December 23, 2005

LICENSEE: R.E. Ginna Nuclear Power Plant, LLC

FACILITY: R.E. Ginna Nuclear Power Plant

SUBJECT: SUMMARY OF NOVEMBER 17, 2005, MEETING WITH R.E. GINNA NUCLEAR POWER PLANT, LLC, REGARDING EXTENDED POWER UPRATE APPLICATION (TAC NO. MC7382)

On November 17, 2005, the Nuclear Regulatory Commission (NRC) staff met with representatives of R.E. Ginna Nuclear Power Plant, LLC (the licensee) and its contractor, Westinghouse Electric Company, in a Category 1 public meeting held at the NRC offices at 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss information regarding the licensee's application dated July 7, 2005, for a 16.8 percent increase in the maximum steady-state thermal power level at the R.E. Ginna Nuclear Power Plant (Ginna). This level of power increase is generally referred to as an extended power uprate (EPU). A list of attendees is provided as Enclosure 1.

The agenda for this meeting consisted of discussions regarding the loss-of-coolant accident (LOCA), long-term cooling, and boron precipitation analyses supporting the EPU application. In particular, the discussion centered on issues presented in the NRC staff's requests for additional information on October 25 and 28, 2005, and issues that the NRC staff identified during its November 1-3, 2005, audit of supporting calculations. The presentation handout slides are provided as Enclosure 2.

For the LOCA analyses, information was presented regarding the analyses and modeling that were conducted. Because Ginna utilizes upper plenum injection, Westinghouse stated that the large-break LOCA analysis is best addressed by modeling a hot-leg break. In this regard, recirculation flow from the containment sump may begin in about 24 minutes. Upon questioning from the NRC staff, the licensee indicated that no credit was taken for containment overpressure for pump net-positive suction head. The licensee agreed to provide COBRA/TRAC computer code results to better understand liquid mixing volumes and boil-off. The licensee also stated that it would be conducting simulator and operator training within several months of the outage to validate operator performance times for such aspects as restoration of cold-leg injection.

In the small-break LOCA analysis, the concern is for a cold-leg break. Westinghouse indicated that the mixing volumes from the NOTRUMP analysis were used for the boric acid precipitation analysis and solubility was based at atmospheric conditions. Break sizes from 1.1 to 4 inches were evaluated to determine reactor coolant system (RCS) depressurization and the resultant boron precipitation times, the results of which were used to test operator actions. The licensee presented a draft plan for responding to the concerns about post-LOCA long-term cooling (Enclosure 2). The NRC staff stated that it would be looking at the guidelines and time limits for initiating depressurization as a job performance measure. The staff also questioned the need to include the operator action to depressurize the RCS in the emergency operating procedures (EOPs). The concern is that, without an explicit action in the EOPs, there may be no assurance that the 1-hour time to initiate cooldown in the analysis will always be achieved. In particular, the staff asked about the licensee's plans for training on postulated bottom-mounted instrument tube breaks. Further, the NRC staff asked about the testing of the service water supply to auxiliary feedwater.

The NRC staff discussed its observations and issues identified during an audit of the calculations supporting the EPU application. The licensee agreed to answer the following questions:

1. With respect to the rod withdrawal at power analysis (Section 2.8.5.4.2 of the Licensing Report):

   a. Figure 2.8.5.4.2-11, which depicts pressurizer pressure versus time for the RCS pressure case, indicates an uninterrupted rise in pressurizer pressure, to almost 2700 psia, until the reactor is tripped. It appears that the pressurizer safety valves are not modeled, and this seems to be an overly conservative assumption. Explain the modeling.

   b. Results of this event are presented as an assembly of the minimum departure from nucleate boiling ratio (DNBR) results of many transient analysis cases, mapping the minimum DNBRs as a function of initial power level, core reactivity feedback, and reactivity insertion rate. From this, it is concluded that the automatic reactor protection system prevents fuel clad damage for all analyzed cases. In a similar fashion, analyze or evaluate the case(s) of operation at low power level with only one loop in operation.

2. With respect to the steam system piping failures analysis (Section 2.8.5.1.2), describe how the shutdown margin is determined and used in the single-loop in operation case.

3. With respect to the spurious safety injection and chemical and volume control system (CVCS) malfunction event (Section 2.8.5.5), verify that the operator, following prescribed procedures, can terminate the CVCS malfunction event before the pressurizer fills (about 13 minutes).

