponding rates in small break LOCAs for Westinghouse plants and that the model will be used within the experimental range. Also in References 1 and 5 Westinghouse submitted analyses demonstrating that the condensation efficiency is virtually independent of RCS pressure and state that the COSI model will be applied within the pressure range of 550 to 1200 psia."
SER Compliance
The coding implementation of the COSI model correlation in the NOTRUMP model restricts the application of the COSI condensation model to a default pressure range of 550 to 1200 psia and limits the injection flow rate to a default value of 40 Ibm/sec-loop. The value of 40 lbm/sec-loop corresponds to the 30 ft./sec velocity utilized in the COSI experiments. As such, the default NOTRUMP implementation of the COSI condensation model complies with the applicable SER restrictions.
WCAP-11145-P-A (Reference A.7-4)
WCAP-11145-P-A, is titled "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code," and is dated 1986. No specific SER restrictions were provided by the NRC as part of this WCAP review; however, the SER contains verification that the requirements of Item II.K.3.31 have been satisfied (i.e. break location study).
1. SER Wording (Page 5)
"We therefore, find that the requirements of NUREG-0737, Item II.K3.31, as clarified by Generic Letter 83-35, have been satisfied.
SER Compliance
We find that a condition of the safety evaluation for NOTRUMP as applied to Item II.K.3.30 has been satisfied. The limiting cold leg break size for a 4-loop plant was reanalyzed at pump suction and at hot leg locations. The results confirmed that the cold leg break was limiting."
WCAP-14710-P-A (Reference A.7-5)
WCAP-14710-P-A, is titled "1-D Heat Conduction Model for Annular Fuel Pellets," and is dated May 1998. No specific SER restrictions are provided by the NRC in this document; however, a conclusion was reached regarding the modeling of annular pellets during Small Break LOCA event.
1. SER Wording
"Based on its conclusions that the explicit modeling of annular pellets, as described in WCAP 14710(P), provides a more realistic representation in W Appendix K ECCS evaluation models of the annular pellets, while retaining conservatism in those evaluation models, the staff finds that the explicit modeling of annular pellets, as described in WCAP-14710(P), in W Appendix K LOCA evaluation models permits those models to continue to satisfy the regulations to which they were approved, and is, therefore, acceptable for incorporation into those models."

SER Compliance
Westinghouse performs sensitivity studies to assess the impact of modeling annular pellets on plant specific analyses.

References

A.7-1 WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," N. Lee, et al., August 1985.

A.7-2 WCAP-10079-P-A, (Proprietary), and WCAP-10080-A, (Non-Proprietary), NOTRUMP - A Nodal Transient Small Break And General Network Code, Meyer, P. E., August 1985.

A.7-3 WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," C. M. Thompson, et al., July 1997.

A.7-4 WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," S. D. Rupprecht, et al., 1986.

A.7-5 WCAP-14710-P-A, "1-D Heat Conduction Model for Annular Fuel Pellets," D. J. Shimeck, May 1998.

A.8 MULTIFLEX for LOCA Hydraulic Forces

The NRC Safety Evaluation Report (SER) for the MULTIFLEX 1.0 Evaluation Model can be found in the front of WCAP-8708 Rev. 2 (Reference A.8-1). This SER stipulates a number of conditions and limitations on the use of the MULTIFLEX 1.0 Evaluation Model for licensing basis calculations. The following is a review of these SER restrictions and requirements.

