November 3, 2005

Mrs. Mary G. Korsnick
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - REQUEST FOR ADDITIONAL INFORMATION RE: EXTENDED POWER UPRATE LICENSE AMENDMENT (TAC NO. MC7382)

Dear Mrs. Korsnick:

By letter to the Nuclear Regulatory Commission (NRC) dated July 7, 2005, as supplemented by letters dated August 15 and September 30, 2005, R.E. Ginna Nuclear Power Plant, LLC submitted an application requesting authorization to increase the maximum steady-state thermal power level at the R.E. Ginna Nuclear Power Plant from 1520 megawatts thermal (MWt) to 1775 MWt, which is a 16.8 percent increase. This requested change is commonly referred to as an extended power uprate (EPU).

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the EPU review. The specific information requested is addressed in the enclosure to this letter, and was sent to your staff by e-mail on October 24, 2005. During a telephone discussion with your staff on October 25, 2005, it was agreed that your response would be provided 45 days from the date of this letter.

The NRC staff considers that timely responses to requests for additional information (RAI) help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-1457.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Plant Licensing Branch A
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: RAI

cc w/encl: See next page

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Division of Operating Reactor Licensing
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Docket No. 50-244
Enclosure: RAI
cc w/encl: See next page
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cc:

Mr. Michael J. Wallace
President
R.E. Ginna Nuclear Power Plant, LLC
c/o Constellation Energy
750 East Pratt Street
Baltimore, MD 21202

Mr. John M. Heffley
Senior Vice President and
Chief Nuclear Officer
Constellation Generation Group
1997 Annapolis Exchange Parkway
Suite 500
Annapolis, MD 21401

Kenneth Kolaczyk, Sr. Resident Inspector
R.E. Ginna Nuclear Power Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, NY 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Peter R. Smith, President
New York State Energy, Research,
and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Carey W. Fleming, Esquire
Senior Counsel - Nuclear Generation
Constellation Generation Group, LLC
750 East Pratt Street, 17th Floor
Baltimore, MD 21202

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Ms. Thelma Wideman, Director
Wayne County Emergency Management
Office
Wayne County Emergency Operations
Center
7336 Route 31
Lyons, NY 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
1190 Scottsville Road, Suite 200
Rochester, NY 14624

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE EXTENDED POWER UPRATE LICENSE AMENDMENT

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

By letter to the Nuclear Regulatory Commission (NRC) dated July 7, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051950123), as supplemented by letters dated August 15 and September 30, 2005 (ADAMS Nos. ML052310155 and ML052800223, respectively), R.E. Ginna Nuclear Power Plant, LLC (the licensee) submitted an application requesting authorization to increase the maximum steady-state thermal power level at the R.E. Ginna Nuclear Power Plant (Ginna) from 1520 megawatts thermal (MWt) to 1775 MWt, which is a 16.8 percent increase. This requested change is commonly referred to as an extended power uprate (EPU). To complete its review, the NRC staff requests the following information:

METEOROLOGICAL INFORMATION

1. In Table 2.9.2-2 of the EPU Licensing Report (see Attachments 5 and 7 for non-proprietary and proprietary versions, respectively, to the July 7 application), the value for the 0-1 minute exclusion area boundary (EAB) tornado missile accident atmospheric dispersion factor (χ/Q value) was listed as 1.87×10^{-6} s/m^3. Table 3 of the NRC staff safety evaluation (SE) that supported Amendment No. 87, dated February 25, 2005 (ML050320491), approved the tornado missile accident χ/Q value as 2.17×10^{-6} s/m^3. In the footnote to Table 3, the NRC staff noted that the tornado missile accident χ/Q value of 2.17×10^{-6} s/m^3 was provided in a response to a request for additional information (RAI) dated December 3, 2004. In its response to this RAI, the licensee explained that the value of 2.17×10^{-6} s/m^3 was based upon the shortest EAB distance (450 meters) mentioned in the Ginna Updated Final Safety Analysis Report (UFSAR), rather than an EAB distance of 503 meters that had been used by the NRC staff in a prior χ/Q calculation.

    a. Explain why the χ/Q value of 1.87×10^{-6} s/m^3 should be used in the dose assessment supporting the EPU amendment application when the shortest EAB distance is 450 meters and the associated χ/Q value is 2.17×10^{-6} s/m^3.

    b. Was a 0-1 minute χ/Q value used for the low population zone (LPZ) tornado missile accident dose assessment? If so, what was the 0-1 minute χ/Q value used? If a 0-1 minute χ/Q value was not used, was the 0-8 hour LPZ χ/Q value of 2.51×10^{-5} s/m^3 used for the entire 0-8 hour time period?

