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SUBJECT: Responds to NRC 970924 ltr re violations noted in insp rept 50-244/97-201 on 970609-0815.Corrective actions:will revise procedure S-8A to reflect current mfg's recommended min flow & added check valves 753A,753B,758A & 758B to IST program.

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November 25, 1997

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: 60 Day Response to NRC Inspection Report No. 50-244/97-201, A/E Inspection

Reference: NRC Letter S.A. Richards to R.C. Mecredy (RGE), R.E. Ginna Nuclear Power Plant Design Inspection (NRC Inspection Report No. 50-244/97-201), dated September 24, 1997.

During the period from June 9 through August 15, 1997, the NRC conducted a design inspection of the R.E. Ginna Nuclear Power Plant. The above referenced letter transmitted the inspection report, with a request for a response providing a schedule for completion of corrective actions associated with the unresolved items and inspector follow items identified in Attachment 1 of the inspection report.

Attachment 1 to this letter provides the requested schedule. Should you have any questions concerning the attached schedule please contact Brian Flynn at (716) 771-3734.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

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Attachment

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ATTACHMENT 1
Responses to A/E Inspection Open Items

1. *NRC Item 50-244/97-201-01 (IFI), CCW Pump Minimum Flow*
The team observed that procedure S-8A "Component Cooling Water System Start-up and Normal Valve Alignment," Revision 36, stated that the minimum required flow for a CCW pump was 230 gpm, which was less than 10 percent of the best efficiency point (BEP) flow rate of the pump. NRC Bulletin 88-04, "Potential Safety-Related Pump Loss," recommended that the minimum flow should be 25 percent of the BEP for this size pump. The licensee's response to the NRC concerning Bulletin 88-04 did not consider the CCW pumps. The team questioned the ability of the CCW pumps to operate without degradation at a minimum flow of 230 gpm. During the inspection, the licensee contacted the pump vendor, who stated that the minimum flow should be 15 percent of the BEP, which was 420 gpm. The licensee issued AR 97-1166 to resolve this discrepancy and stated that procedure S-8A would be revised appropriately.

Investigation of this apparent discrepancy reveals that the 230 gpm minimum flow limit was specified in the original system start-up test procedure from Westinghouse. More recent guidance from the manufacturer indicates that 15% of the BEP should be used.

The configuration of the CCW system at Ginna is such that the minimum flow value is rarely, if ever approached, and therefore long term degradation of these pumps is not expected to occur from this condition. Procedure S-8A will be revised to reflect the current manufacturer's recommended minimum flow (435 gpm) by 12/31/97.

2. *NRC Item 50-244/97-201-02 (URI) Valve Testing*
The team noted that check valves 753A and B were not required to be leak tested in the in-service inspection program. These check valves form the boundary between high pressure (2500 psig) piping which could be exposed to RCS pressure and the low pressure (150 psig) piping. Should valves 753A or B leak in the event of a RCP thermal barrier cooler failure, the low pressure CCW piping could be exposed to a pressure above design pressure. Upon questioning by the team, the licensee stated that valves 753A and B would be added to the in-service inspection program and require leak testing. The licensee also issued AR 97-1187 to evaluate the current condition. 10CFR50.55a required in-service inspection in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel code. This code requires testing of valves which perform a safety function. It appeared that the licensee did not fully implement these requirements with regard to these check valves.

Check valves 753A and 753B act as a class boundary between the 150 psig CCW piping (upstream) and downstream CCW piping to the RCPs, rated at 2500 psig. The 150 psig CCW piping in containment is protected by relief valves 758A and 758B set

at 132 psig and sized to relieve 380 gpm. Therefore, RG&E does not believe that the check valves perform a safety function for isolation. However, as stated during the inspection, RG&E does agree that it is prudent to test these valves. RG&E has added these valves to our IST program for closure testing (CV-C), see RG&E correspondence R. Mecredy to G. Vissing (NRC) dated November 3, 1997, same subject.

10CFR50.55a requires in-service inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel code and testing of valves which perform a safety function. It appeared that the licensee did not fully implement these requirements with regard to valves 854; 860A, B, C and D; 896A and B; 897; and 898.

As discussed during the A/E Inspection, RG&E agrees that it is prudent to add valves 860A, B, C and D; valves 896A and 896B; and valves 897 and 898 and 854 to our IST program. These change have been made. Valves 860A, B, C, D, 897, and 898 were leak tested during the 1997 refueling outage with acceptable results. Valve 854 was radiographed to verify closure to satisfy a CV-C test. Valves 896A and B require a modification to provide capability to leak test. As discussed during the A/E Inspection, this is currently scheduled to be completed during the 1999 Refueling Outage.