	Table A.8-1: MULTIFLEX 1.0
Limitations, Restrictions, and Conditions	
1.	**SER Restriction** - Use of Corrected Sonic Velocity (SER, page 11) ***SER Wording - "The sonic velocity, or wave speed, computed with the empirical equation of state was not consistent with the 1967 ASME Steam Tables. The corrected sonic velocity data is required for a licensing calculation."***
	SER Compliance – The MULTIFLEX code has been changed (prior to the issuance of Revision 1 to WCAP-8708) to compute revised sonic velocity. Therefore, Westinghouse is in compliance with this restriction.
2.	**SER Restriction** - Lower Plenum Modeling (SER, page 12) ***SER Wording - "In the modeling region from the downcomer annulus to the lower plenum, the equivalent pipe network provided an artificially short transport distance across the length of the lower plenum. The correct radial transport distance, the diameter of the pressure vessel, is required in the model for a licensing calculation."***
	SER Compliance - Westinghouse does not use the "artificially short" lower plenum length cited in the SER. Therefore, it can be concluded that Westinghouse is in compliance with this modeling requirement.
3.	**SER Restriction** - 10 Mass Point Downcomer (SER, page 12, 18, 19) ***SER Wording - "The peak lateral force for a calculation using a 10 mass point representation for the core support barrel shows an increase in loading of 4% over the reference 5 mass point case. The NRC, therefore, requires a 10 mass point model be used for a coupled licensing calculation."***
	SER Compliance - Standard methodology uses a 10 mass point structural model. Therefore, Westinghouse is in compliance with this requirement.
4.	**SER Restriction** - 1 Millisecond Break Opening Time (BOT) (SER, page 13) ***SER Wording - "The use of a one millisecond opening time, as specified by Westinghouse, is required for a licensing calculation. Longer break opening times will not be considered unless Westinghouse demonstrated that the proposed break opening time with current equivalent pipe network adequately predicts the results of applicable experimental data."***
	SER Compliance - Standard methodology uses a 1 millisecond BOT. Therefore, Westinghouse is in compliance with this restriction.
5.	SER Restriction - Use of "Question 18" Input Parameters (SER, page 12). Question 18 establishes a line-by-line review of MULTIFLEX input. Parameters, identifying those that are "Required for design basis blowdown analysis" ***SER Wording - "The response to Question 18 of reference 4 is to be included in the MULTIFLEX report to identify the acceptable input option for a licensing calculation."***
	SER Compliance - The inputs used in the response to Question 18 were reviewed against the MULTIFLEX inputs established as Westinghouse's current methodology. We can state that our current models conservatively bound the requirements for licensing basis calculations as described in the MULTIFLEX SER. Therefore, Westinghouse is in compliance with this restriction.

MULTIFLEX 3.0 Applications

As indicated in the SER of WCAP-15029-P-A (Reference A.8-3), WCAP-9735, Rev. 2 (Reference A.8-2) topical was submitted for NRC review, and again subsequently withdrawn. It was determined that "Evaluation of the MULTIFLEX 3.0 methodology is not a requisite for concluding that WCAP-15029 is acceptable". The Staffs discussion of MULTIFLEX 3.0 is shown below:

"The MULTIFLEX 3.0 program is described as a more sophisticated analysis tool for LOCA hydraulic force calculations than the currently approved version, MULTIFLEX 1.0. WCAP-15029 indicates that the MULTIFLEX 3.0 program enhancements of MULTIFLEX 1.0 include: the use of a two dimensional flow network to represent the vessel downcomer region in lieu of a collection of one dimensional parallel pipes; the allowance for non-linear boundary conditions at the vessel and downcomer interface at the radial keys and the upper core barrel flange in lieu of simplified linear boundary conditions; and the allowance for vessel motion in lieu of rigid vessel assumptions. WCAP-15029 indicates that these modifications are included in the MULTIFLEX 3.0 program that is used to estimate the LOCA hydraulic forces on the vessel and consequential forces induced on the fuel and reactor vessel internal structures. The staff concurs with the WOG that MULTIFLEX 3.0 provides a more accurate and realistic modeling approach. On this basis, and considering that MULTIFLEX 3.0 is based on the previously approved MULTIFLEX 1.0, the staff considers the application of MULTIFLEX 3.0 with the WCAP-15029 methodology reasonable and acceptable.".

Only one of the four SER restrictions in WCAP-15029-P-A (Reference A.8-3) applies to analyses performed using MULTIFLEX 3.0. Limitation number 2 reads: "The noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology."

The current nodalization employed in the Westinghouse baffle-former bolting analyses has been validated through a series of calculations. Westinghouse has verified that the current MULTIFLEX code version produces equivalent results to those used in the original development of MULTIFLEX 3.0 modeling features, despite several changes in operating system and computer platform. Westinghouse has demonstrated that the current standard nodalization produces equivalent results to those used in original test cases. Westinghouse has performed a series of sensitivity studies on MULTIFLEX 3.0 models using the current nodalization. Also, the historical model validation cases were found to yield conservative results relative to test data. This collection of documentation supports the conclusion that analyses performed to the current nodalization meet the limitation in WCAP-15029-P-A (Reference A.8-3).

MULTIFLEX 3.0 has also been accepted for use in other applications which are limited by the same acceptance criteria, i.e. fuel qualification. The Control Rod Insertion program, documented in WCAP-15245 (Reference A.8-4), was performed using MULTIFLEX 3.0 and the analyses were reviewed and accepted by the Staff (Reference A.8-5). These analyses have been used as a template for additional applications limited by the same acceptance criteria.

The use of break opening times greater than 1 millisecond has also been approved by the US-NRC (Reference A.8-6) for baffle barrel-bolting analyses. However, the use of longer break opening times is not approved for use on a generic basis. Such applications will require additional justification.

References

A.8-1 WCAP-8708-P-A, (Proprietary) and WCAP-8709-A, (Nonproprietary), MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, K. Takeuchi, et al., September 1977.