RADIOLOGICAL CONSEQUENCES OF DESIGN-BASIS ACCIDENTS INFORMATION

1. In Section 2.9.2.2.3 of the Licensing Report, the radiological consequences analysis for the locked rotor accident was described. The discussion does not indicate if a radial peaking factor was applied in determining the source term for this design-basis accident

(DBA) analysis. Was a radial peaking factor applied, and if so, what value was applied? Provide the basis to support the value used.

2. In Section 2.9.2.2.4 of the Licensing Report, the radiological consequences analysis for the rod ejection accident was described. The release to the environment is assumed to occur through both containment atmosphere and reactor coolant system (RCS) inventory via primary-to-secondary leakage through the steam generators (SGs). What is the primary-to-secondary leakage rate assumed for each SG?

3. Table 2.9.2-6 of the Licensing Report indicates that the containment net free volume is 106 ft^3. This appears to be a misprint. Verify this parameter as used in the DBA dose analyses.

ELECTRICAL

1. Identify the nature and quantity of the megavolt amperes reactive (MVAR) necessary to maintain post-trip loads and minimum voltage levels as a result of the EPU.

2. Identify the MVAR contributions that Ginna will be credited for providing to the grid following implementation of the EPU.

3. After the implementation of the EPU, identify any anticipated changes in MVAR associated with Items 1 and 2 above.

4. Address the compensatory measures that the licensee would take to compensate for the depletion of the Ginna unit MVAR capability on a grid-wide basis. As a result of the implementation of the EPU, evaluate the impact of any MVAR shortfall on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document information exchanges with the transmission system operator.

5. Address whether the Station Blackout coping duration has changed as a result of the implementation of the EPU.

MATERIALS

1. In its evaluation of the effects of the 8.6 EF increase in temperature due to the EPU, the licensee listed the inspection requirements under First Revised NRC Order EA-03-009, Electric Power Research Institute Materials Reliability Program 117 (EPRI-MRP-117), and a potential American Society of Mechnical Engineers Boiler and Pressure Vessel Code (ASME Code) Case as requirements to manage the effects of primary water stress-corrosion cracking (PWSCC) on Alloy 690/52/152 materials. The licensee also stated that Ginna will continue to monitor the Industry programs and recommendations to manage the issue for the new vessel head and take appropriate actions as necessary. Provide more specific information as to what requirements will be followed at Ginna, or reference the pertinent commitment(s) that were accepted by the NRC staff under your license renewal application, to assure the effects of PWSCC will be managed.

2. Under its evaluation of the effects of the 3.2 EF decrease in bottom mounted instrument (BMI) penetrations temperature due to the EPU, the licensee listed the inspection requirements that may apply such as Materials Reliability Program (MRP) guidance and NRC Bulletin 2003-02. Provide more specific information as to what requirements will be followed at Ginna, or reference the pertinent commitment(s) that were accepted by the staff under the Ginna license renewal application, to assure the effects of PWSCC on the BMI penetrations will be managed.

3. Under its assessment for the effects of thermal aging of cast austenitic stainless steel (CASS), the licensee indicated that programs were proposed in Westinghouse Report WCAP-14575-A to manage the effects of thermal aging of CASS components. Furthermore, the licensee stated that a reconciliation of the subject WCAP lists applicant action items in Table 3.2.0-1.2 of the Licensing Report. Finally, the licensee stated that the 8.6 EF increase in the hot leg temperature was assessed due to the EPU and that the effect of this change in the service temperature on the thermal aging was considered. Discuss in detail the applicant action items for the subject WCAP and why the 8.6 EF increase in temperature due to the EPU is acceptable since there are action items associated with the WCAP that was referenced as the basis for acceptability. The discussion should include why the programs under the subject WCAP will adequately manage any increased thermal aging (if any) due to the 8.6 EF temperature increase.