3. *NRC Item 50-244/97-201-03 (IFI) CCW System Evaluation as Closed System*
UFSAR Section 6.2.4.4.5 stated that the containment penetration arrangement for the CCW lines to and from the reactor coolant pumps (penetrations 125, 126, 127 and 128) and the reactor support coolers (penetrations 130 and 131) satisfied the requirements of the current General Design Criterion (GDC) 57 in 10CFR50, Appendix A. This GDC addresses systems which are closed systems inside containment. The UFSAR stated that, for these closed systems to qualify as a containment isolation boundary, they must be safety-grade design and, in part, protected against missiles and pipe whip. Safety-grade design also includes protection against adverse effects of jet impingement from pipe breaks and cracks. The team reviewed Safety Evaluation (SE) NSL-0000-SE019, "Containment Isolation Assessment for Penetrations 125, 126, 127, and 128," Revision 0, and calculation DA-ME-95-088, "Evaluation of Effects of High Energy Line Break and Jet Forces on Reactor Support Coolers CCW Lines (Closed Loop Inside Containment)," Revision 0. These analyses documented the acceptability of the portion of the CCW system inside containment that passed through these penetrations as a closed system. The team identified the following deficiencies in these analyses:

1. Neither analysis evaluated reactor coolant system (RCS) lines other than the main loop and pressurizer surge lines. The stated justification was

that the entire RCS was covered by a leak-before-break exclusion in a letter from Dominic DiIanni, NRC, to Roger Kober, RG&E dated September 9, 1986, concerning the resolution of USI A-2, "Asymmetric LOCA Loads". However, this letter only applied to the main loop piping.

2. Both analyses incorrectly accounted for the effects of jet impingement. SE019 stated that jet forces were not significant since the jet pressure in psig was less than the post-accident containment pressure. DA-ME-95-088 stated that the CCW lines would not be damaged by jet forces since the jet pressure was less than the pipe internal design pressure. These comparisons were unrelated to the total jet force on the piping and its supports.

3. Neither analysis contained adequate documentation to enable a reviewer to readily understand and reconstruct the analysis. Specifically, complete documentation of the spatial analysis of the effects of high energy pipe ruptures and cracks was not included in the analyses.

4. DA-ME-95-088 incorrectly applied the leak-before-break exclusion to a section of high energy 10 inch RHR piping. As required by GDC 4 of 10CFR50, Appendix A, leak-before-break exclusions require approval from the NRC of a specific fracture mechanics evaluation for the piping; such an evaluation was not done for this RHR piping.

5. DA-ME-95-088 incorrectly excluded the effect of jet impingement from broken pipes with a smaller section modulus than the target CCW piping. Consideration of section modulus is only appropriate when analyzing pipes striking each other, as stated in a letter from Dennis M. Crutchfield, NRC, to John E. Maier, RG&E, dated February 22, 1982, entitled "Ginna - SEP Topic III-5.A, Effects of Pipe Break on Systems Structures and Components Inside Containment."

The licensee issued AR 97-1235 to address the current condition presented by the 10 inch RHR piping and performed a preliminary analysis which determined that the other identified deficiencies did not invalidate the classification of the CCW piping as a closed system inside containment. The licensee generated CATS item M06310 to track the revision of the analyses.

As discussed above, ACTION Report 97-1235 performed an operability assessment of the current configuration and determined that it is operable. A fracture mechanics evaluation of the 10" RHR piping has been completed and submitted to the NRC for review and approval. DA-ME-95-088 will be updated to reflect this information within 6 months of the approval of the submittal.

4. *NRC Item 50-244/97-201-04 (URI) Calculation Control*

The team reviewed the licensee's disposition of three AR's concerning the CCW system. The team identified a concern with one disposition. AR 96-0376, "Evaluation of Valve 815A's Seismic Qualification Needed Before Startup," evaluated the effect of a weight change for valve 815A on stress analysis SDTAR-80-05-88 as acceptable. The isometric drawing of the piping containing this valve, C-381-356, sheet 5, was updated with the new valve weight but the stress analysis was not updated nor identified as having an unincorporated change. The licensee did not have a procedure requiring the analysis to be updated. The team was concerned that, if this stress analysis was revised in the future for another piping system change, the revised weight of valve 815A might not be incorporated in the analysis and thus cause an incorrect result. The licensee stated that this discrepancy would be addressed in the resolution of AR 97-1149, "Change Process Controls do not Adequately Address Effects on Calculations."