A.8-2 WCAP-9735, Rev. 2, (Proprietary), and WCAP-9736, Rev. 1, (Nonproprietary), MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model, K. Takeuchi, et al., February 1998.

A.8-3 WCAP-15029-P-A, WCAP-15030-NP-A "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions", December 1998.

A.8-4 WCAP-15245 (proprietary), WCAP-15246 (non-proprietary), "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units I and 2", May 28, 1999.

A.8-5 Letter from John F. Stang (US-NRC) to Robert P. Powers (Indiana Michigan Power Company), "Issuance of Amendments - Donald C. Cook Nuclear Plant, Units I and 2 (TAC Nos. MA6473 and MA6474)", December 23, 1999.

A.8-6 WCAP-14748-P-A, revision 0, WCAP-14749-NP-A, revision 0, "Justification for Increasing Postulated Break Opening Times in Westinghouse Pressurized Water Reactors", December 1998.

Appendix B
Additional Codes and Methods

Numerous analytical codes and methods were used to support the proposed Ginna Power Uprate. These have been reviewed against the codes and methods currently described in the UFSAR. The codes listed below, which do not currently appear in the UFSAR, are identified for the NRC's information, along with their functional application.. All of these codes/methods have previously been submitted to the NRC under other licensees' dockets, and have been determined by CEG/Ginna to be appropriate for use in their respective applications.

CODE	APPLICATION
CIRC	SG Inventory
RETRAN	Thermal-Hydraulic Analysis
CALOPR	Thimble Bypass Flow
APOLLO	RWSC, RWAP
ANCSUM	Core Design
ASTRUM- WCOBRA TRAC	BELOCA Analysis
DROP	Control Rod Drop Times
TEMFOR	Baffle/Barrel Bolts
MULTIFLEX 3.0	Baffle/Barrel Bolts
THRIVE	Vantage V422+ Fuel
FIPCO	Radiation Source Terms
DORT	Reactor Vessel Fluence
TRICAL	Tritium Sources
FLOMAP	Hydraulic Flows
BORDER	Fuels Analysis
VIPRE	Non-LOCA T/H
STEHAM-PC	Feed Reg Valve Forcing Functions
WATHAM-PC	Turbine Stop Valve Forcing Functions
PSAP-PC	Hydraulic Flow
PC-PREPS	Pipe Support Evaluations
PILUG-PC	Stress Intensity
PERC2	Gamma and Beta Radiation
SW-QADCGGP	Radiation Shielding

APPENDIX C
MATRIX 1

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Materials and Chemical Engineering

Areas of Review	Other Guidance
Reactor Vessel Material Surveillance Program LR section 2.1.1	UFSAR Section 3.1 3.1.1 3.1.2 3.1.2.2.5 3.1.2.3.4 3.1.2.4.2 5.1.3.9 5.3.1.3 5.3.3 5.3.3.2
Pressure-Temperature Limits and Upper-Shelf Energy LR section 2.1.2	UFSAR Section 3.1 3.1.1 3.1.2 3.1.2.2.5 3.1.2.4.2 3.4.1 3.4.3 5.1.3.9 5.3.2
Pressurized Thermal Shock LR section 2.1.3	UFSAR Section 3.1 3.1.1 3.1.2 3.1.2.2.5 3.1.2.4.2 4.2.2 5.3.3 5.3.3.5
Reactor Internal and Core Support Materials LR section 2.1.4	UFSAR Section 3.1 3.1.1 3.1.1.2.1 3.1.2 3.1.2.1.1 3.2.1 3.9.5 4.2.1 17.1 17.2

Reactor Coolant Pressure Boundary Materials LR section 2.1.5	UFSAR Section 2.3.1.1 3.1 3.1.1 3.1.1.1.1 3.1.2 3.1.2.1.1 3.1.2.1.4 3.1.2.2.5 3.1.2.4.2 3.2.1 3.2.2.1.2 3.5 3.6 3.7.1.1.1 3.11 5.1 5.2 5.2.2.2 5.2.3 5.3.2.2 5.3.2.3 5.3.3 5.4.5.3.2 7.6.1 17.1 17.2 Table 5.2-1 Table 5.2-2
Leak-Before-Break LR section 2.1.6	UFSAR Section 3.6.1.3.2.3 3.6.1.3.2.13 3.6.1.3.2.14 4.7.7 5.4.11.1.2
Protective Coating Systems (Paints) - Organic Materials LR section 2.1.7	UFSAR Section 1.8.2.32 1.8.2.35 1.54 2.5 3.1 3.1.1 3.1.2 3.8.1.4.7.6 3.8.1.6.5.5 6.1.2.4.2 6.1.2.6 6.1.2.9 6.2.1.1.2 6.3.2.1.1 17.2 Table 6.1-2
Flow-Accelerated Corrosion LR section 2.1.8	UFSAR Section 3.1 3.1.1 3.1.2 10.7.5.1 10.7.8