<u>CIVIL AND MECHANICAL ENGINEERING</u>

1. The licensee stated on Page 2.2.2.2-5 of Licensing Report that, "[d]uring the review of the present piping stress analysis design bases for the Service Water and Component Cooling Water Systems, some inconsistencies were identified between the operating temperatures assumed in the analyses and the maximum possible operating temperatures. The impact of these differences in operating temperature upon the piping thermal stresses has been evaluated. The evaluations have determined that the existing piping design is acceptable due to the flexibility of the piping systems and high thermal stress margins available in the existing analyses." Provide a summary of the evaluation methodology, including acceptance criteria and results identifying the specific margins available in the existing analysis. In addition, provide the specific inconsistencies that were identified between the assumed operating temperature and maximum possible operating temperature.

2. On Page 2.2.2.2.-5, the licensee stated "[f]or piping systems which will experience plant modifications (e.g., MSR [moisture separator reheater] piping and relief valve modification) to address EPU conditions, the piping and support evaluations will be performed as part of the overall design change package associated with the specific plaint modification." Provide a description of the modifications, including the location in the piping system and the EPU condition which necessitated the modification. Also, indicate when these evaluations would be available for staff review.

3. On Page 2.2.2.2-9, the licensee stated "[t]he results of the pipe support evaluations for systems impacted by EPU concluded that all supports remain acceptable, except for certain main steam and feedwater system pipe supports that require modification to accommodate the revised loads related to EPU conditions. The main steam and feedwater pipe support modifications are required to mitigate the larger flow induced

fluid transient loads that resulted due to EPU conditions. The majority of these support modifications are required to mitigate the larger loads resulting from a turbine stop valve closure transient event. Also, one new snubber will also be installed on the main steam piping system. These pipe support modifications will be installed before the implementation of the EPU." Provide the following:

a. Specific location and description of the main stream and feedwater system pipe supports that require modification to accommodate the revised loads related to EPU.

b. Description of the modification to supports to accommodate the larger flow reduced transient loads, inducing the magnitude and nature of the existing and EPU loadings.

c. Description of the analytical evaluation of the new snubbers in the main steam piping system.

4. Identify all piping systems that would experience high flow rates resulting from the EPU. Also, discuss the potential vibration issues that are likely to occur as well as the mitigating measures and corrective actions which would be adopted. Clarify whether or not the proposed vibration testing and verification program subsequent to the implementation of the EPU will conform with the requirement of ASME OM Code, Part 3, "Requirements for Preoperational and Start-up Vibration Testing of Nuclear Plant Piping System" and OM Code, Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems."

5. The licensee provided a summary of the piping analysis results at EPU conditions in Table 2.2.2.2-1 of the Licensing Report. For some of the piping systems (e.g., main steam outside containment, feedwater inside loop B), the EPU stresses are very close to allowable values. Provide detailed calculations, including description of the service loading conditions operating temperatures transient, and flow induces vibration, for those cases where design margin have been determined to be 0.90 or greater.

6. The licensee provided the maximum ranges of stress intensity and maximum cumulative fatigue usage factors from the analytical evaluation of the reactor vessel in Table 2.2.2.3-1 of the Licensing Report. In some instances, the calculated maximum range of stress intensity exceeded the limiting value. A simplified elastic-plastic analysis per Section NB3228.5 of the ASME Code was performed for these locations and shown to satisfy all applicable requirements. Provide a more detailed summary of the analysis results for the following locations where the calculated values are close to limiting values.

- Closure Studs
- Control Rod Drive Mechanism (CRDM) Nozzle
- Inlet Nozzle to Shelf Function
- External Support Brackets

7. The licensee provided a comparison of the calculated vessel support loads in Tables 2.2.2.3-3 and 2.2.2.3-4 of the Licensing Report. Discuss, and justify the basis for, the

limiting loads determined by Gilbert Associates for the normal/operating and faulted conditions.

8. The licensee stated on Page 2.2.2.3-2 of the Licensing Report " [a]nalysis of flow induced vibration is not included in the licensing basis for Ginna. However, it was considered for more susceptible components that would experience a significant flow increase under EPU conditions. Reactor vessel components were evaluated and deemed unaffected by EPU conditions due to their heavy construction and small increase in flow, if any." Identify the components which were considered more susceptible to flow induces vibration. Also, provide justification to demonstrate their structural adequacy.