4. With respect to the boron dilution event (Section 2.8.5.4.5):

   a. Identify the Ginna licensing basis for the boron dilution event for all modes. Indicate (1) whether the time allotted for operator action begins with operator notification or not, and (2) which operating modes must be analyzed.

   b. Explain why the parameters used in the boron dilution calculations are conservative.

   c. Since some sections of the boron dilution submittal are written in past tense, verify that the acceptance criteria are currently satisfied.

5. With respect to the functional design of the control rod system (Section 2.8.4.1), the licensee stated that "[c]hanges to the rod position indication systems ... are currently being assessed to ensure that correct individual rod position indications are available to

the operator." Provide the results of that assessment. Provide the rod drop time analysis.

6. With respect to the fuel system design (Section 2.8.1) and nuclear design (Section 2.8.2):

   a. The Fuel Criteria Evaluation Process (FCEP) for the 9 grid 422V+ fuel does not provide the overall assembly flow loss coefficient, nor an explanation of how it is determined. Please provide that information, either in the FCEP or in the Ginna submittal.

   b. The analysis for the transition cycles indicates that the current fuel (Optimized Fuel Assembly) could possibly exceed 60 gigawatt-day per metric ton of uranium (GWD/MTU) during the first transition cycle. Since this does not represent an actual core loading, it is not necessary, at this time, to submit any information to support exceeding 60 GWD/MTU. The licensee agreed to provide this information in the future, in the event that an actual core loading is predicted to exceed 60 GWD/MTU.

   c. The issue and commitment in Item (b), above, also apply to the 422V+ fuel assemblies in the first equilibrium cycle.

During the meeting, the licensee also discussed the proposed power ascension and testing plans associated with the implementation of the EPU. The NRC staff discussed and clarified its prior concerns on large transient testing.

Members of the public were not in attendance. Public meeting feedback forms were not received.

Please direct any inquiries to me at 301-415-1457 or pdm@nrc.gov.

/RA/

Patrick Milano, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. List of Attendees
2. Licensee Handout

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the operator." Provide the results of that assessment. Provide the rod drop time analysis.

6. With respect to the fuel system design (Section 2.8.1) and nuclear design (Section 2.8.2):

   a. The Fuel Criteria Evaluation Process (FCEP) for the 9 grid 422V+ fuel does not provide the overall assembly flow loss coefficient, nor an explanation of how it is determined. Please provide that information, either in the FCEP or in the Ginna submittal.

   b. The analysis for the transition cycles indicates that the current fuel (Optimized Fuel Assembly) could possibly exceed 60 gigawatt-day per metric ton of uranium (GWD/MTU) during the first transition cycle. Since this does not represent an actual core loading, it is not necessary, at this time, to submit any information to support exceeding 60 GWD/MTU. The licensee agreed to provide this information in the future, in the event that an actual core loading is predicted to exceed 60 GWD/MTU.

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Patrick Milano, Senior Project Manager
Plant Licensing Branch I-1
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MEETING ATTENDEES

R.E. GINNA NUCLEAR POWER PLANT, LLC

R.E. GINNA NUCLEAR POWER PLANT

NOVEMBER 17, 2005

NRC Staff:

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Samuel Miranda	Reactor Engineer	NRR/Division of Safety Systems (DSS)/PWR Systems Branch (SPWB)
Leonard Ward	Sr. Reactor Engineer	NRR/DSS/Nuclear Performance and Code Review Branch
Kent Wood	Reactor Engineer	NRR/DSS/SPWB
Patrick Milano	Sr. Project Manager	NRR/Division of Operating Reactor Licensing (DORL)/Plant Licensing Branch I-1 (LPLA)
John Stang	Sr. Project Manager	NRR/DORL/LPLC

Constellation Energy/R.E. Ginna Nuclear Power Plant, LLC:

Name	Title
Mark Flaherty	Manager, Fleet Licensing
Mark Finley	Manager, Power Uprates
George Wrobel	Manager, Licensing

Westinghouse Electric Company:

Name	Title
Dave Dominicis	Project Manager
Dave Sklarsky	Project Manager
Patricia E. Vaughn	Project Manager
Brett Kellerman	Senior Engineer
Dawn Crytzer	Senior Engineer
David Fink	Principal Engineer
J. Hartz	Principal Engineer
Andy Gagnon	Principal Engineer
Jayashree Iyengar	Contractor for Westinghouse

Other Participants: None

Enclosure 1