Steam Generator Tube Inservice Inspection <u>LR section 2.1.9</u>	UFSAR Section 3.1 3.1.1 3.1.2 5.4.2.2
Steam Generator Blowdown System LR section 2.1.10	UFSAR Section 3.1 3.1.1 3.1.2 3.1.2.2.5 3.6.1 3.6.2 3.7.3.7 6.2.4 9.3.2 10.7.5 10.7.7 10.7.8 11.5
Chemical and Volume Control System <u>LR section 2.1.11</u>	UFSAR Section 3.1 3.1.1 3.1.2 3.1.2.2.5 3.1.2.3.10 3.7 6.2.4.4 9.3.4 9.3.4.1.1 9.3.4.2

APPENDIX C
MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

Areas of Review	Other Guidance
Pipe Rupture Locations and Associated Dynamic Effects LR section 2.2.1	UFSAR Section 3.6 3.6.1.3.2.13 3.6.1.3.2.14 5.4.11.1.2

LR section 2.2.2 Pressure-Retaining Components and Component Supports LR section 2.2.2.1 NSSS Piping, Components and Supports LR section 2.2.2 2 BOP Piping Components and Supports LR section 2.2.2.3 Reactor Vessel and Supports LR section 2.2.2.4 Control Rod Drive Mechanism and Supports LR section 2.2.2.5 Steam Generators and Supports LR section 2.2.2.6 Reactor Coolant Pumps and Supports LR section 2.2.2.7 Pressurizer and Supports	UFSAR Section 3.1 Section 3.1.1 Section 3.1.1.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.1.2.2.5 Section 3.1.2.2.6 Section 3.2 Section 3.2.1 Section 3.5 Section 3.6 Section 3.7.1.1.1 Section 3.9 Section 3.11 Section 17.1 Section 17.2 Section 3.5 Section 3.6 Section 3.9.1.2 Section 3.2 Section 3.7 Section 3.7.3.1 Section 3.7.3.7.4 Section 3.9 Section 3.9.2.1 Section 3.9.2.1.8 Section 3.9.2.2.1 Section 5.3 Section 3.1.1.4.1 Section 3.7.3.1.1.3 Section 3.9.2.2.4.11 Section 3.9.4 Section 3.7.3.1.1.2 Section 3.9 Section 3.9.2.2.4.8 Section 5.4.2 Section 3.9.2.2.4.9 Section 3.9.3 Section 3.9.3.2.3 Section 5.4.1 Section 5.1.4 Table 5.2-1 Table 5.3-1 Table 5.1-4

Reactor Pressure Vessel Internals and Core Supports LR section 2.2.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.1.2.5 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.6 Section 3.9.2.3.6 Section 3.9.2.5.1 Section 3.9.5 Section 3.11 Section 4.2.1.3.4.1 Table 5.1-4
Safety-Related Valves and Pumps LR section 2.2.4	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.4.8 Section 3.1.2.4.11 Section 3.1.2.4.14 Section 3.1.2.4.16 Section 3.1.2.5.5 Section 3.9.6 Section 5.4.9.3 Table 3.2-1 Table 3.11-1
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment LR section 2.2.5	UFSAR Section 2.1.1 Section 2.5 Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1 Section 3.1.2.1.4 Section 3.1.2.2 Section 3.1.2.14 Section 3.1.2.30 Section 3.2 Section 3.3 Section 3.4 Section 3.5 Section 3.5.1 Section 3.6 Section 3.7 Section 3.9 Section 3.9.2 Section 3.10 Section 3.11 Section 17.1 Section 17.2 Table 3.2-1 Table 5.2-1

NSSS Design Transients LR section 2.2.6	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.1.2 Section 3.1.2.2.5 Section 3.1.2.2.6 Section 3.2 Section 3.7 Section 3.9 Section 5.1 Section 17.1 Section 17.2 Table 5.2-1
Bottom-Mounted Instrumentation Guide Tubes and Flux Thimbles LR section 2.2.7	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.1.2.2.5 Section 3.9.5.1.3 Section 5.2 Section 7.7.4.2.1 Section 7.7.4.2.3 Section 7.7.4.2.4