9. The licensee provided the EPU evaluation summary at critical locations of primary and secondary side pressure boundary components in Table 2.2.2.5.2-1. The results indicated that, at several critical locations, the fatigue limit was determined to be very close to the allowable value 1.0. Provide a detailed summary of the analytical evaluation in the following locations:

   - Cone-to-lower-shell juncture
   - Lower-shell at ring girder
   - Secondary mainway studs
   - Primary head at support
   - Tubesheet blowdown and mainway drain holes
   - Lower Shell at Tubesheet
   - Lower shell handholes and studs
   - Seal Skirt

   In addition, discuss the fatigue monitoring and/or other mitigating measures relative to the primary mainway studs and other locations where the calculated fatigue limit does not meet the 40-years design life limit. Also, provide a detailed discussion regarding the decrease in cumulative usage factors (CUFs) in the lower shell handholes and the seal skirt for the EPU condition.

10. The licensee provided the calculated stresses and fatigue usage factors for the reactor internal component in Table 2.2.3-3 of the Licensing Report. For several components, the stress intensity exceeds the 3Sm limit and a simplified elastic-plastic analysis was performed to calculate fatigue strength, as allowed by ASME Code, Section III, Subsection NB 3228.5. Provide a detailed summary of the evaluation for the following components: lower support plate, lower core plate, core barrel assembly outlet nozzle, thermal shield flexure bolts, and lower radial inserts.

11. On Page 2.2.4-10 of the Licensing Report, the licensee stated that "[d]ue to the increase in main feedwater system flow at EPU conditions, a modification to the main feedwater regulating valves to allow for proper flow control of these valves will be implemented (refer to LR Section 2.5.5.4. Condensate and Feedwater). The EPU does not affect the Technical Specification (TS) requirement for these valves to close in less than or equal to 10 seconds. The design specification associated with the main feedwater regulating valve modification includes the requirement that the modified valves close in less than or equal to 10 seconds. Any required changes to inservice Testing Program requirements

for the modified main feedwater regulating valves will be developed as part of the plant change process." Provide a more detailed discussion of the required changes to the inservice testing program requirements for the proposed modification.

12. On Page 2.2.4-11, the licensee stated that "[t]he required standby auxiliary accident analysis flow rate will increase from 200 gpm at current conditions to 235 gpm at EPU conditions. The Inservice Testing program analysis/procedures for the standby auxiliary feedwater pumps will be revised to address testing the standby auxiliary feedwater pumps at EPU conditions." Discuss the changes in program analysis and procedures that are likely to occur as a result of the proposed modification. Indicate when the modification would be available for review by the NRC staff.

13. On Page 2.2.4-11, the licensee stated that, "[a]s addressed in the design analysis which determines check valve safeguards flow rates, the check valves in the standby auxiliary feedwater pump suction and discharge lines have an open position safety function to pass 200 gallons per minute (gpm) from the standby auxiliary feedwater pumps to the steam generators. The Inservice Testing program analysis / procedures for the standby auxiliary feedwater pump suction and discharge check valves will be required to be revised to address testing the standby auxiliary feedwater pump suction and discharge check valves at EPU flow conditions." Discuss more specifically what revisions in the standby auxiliary feedwater pump suction and discharge check valves testing program are likely to occur as a result of operation at EPU conditions. Also, indicate when these proposed revisions would be available for staff review and approval.

14. On Page 2.2.4-11, the licensee stated that the Inservice Testing analysis and procedures for the standby auxiliary feedwater (AFW) pumps and their suction and discharge check valves will need to be revised to reflect EPU conditions. Discuss the change in operating conditions for these components from original licensed thermal power (OLTP) to EPU power levels, and the status of the completion of the revision to the IST analysis and procedures.

15. In Table 2.2.4-2 of the Licensing Report, the licensee stated that residual heat removal (RHR) cross-connect pump section motor-operated valves (MOVs) 704A and B have certain performance parameters calculated for EPU conditions. Discuss the impact of these parameters on the capability of MOVs 704A and B to perform their safety functions under EPU conditions.

16. In Section 2.2.4, the licensee discussed the potential impact of EPU conditions on the performance of safety-related MOVs at Ginna. Discuss, with examples, the potential impact of EPU conditions on other safety-related power-operated valves at Ginna (such as air-operated and solenoid-operated valves).

17. Discuss whether any safety valves or safety relief valves might need to operate with liquid flow to perform their safety functions following EPU implementation, and the justification for such reliance on those valves to perform their safety functions.

18. In Section 2.5.5.4 of the Licensing Report, the licensee discussed the evaluation of the feedwater and condensate systems and components for EPU conditions. Discuss the evaluation of potential adverse flow effects, such as flow-induced vibration, on the

feedwater and condensate piping and components (including sample probes) as a result of EPU operation.

19. In Table 2.12-1, the licensee listed vibration monitoring to be conducted at 85, 88, 91, 94, 97, and 100% of EPU power level as part of the Ginna EPU Power Ascension Test Plan. Discuss the activities to be performed during these hold points, including: (a) vibration monitoring of the reactor, steam, feedwater, and condensate systems and components, and (b) plant walkdowns and inspections of plant systems and equipment. Discuss the acceptance criteria to be applied to the vibration data, walkdowns, and inspections and the actions to be taken if the acceptance criteria are not satisfied.

20. As referenced in Section 2.2.2.1.5.4, "Tube Vibration," the Ginna SG tubes were evaluated by calculating the most limiting fluid-elastic stability ratio and the maximum turbulent induced bending stresses on the limiting tube. Provide a summary of evaluation regarding the vortex induced vibration stresses on the limiting SG tubes for the EPU condition.

21. As referenced in Section 2.2.2.5, provide a summary of the evaluation for the SG internals (baffle, feedwater sparger, steam dryer, flow reflector, tubes) and their supports with respect to the maximum stress and fatigue usage factor for the EPU condition. Also, identify the Code, and Code edition for the evaluation of the proposed EPU. If different from the Code of record, provide the justification. Also, provide an evaluation of flow induced vibration of the steam dryer, dryer supports and flow-reflector with respect to the fluid-elastic instability, acoustic loads and vortex shedding due to steam flow for the EPU.

22. In Section 2.2.2.3 of the Licensing Report, the licensee indicated that the 40-year design transient sets have been shown to be bounding for 60 years of operation, and the fatigue evaluations performed for the EPU program demonstrate that the current design is acceptable to support EPU conditions for 60 years of plant operation. Confirm whether the EPU evaluation was performed for 60 years of plant operation at EPU condition. Explain how the 40-year design transient sets have been shown to be bounding for 60 years of operation.

PLANT SYSTEMS

Spent Fuel Pool Cooling System

1. Because the alternate spent fuel pool (SFP) makeup capability is not quite adequate for the worst-case boil-off rate of 52.8 gpm, the licensee indicated that the off-load time can be delayed until the boil-off rate is reduced to less than 50 gpm. Confirm that the criteria in the Ginna Technical Requirements Manual (TRM) and in the UFSAR for performing normal and full-core offloads will be revised to include verification that both the normal and the alternate SFP makeup capability will exceed the maximum SFP boil-off rate that could occur should there be a complete loss of SFP cooling.

2. The evaluation discussed in the Licensing Report is based on the worst-case decay heat load that is generated from 1321 fuel assemblies. Ginna TS 4.3.3 currently permits up

to 1879 fuel assemblies to be stored in the SFP. Explain why the worst-case decay heat load analysis is not consistent with the current TSs.

Service Water System

1. Confirm that the results of heat exchanger performance monitoring per Generic Letter (GL) 89-13 recommendations demonstrate acceptable performance for EPU conditions.

Component Cooling Water (CCW) System

1. The Licensing Report indicated that administrative controls will be used to limit the CCW outlet temperature from the RHR heat exchangers during normal plant cooldown evolutions following EPU operation to 170 EF. Explain why it is necessary to impose this new temperature limit for EPU operation, and discuss how it will be implemented and managed.

2. Explain what impact the proposed EPU will have on the flow-induced vibration considerations discussed in Section 9.2.2.4.1.6 of the Ginna UFSAR, including a discussion of any additional limitations that must be relied upon.

3. Section 9.2.2.4.3 of the Ginna UFSAR indicates that following a loss-of-coolant accident (LOCA), one CCW pump and one CCW heat exchanger are capable of accommodating the heat loads. However, the Licensing Report indicates that both CCW heat exchangers are relied upon for decay heat removal during the recirculation mode following a LOCA. Explain this apparent inconsistency.

4. Explain what impact the proposed EPU will have on the capability of the CCW system to cool the plant to cold shutdown conditions within 72 hours in accordance with Appendix R requirements, as described in Section 9.2.2.4.3 of the Ginna UFSAR.

REACTOR SYSTEMS

General

1. Confirm that only safety grade systems and components are credited in the re-analyses of all transients and accidents in Licensing Report.