APPENDIX C
MATRIX 3

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Electrical Engineering

Areas of Review	Other Guidance
Environmental Qualification of Electrical Equipment LR section 2.3.1	UFSAR Section 3.1.1.4.5 Section 3.10 Section 3.11 Section 3.11.1.2 Section 3.11.2 Section 3.11.3 Section 3.11.3.1.2 Section 3.11.3.2.3 Section 3.11.4 Section 3.11.5 Section 6.1.2.1 Section 12.4.3.3 Figure 6.1-1 Figure 6.1-2 Table 3.11-1 Table 3.11-2 Table 3.11-3 Table 3.11-6 Table 3.11-I
Offsite Power System LR section 2.3.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.8 Section 8.2 Section 8.2.1.1.1 Section 8.2.1.1.3 Section 8.2.1.2 Section 8.2.2.1 Section 8.2.2.2.1
AC Onsite Power System LR section 2.3.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.8 Section 8.1 Section 8.1.1 Section 8.1.3 Section 8.3 Section 8.3.1.1.2 Section 8.3.1.1.3 Section 8.3.1.1.4 Section 8.3.1.1.5 Section 8.3.1.1.6 Section 10.2.2

DC Onsite Power System LR section 2.3.4	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.8 Section 8.3.2
Station Blackout LR section 2.3.5	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 8.1.4.4 Section 8.1.4.5 Section 8.1.4.5.1 Section 8.1.4.5.2 Section 8.3.2.2

APPENDIX C
MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

Areas of Review	Other Guidance
Reactor Protection, Safety Features Actuation, and Control Systems <u>LR section 2.4.1</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.1.2.2.4 Section 3.1.2.2.10 Section 3.1.2.3.1 Section 3.1.2.3.2 Section 3.1.2.3.3 Section 3.1.2.3.4 Section 3.1.2.3.5 Section 3.1.2.3.6 Section 3.1.2.3.10 Section 3.6 Section 3.11 Section 7.1.2 Section 7.2 Section 7.2.1 Section 7.2.6 Section 7.3 Section 7.3.2.3.1 Section 7.4 Section 7.5 Section 7.6 Section 7.7 Section 7.7.1 Section 7.7.1.1.4 Section 7.7.1.2.6 Section 7.7.1.5 Section 7.7.4 Section 7.7.6 Table 7.5-1
Plant Operability <u>LR section 2.4.2</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 7.1.2 Section 7.2 Section 7.3 Section 7.5 Section 7.7 Section 7.7.1.1.4

Pressurizer Component Sizing LR section 2.4.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.1 Section 3.1.2.2.4 Section 3.1.2.2.10 Section 3.1.2.3.5 Section 3.2.1 Section 5.1.3.1 Section 5.3.2.5 Section 5.4.7.1 Section 5.4.10.1 Section 7.2.5 Section 7.7 Section 7.7.1.1.5 Section 7.7.1.3.1

APPENDIX C
MATRIX 5

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Plant Systems

Areas of Review	Other Guidance
Flood Protection LR section 2.5.1.1.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.4 Section 3.4.2 Section 3.6.2.4.8.2 Section 3.11.3.2.4 Section 3.11.3.6 Section 3.11.3.8 Section 9.3.3 Section 9.5.1.2.4.5 Section 10.6.2.9 Section 11.2.2
Equipment and Floor Drains LR section 2.5.1.1.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 9.3.3 Section 9.5.1.2.4.5 Section 11.2.2
Circulating Water System LR section 2.5.1.1.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.6.2.4.8.2 Section 10.6.2.9
Internally Generated Missiles LR section 2.5.1.2.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.5.1
Turbine Generator LR section 2.5.1.2.2	UFSAR Section 3.1.2.1.4 Section 3.5.1.2 Section 10.2

Pipe Failures <u>LR section 2.5.1.3</u>	UFSAR Section 2.3.4.1 Section 2.3.4.1.2 Section 2.3.4.2 Section 2.3.4.2.1 Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.6 Section 3.6.2 Section 3.6.2.5.1.2 Section 3.6.2.5.1.4 Section 3.11 Section 3.11.3.5 Table 3.11-1
Fire Protection <u>LR section 2.5.1.4</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.1.1.3 Section 3.1.2 Section 3.1.2.1.3 Section 3.1.2.1.5 Section 9.5.1 Section 9.5.1.1.1 Section 9.5.1.1.2 Section 9.5.1.4
Pressurizer Relief Tank <u>LR section 2.5.2</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.5 Section 3.6 Section 3.9.2.1.4 Section 3.11 Section 5.4.8.1 Section 5.4.8.2 Section 5.4.10 Figure 5.4-9 Table 5.4-8
Main Condenser Evacuation System <u>LR section 2.5.3.2</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 3.1.2.6.5 Section 10.7.6.2 Section 11.5 Section 12.5
Turbine Gland Sealing System <u>LR section 2.5.3.3</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 3.1.2.6.5 Section 10.3.2 Section 10.4.2 Section 10.7.6.1 Section 10.7.8 Section 11.5