Thermal and Hydraulic Design

1. Provide an evaluation that shows the P-7 setpoint (#8.5% RTP), the P-8 setpoint (< 49% rated thermal power), and the P-9 setpoint (> 50%) will continue to perform their intended functions after the EPU.

2. Provide a core limits and protection line diagram, equivalent to UFSAR Figure 15.0-1 for the EPU.

3. Provide a tabulation of the thermal design parameters and compare them to values assumed in safety analyses to demonstrate that the safety analyses assumptions are conservative.

<u>Rupture of a Steam Pipe – Hot, Zero Power (HZP) Core Response and Hot, Full Power (HFP) Core Response</u>

1. Provide a graph of SG mass versus time.

2. The graph of core average temperature vs. time on Page 2.8.5.1.2-11 of the Licensing Report only goes up to 550 EF, discuss whether it should go to at least 557 EF (e.g., the no-load average coolant temperature)?

3. Justify the 10-minute assumption for operators to manually close the main steam isolation valves (MSIVs) (see Page 2.8.5.1.2-7 of the Licensing Report) for the smallest break that does not actuate protection.

4. Verify that the break analyzed on Page 2.8.5.1.2-7 is the smallest break that does not actuate protection.

5. On Page 2.8.5.1.2-14, the first paragraph, the third sentence of Section 2.8.5.1.2.2.2.2 (Input Parameters, Assumptions and Acceptance Criteria) states that "[w]hen 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.". What is meant by the portion which states "or are conservatively applied to the limiting transient condition in the calculation of the minimum DNBR."?

6. On Page 2.8.5.1.2-14, Section 2.8.5.1.2.2.2.2 (Input Parameters, Assumptions and Acceptance Criteria), the fourth bullet (Feedwater Temperature), the last sentecnce asserts that "Sensitivity studies have shown that HFP SLB results are not influenced by the assumed initial feedwater temperature.". Does the phrase "not influenced" mean that the initial feedwater temperature has no effect on the results? If so, what is the reason for this phenomenon?

7. In the nuclear power versus time graph on Page 2.8.5.1.2-18, power seems to turn and increase again before decreasing drastically at the time of 14 seconds. What is the cause of this slight upturn?

8. In Table 2.8.5.1.2.2.1-1 on Page 2.8.5.1.2-9 (the Sequence of Events for the HZP case), the MSIVs close 7 seconds after the safety injection (SI) system actuation signal is generated at 8.4 seconds. The low steam pressure SI system actuation setpoint is reached at 1.4 seconds. This implies that there is no processing time between when the low steam pressure SI system actuation setpoint is reached and the SI system actuation signal is produced. Verify if this is correct.

9. In Table 2.8.5.1.2.2.1-1, the licensee stated that the high-head SI pump reaches rated speed at 12 seconds after the SI system actuation signal is generated at 15.4 seconds, which implies that the SI system actuation signal is generated at 3.4 seconds. Verify if this is correct.

Loss of External Load

1. The licensee indicated that the allowable peak RCS pressure during an anticipated operational occurrences (AOO) is 2485 psig (100% of the RCS design pressure). Confirm that this value is the licensing basis for Ginna.

2. In Section 2.8.5.2.1.2.1, the licensee indicated that Ginna is designed to accept a 50% rapid decrease (200% per minute) in electrical load while operating at full power without actuating a reactor trip. Describe the capacity of the steam dump system, and confirm that the operation of the steam bypass valves are fast enough to handle this transient given EPU conditions.

3. Provide a quantitative discussion regarding instrumentation uncertainties to support the use of nominal values for RCS temperature and pressure in the analysis as initial plant conditions while the revised thermal design procedure is used.

4. The limiting single failure assumed in the analysis for loss of load transient is the failure of one train of the reactor trip system (RTS). This assumption would not affect the transient scenario at all. Discuss the process used for selecting the limiting single failure in the safety analysis that could cause negative effects to the transient.

Loss of Normal Feedwater

1. Provide the transient data for departure from nucleate boiling ration (DNBR), peak RCS pressure and peak main steam system pressure to support the licensee's conclusion that the acceptance criteria for this event are met.

2. Provide a discussion of the limiting single failures assumed in the analyses.

3. Discuss the provisions made in plant emergency operating procedures (EOPs) for controlling AFW at the beginning of the event to prevent excess cooldown during this event.

Feedwater System Pipe Breaks

1. What is the physical basis for the 1.418 sq. ft. break size? Is this the feed line pipe size, the equivalent flow area through the feed ring, or is it something else?

2. Justify crediting closure of the check valve to isolate the faulted SG.

3. In Table 2.8.5.2.4-1, explain the statement that the steamline check valves close on turbine trip.

4. How is SG water level determined during a feedline break?

5. What operator actions were credited for the feedline break analysis, and when were they assumed to occur?

6. What is the transient break flow quality? How is that determined?

Loss of Forced Reactor Coolant Flow

1. Discuss the single failure modeled in the analysis. Also, provide the input parameters used in the analysis as described in UFSAR Section 15.3.1.4.1, and state why these values are conservative.

2. Provide the technical justification explaining why the results of the partial loss of flow event were non-limiting compared to the complete loss of flow event and why the complete loss of flow is the most limiting event between the two.

3. Provide the transient data and values for DNBR, peak RCS pressure, and peak main steam system pressure to support the licensee's conclusion that the acceptance criteria for this event are met.

Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

1. In Section 2.8.5.3.2, the licensee only provided the results for the locked rotor event. Clarify whether the locked rotor or reactor coolant pump shaft break event is more limiting, and provide the basis for this conclusion.

2. In Section 2.8.5.3.2.3, the licensee stated 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 analysis. Provide the technical justification that shows it is acceptable to remove this time from the current licensing basis.

3. In Section 2.8.5.3.2.3, the licensee stated with respect to secondary overpressurization, that there are other transients that demonstrate that the secondary pressure limit is met for this event. Provide the technical justification that explains how these transients are more limiting and how the secondary side pressure acceptance criteria continues to be met.

4. State the single failure modeled in the analysis and justify why this is the worst case modeled. Was a loss of offsite power considered? Explain why or why not.

5. Provide the DNBR value for this event and quantify the fuel failed if the DNBR limit was exceeded.

Overpressure Protection during Power Operation

1. In the Licensing Report, the licensee included descriptions of provisions to address overpressure protection for Ginna operating at the uprated power. This information addresses only the change in the pressurizer safety valve upper lift setting; not the adequacy of the safety valve capacity. Although UFSAR Table 5.2-1 refers to the ASME Code, Section III, Nuclear Vessels, 1965, it does not detail the analyses that were performed assuming the uprated power to demonstrate the adequacy of the safety valve capacity and to quantify the sufficiency of the design margin of the safety valve(s).

Note that WCAP-7769, Revision 1, provides a demonstration of compliance for Ginna, based upon ASME Code, Section III, Articles NB-7300 and NC-7300, "Protection Against Overpressure," 1971. However, this demonstration was for Ginna operating at 1518.5 MWt.

Provide the results of analyses based upon methods consistent with those of WCAP-7769 (including credit for the second (or later) safety grade trip from the reactor protection system) to show continued sufficiency of margin of design capacity for the Ginna pressurizer and steam line safety valves, with Ginna operating at the uprated power of 1775 MWt.

Chemical and Volume Control System (CVCS) Malfunction and Boron Dilution

1. Three positive displacement charging pumps can deliver a maximum of 180 gpm (charging flow is normally maintained at 46 gpm). The nominal steam volume in the pressurizer is 333 cu. ft., at the uprated power level (reduced from 397 cu. ft.). Dividing the steam volume by the maximum charging flow indicates that it would take less than 14 minutes to fill this volume. Tripping the reactor could delay the filling of the pressurizer. Perform a transient analysis to better estimate the pressurizer fill time, and provide information, such as simulator test results and emergency operating procedures, to confirm that the operator would have adequate time and indication to terminate the transient.

2. Explain why Mode 3 and 4 inadvertent Boron Dilution events are not included in Ginna's EPU application.

3. Provide a description of the analysis which shows that the inadvertent Boron Dilution event requirements continue to be met, and include all inputs, assumptions, limitations, and results of that analysis. Identify any controls necessary to ensure the analysis remains bounding. Include the justification for the inputs and assumptions.

4. Confirm that the Ginna TS 3.9.1 requirement for the refueling boron concentration to be greater than 2300 parts per million (ppm) continues to provide sufficient shutdown margin under EPU conditions, include transition and steady state cycles.