Spent Fuel Pool Cooling and Cleanup System LR section 2.5.4.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.5 Section 3.1.2.4.15 Section 3.1.2.6.2 Section 3.1.2.15 Section 3.5.1.3.2 Section 9.1.2.2 Section 9.1.3 Section 9.1.3.1 Section 9.1.3.4.1.2 Section 9.1.3.4.1.8 Section 9.2.1 Section 9.4.4
Station Service Water System LR section 2.5.4.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.15 Section 3.6 Section 3.7.3.7.3.10 Section 3.11 Section 6.2.4 Section 6.5.1.3 Section 9.2.1 Section 9.2.1.2.6 Section 9.2.1.4.1 Section 9.2.2 Section 11.5.2.2.9 Section 11.5.2.2.13
Reactor Auxiliary Cooling Water Systems LR section 2.5.4.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.15 Section 3.2.2.1.3 Section 3.2.2.5 Section 3.5.1.3.1.6 Section 3.6 Section 3.7.3.7.3.11 Section 3.11 Section 5.2.5.2 Section 6.2.4 Section 6.5.1.3 Section 9.2.1 Section 9.2.1.2.6 Section 9.2.2

Ultimate Heat Sink <u>LR section 2.5.4.4</u>	UFSAR Section 1.8.1.27 Section 2.4.1 Section 2.4.6 Section 2.4.8 Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.5 Section 3.1.2.4.15 Section 8.1.4.4 Section 9.2.1 Section 9.2.2 Section 9.5.1
Auxiliary Feedwater System <u>LR section 2.5.4.5</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.2.10 Section 3.1.2.4.5 Section 3.1.2.4.15 Section 3.5 Section 3.6 Section 3.7.3.7 Section 3.11 Section 5.4.5.3 Section 6.2.1.2.3 Section 6.2.4 Section 8.1.4.4 Section 8.1.4.5 Section 9.5.1.4 Section 10.5 Figure 6.2-26 Figure 6.2-78 Table 6.2-15
Main Steam <u>LR section 2.5.5.1</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.5 Section 3.5 Section 3.6 Section 3.9.2.1.5 Section 3.9.2.1.8 Section 3.11 Section 5.4.5 Section 10.3
Main Condenser <u>LR section 2.5.5.2</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 10.4.3 Section 10.6 Section 10.7.1 Section 11.5.2.2.8

Turbine Bypass LR section 2.5.5.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.4.5 Section 3.5 Section 3.6 Section 3.9.2.1.5 Section 3.9.2.1.8 Section 3.11 Section 5.4.5 Section 10.7.1
Condensate and Feedwater LR section 2.5.5.4	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.15 Section 3.5 Section 3.6 Section 3.7.3.7 Section 3.9.2.1.6 Section 3.9.2.1.8 Section 3.11 Section 6.2.1.2.3 Section 6.2.4 Section 10.4.4 Section 10.4.5
Gaseous Waste Management Systems LR section 2.5.6.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.3 Section 3.1.2.6.1 Section 3.1.2.6.2 Section 9.5.1 Section 11.1.1.1 Section 11.1.1.2 Section 11.3
Liquid Waste Management Systems LR section 2.5.6.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 3.1.2.6.2 Section 9.3.3 Section 11.1.1.1 Section 11.1.1.2 Section 11.2
Solid Waste Management Systems LR section 2.5.6.3	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 3.1.2.6.4 Section 3.1.2.6.5 Section 11.1.1.1 Section 11.1.1.2 Section 11.4 Section 11.5 Section 12.5

Emergency Diesel Engine Fuel Oil Storage and Transfer System <u>LR section 2.5.7.1</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.2.8 Section 3.5 Section 3.6 Section 3.11 Section 8.1.1 Section 8.3.1.1.6 Section 9.5.4
Light Load Handling System (Related to Refueling) <u>LR section 2.5.7.2</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.2 Section 3.1.2.6.3 Section 9.1.2.1.7 Section 9.1.2.1.13 Section 9.1.4 Section 9.1.5
Circulating Water System <u>LR section 2.5.8.1</u>	UFSAR Section 1.2.6 Section 3.4 Section 10.6 Section 11.2.1

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Containment Review Considerations