New Fuel and Spent Fuel Storage

1. In its License Report, the licensee described the current licensing bases for the new and spent fuel storage systems. In Section 2.8.6.2, “Spent Fuel Storage,” the licensee stated that the acceptance criteria for the spent fuel criticality analysis is based on maintaining the effective multiplication factor (k_{eff}), including all biases and uncertainties, less than 1.0 with full density unborated water and less than 0.95 with credit for borated water. Additionally, in Section 2.8.6.1, “New Fuel Storage,” the licensee stated that it operated the new fuel storage racks under a 10 CFR 70.24 exemption for criticality monitors.

   The NRC staff reviewed the Ginna new and spent fuel storage licensing bases, including any previously NRC-approved licensing actions, to determine the appropriate regulatory criteria for reviewing the proposed EPU. On July 16, 1997, the NRC issued

Ginna an exemption to the requirements of 10 CFR 70.24 for criticality monitors in the spent fuel pool. Subsequently, on July 30, 1998, the NRC issued Amendment No. 72 to the Ginna operating license to revise the criticality licensing basis of the Ginna spent fuel pool. The TS changes approved in that amendment were based on maintaining the k_{eff} less than 0.95 (not 1.0) with full density unborated water. The 1998 amendment invalidated the previous 10 CFR 70.24 exemption because it resulted in a change to the licensing basis.

On November 12, 1998, the NRC issued 10 CFR 50.68, "Criticality accident requirements." 10 CFR 50.68(a) requires that "Each holder of a construction permit or operating license for a nuclear power reactor issued under this part,...shall comply with either 10 CFR 70.24 of this chapter or the requirements of paragraph (b) of this section." The NRC staff requests that the licensee describe how it complies with either 10 CFR 70.24 or 10 CFR 50.68.

In Section 2.8.6.2.2.3, "Description of Analyses and Evaluations [for Spent Fuel Storage]", the licensee stated that the EPU core power level would not change the limiting axial burnup profile that was assumed in the UFSAR analysis. However, the licensee did not provide a technical justification for this conclusion. The NRC staff is concerned that to achieve the uprated power levels, fuel assemblies will be both burned harder and with different burnable poison loadings than was assumed in the current licensing basis criticality analyses. This could affect the axial burnup profiles of the irradiated assemblies. Describe the analysis performed to determine that under uprated conditions, the axial burnup profile assumed in the current spent fuel storage licensing basis remains bounding.

2. A major component of the licensee's current spent fuel storage licensing basis was the incorporation of a reactivity equivalencing methodology for performing criticality analyses. The NRC and nuclear industry have determined that the potential exists for this type of methodology to provide nonconservative results unless special care is taken to ensure that other parameters used in the analyses, such as soluble boron concentration, are not varied. Since it does not appear that the licensee has performed new criticality analyses to demonstrate that the new fuel design will meet NRC regulations, describe how the potential nonconservative effects of reactivity equivalencing have been evaluated under the proposed uprated conditions.

3. The licensee stated that the Westinghouse 14 x 14 422V+ fuel assemblies to be used under uprated conditions differ slightly in two important parameters from the Westinghouse fuel assemblies currently in use at Ginna. Based on its review of the licensee's EPU request, the NRC staff is unclear of all the differences between the new 422V+ fuel assemblies and the design basis fuel assemblies used in the new and spent fuel storage criticality analyses. Therefore, provide a detailed table showing all of the design parameters for the 422V+ and design basis fuel assemblies, including the allowed tolerances. Additionally, for any parameters where the new 422V+ fuel assembly design is not bounded by the design basis fuel assembly design, describe any evaluations or analyses performed to demonstrate that NRC regulations and safety limits are met.

LOCA Analysis

1. In order to show that the referenced, generically approved LOCA analysis methodologies apply specifically to the Ginna plant, provide a statement that the applicant and vendor have ongoing processes that assure that the ranges and values of the input parameters for the Ginna LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters. Furthermore, if the Ginna plant-specific analyses are based on the model and or analyses of any other plant, then justify that the model or analyses apply to Ginna (e.g., the model wouldn't apply to Ginna, if the other design has a different vessel internals design).

2. Provide a justification that the 1.5, 2, and 3-inch break sizes are sufficient to determine the limiting small-break LOCA (SBLOCA). If the SBLOCA is limiting for peak clad temperature, local oxidation, or core wide oxidation, present the limiting information requested above for the limiting SBLOCA break size if other than 1.5, 2, or 3 inches.