Areas of Review	Other Guidance
Primary Containment Functional Design LR section 2.6.1	UFSAR Section 3.11.3.1.1 Section 6.2 Section 6.2.1.2 Section 6.2.1.2.2.6 Section 7.2
Subcompartment Analyses LR section 2.6.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.5.1 Section 3.6.1.3.2 Section 5.4.11.1 Section 6.2.1.3 Section 6.2.1.3.2 Section 6.2.1.3.4
Mass and Energy Release Analysis for Postulated Loss of Coolant LR section 2.6.3.1	UFSAR Section 6.2.1.2.2 Section 6.2.1.2.2.6 Section 6.2.1.3 Section 6.2.1.3.2 Section 6.2.1.3.4
Mass and Energy Release Analysis for Secondary System Pipe Ruptures LR section 2.6.3.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.5.1 Section 6.2.1.2.1.1 Section 6.2.1.2.3
Containment Heat Removal LR section 2.6.5	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.4.10 Section 3.1.2.4.11 Section 3.1.4.2.9 Section 3.11.3.1.1 Section 6.2.1.2.2.6 Section 6.2.1.2.3 Section 6.2.1.2.3.3 Section 6.2.2 Figure 6.1.1 Figure 6.1-1 Table 6.1-1
Pressure Analysis for ECCS Performance Capability LR section 2.6.6	UFSAR Section 15.6.4.2.4.2

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Habitability, Filtration, and Ventilation

Areas of Review	Other Guidance
Control Room Habitability System LR section 2.7.1	UFSAR Section 3.1.2.1.4 Section 3.1.2.2.10 Section 3.5 Section 3.6 Section 3.11
Spent Fuel Pool Area Ventilation System LR section 2.7.4	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 3.1.2.6.2 Section 9.4.2 Section 9.4.4
Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems LR section 2.7.5	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.6.1 Section 9.4 Section 9.4.5 Section 9.4.6 Section 9.4.7
Engineered Safety Feature Ventilation System LR section 2.7.6	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.2.8 Section 3.1.2.6.1 Section 3.5 Section 3.6 Section 3.11 Section 9.4 Section 9.4.2 Section 9.4.9
Other Ventilation Systems (Containment) LR section 2.7.7	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.2.8 Section 3.1.2.6.1 Section 3.1.2.6.2 Section 3.5 Section 3.6 Section 3.11 Section 6.2.2 Section 6.2.4 Section 9.4.1

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Fuel System Design LR section 2.8.1	UFSAR Section 4.2
Nuclear Design LR section 2.8.2	UFSAR Section 4.3
Thermal and Hydraulic Design LR section 2.8.3	UFSAR Section 4.4
Functional Design of Control Rod Drive System LR section 2.8.4.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.3.4 Section 3.1.2.3.6 Section 3.1.2.3.7 Section 3.1.2.3.8 Section 3.1.2.3.9 Section 3.1.2.3.10 Section 3.5 Section 3.6 Section 3.9.4 Section 4.2 Section 7.2 Section 7.7.1.2 Section 7.7.1.2.6 Section 9.4.1.2.3 Section 15.4
Overpressure Protection During Power Operation LR section 2.8.4.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.6 Section 3.1.2.4.2 Section 5.1.3.5 Section 5.2.2.1 Section 5.3.1.2

Overpressure Protection During Low Temperature Operation <u>LR section 2.8.4.3</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.6 Section 3.1.2.4.2 Section 3.2.2.1.1 Section 5.1.3.5 Section 5.2.2.2 Section 5.2.2.2.1 Section 5.2.2.2.2 Section 5.2.2.2.3 Section 5.2.2.2.3.2 Section 5.2.2.2.3.3 Section 5.2.2.2.3.4 Section 5.4.5.3.2.2 Section 5.4.10 Section 7.6.1
Residual Heat Removal System <u>LR section 2.8.4.4</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.5 Section 3.5 Section 3.6 Section 3.11 Section 5.4.5 Section 5.4.5.3
Non LOCA Analyses Introduction <u>LR section 2.8.5.0</u>	UFSAR Section 15.1.5 Section 15.1.5 Section 15.1.6 Section 15.2.2 Section 15.2.5 Section 15.2.6 Section 15.2.7 Section 15.3.1 Section 15.3.2 Section 15.4.1 Section 15.4.2 Section 15.4.5 Section 15.4.6 Section 15.6.1 Section 15.6.3 Section 15.8
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve <u>LR section 2.8.5.1.1</u>	UFSAR Section 2.8.5.1.2.2.1 Section 3.1 Section 15.1.4 Section 15.1.5
Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam~Pressure Regulatory Failure <u>LR section 2.8.5.2.1</u>	UFSAR Section 3.1.1 Section 3.1.2 Section 3.1.2.2.1 Section 3.1.2.2.6 Section 3.1.2.3.7 Section 9.3.4 Section 15.2.2

Loss of Nonemergency AC Power to the Station Auxiliaries LR section 2.8.5.2.2	UFSAR Section 3.1.1 Section 3.1.2 Section 3.1.2.2.1 Section 3.1.2.2.6 Section 3.1.2.3.7 Section 9.3.4 Section 15.2.6.3.1
Loss of Normal Feedwater Flow LR section 2.8.5.2.3	UFSAR Section 3.1.1 Section 3.1.2 Section 3.1.2.2.1 Section 3.1.2.2.6 Section 3.1.2.3.7 Section 4.2.1 Section 9.3.4 Section 15.2.6
Feedwater System Pipe Breaks Inside and Outside Containment LR section 2.8.5.2.4	UFSAR Section 3.1.1 Section 3.1.2 Section 3.1.2.3.8 Section 3.1.2.4.2 Section 3.1.2.4.6 Section 3.2 Section 4.2.1 Section 5.2 Section 5.3.1.2 Section 5.3.3 Section 6.3 Section 15.2.7
Loss of Forced Reactor Coolant Flow LR section 2.8.5.3.1	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.1 Section 3.1.2.2.6 Section 3.1.2.3.7 Section 4.2 Section 4.2.1 Section 15.3.1
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break LR section 2.8.5.3.2	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.3.8 Section 3.1.2.3.9 Section 3.1.2.4.2 Section 4.2.1 Section 5.3.3 Section 6.3.1 Section 6.3.2 Section 15.3.2
Steam Generator Tube Rupture LR section 2.8.5.6.2	UFSAR Section 15.6.3 Section 15.6.3.4.3
New Fuel Storage LR section 2.8.6.1	UFSAR Section 9.1.2.4.1
Spent Fuel Storage LR section 2.8.6.2	UFSAR Section 9.1.2.4.1

Auxiliary Systems Pumps, Heat Exchangers, Valves and Tanks <u>LR section 2.8.7.1</u>	UFSAR Section 3.1.1 Section 3.1.2 Section 3.1.2.1.2 Section 3.1.2.1.4 Section 3.1.2.1.5 Section 3.1.2.4.5 Section 3.1.2.4.15 Section 3.1.2.6.2 Section 3.5 Section 3.6 Section 3.11 Section 5.4.5
Natural Circulation Cooldown <u>LR section 2.8.7.2</u>	UFSAR Section 14.6.1.5.6
Loss of Residual Heat Removal at Midloop <u>LR section 2.8.7.3</u>	UFSAR Section 5.4.5.4 Section 5.4.5.4.1 Section 5.4.5.4.3 Section 5.4.5.4.5

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MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Other Guidance
Source Terms for Input into Radwaste Management Systems Analyses LR section 2.9.1	UFSAR Section 3.1.2.6.1 Section 11.1 Section 11.2 Table 9.1-5 Table 9.3.4.4.8 Table 9.3.4.4.9 Table 9.3-11c
Radiological Consequences Analyses Using Alternative Source Terms LR section 2.9.2	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.

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	Other Guidance
Occupational and Public Radiation Doses LR section 2.10.1	UFSAR Section 2.9.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.10 Section 9.3.2.3.1 Section 11.1.1.2 Section 11.2.3 Section 11.3.3 Section 11.5 Section 12.1 Section 12.1.3 Section 12.3 Section 12.3.2 Section 12.3.2.1 Section 12.3.2.2.4 Section 12.3.2.2.6 Section 12.3.4 Section 12.5.2 Table 12.3-2 Table 12.3-5

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Other Guidance
Human Factors <u>LR section 2.11.1</u>	UFSAR Section 3.1 Section 3.1.1 Section 3.1.2 Section 3.1.2.2.10 Section 7.5.1.2 Section 7.7.6.3 Section 13.1.3.1 Section 13.2.2.1 Section 13.2.2.2 Section 13.5.1.2

Appendix D
Acronyms in addition to those in RS-001

AIF-GDC	Draft Atomic Industry Forum General Design Criteria
AFW	Auxiliary Feedwater
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ARV	Atmospheric Relief Valve
CCW	Component Cooling Water
CE	Combustion Engineering
CGG	Constellation Generation Group
COLR	Core Operating Limits Report
IPSAR	Integrated Plant Safety Assessment Report (NUREG-0821)
ISI	In Service Inspection
IST	In Service Testing
LAR	License Amendment Request
LR	Licensing Report
LRA	License Renewal Application
NYISO	New York Independent System Operator
ODB	Original Design Basis
ODCM	Offsite Dose Calculation Manual
PCWG	Performance Capability Working Group
PL/TB	Pressure Locking/Thermal Binding
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PTS	Pressurized Thermal Shock
RSG	Replacement Steam Generator
SAFW	Standby Auxiliary Feedwater
SCF	Stress Concentration Factor
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SGTP	Steam Generator Tube Plugging