The team also reviewed fourteen SI and RHR system uncertainty calculations along with pressurizer pressure and steam generator pressure uncertainty calculations. Uncertainty calculations DA-EE-92-035-21, "Calibration of Refueling Water Storage Tank Level Loops 920," Revision 0, and DA-EE-92-036-21, "Calibration of Refueling Water Storage Tank Level Loops 921," Revision 0, determined the overall loop uncertainty associated with level channels LVL 920 and LVL 921. UFSAR Tale 7.5-1 compared the loop characteristics with RG 1.97, Revision 3 criteria. The team found a discrepancy between calculation DA-EE-92-035-21 and UFSAR Table 7.5-1 with regards to the referenced RG 1.97 category. The calculation stated that the instrument loop was an NRC Category 2, Type A variable. Category 2 variables as defined in RG 1.97 did not include seismic qualification, redundancy, or continuous display and required only a highly reliable power source (not necessarily standby power). This was inconsistent with UFSAR Table 7.5-1 which stated this loop was a Category 1, Type A variable. The team observed that it should have been classified as a Category 1 variable requiring redundancy, continuous real-time display, Class 1E power, and seismic qualification. In response to the team's question, the licensee made the appropriate changes to the calculation. The team also found that calculations DA-EE-92-035-21 and DA-EE-92-036-21 referenced an incorrect RWST boron concentration range of 2300 to 2448 ppm. This was inconsistent with UFSAR Table 6.3-4 and the TS, which documented the boron concentration range as 2300 to 2600 ppm. The team was concerned that using the incorrect boron concentration in the calculation for specific gravity could have an adverse affect on the loop performance evaluation. The total loop uncertainty was re-analyzed by the licensee using a boron concentration of 2300 to 2600 ppm. the results were found to have minimal impact on the instrument uncertainty. Based on the team's finding, the licensee revised both calculations. It appeared that the calculations were not maintained current as required by 10CFR50, Appendix B, Criterion III, "Design Control".

An evaluation of Ginna Station's "Calculation Control" process is being reviewed under our 50.54(f)/DBD efforts. Various procedure control processes are being evaluated as well as the use of Information Technology to provide the electronic linking of design basis calculations. In addition, various enhancements to the modification process have recently been performed to ensure design basis analyses are maintained current during the modification process. Enhancements to the calculation control process will be a continuous improvement effort with completion of the 50.54(f) Project efforts associated with calculation control scheduled for 12/31/99.

5. *NRC Item 50-244/97-201-05 (IFI) Cable Ampacity*

The team reviewed calculations DA-EE-96-068-03, "Offsite Power Load Flow Study," Revision 0; DA-EE-96-098-03, "AC Electrical System Fault Current Analysis," Revision 0; DA-EE-92-098-01, "Diesel Generator A Steady State Loading Analysis," Revision 1; DA-EE-92-120-01, "Diesel Generator B Steady State Loading Analysis," Revision 1; DA-EE-92-111-01, "Diesel Generator A Dynamic Loading Analysis," Revision 0; and DA-EE-92-011-07, "Class 1E Motor Control Center Loading," Revision 4. The team verified that all major CCW electrical loads were accounted for in these calculations for both normal and accident conditions and that the motors were sized to accelerate the CCW pumps and to drive them for long-term continuous operation. The team determined that the methodology and assumptions used were appropriate.

The team also reviewed calculation DA-EE-93-104-07, "480 Volt DB Breaker with Amptector Retrofit Coordination and Circuit Protection Study," Revision 1. The team determined that the overcurrent protection for the CCW electrical loads and their cables conformed to industry standards. The team noted that ampacity derating calculations did not exit for the 480V CCW pump feeder cables. The licensee performed calculations during the inspection that verified that the cables were adequately derated for their raceway routings. Electrical Design Guide EDG-4A, "Cable Sizing Analysis for Cables Installed in Conduit and Cable Trays," Revision 0, was reviewed and found to be consistent with industry standards ICEA P-54-440, "Ampacities of Cables in Open-Top Cable Trays," and IEEE S-135 (IPCEA Publication P-46-426), "Power Cable Ampacities," except that cables routed through duct lines and specific ampacity derating data for HEYMC fire wrap were not addressed. Calculations were performed by the licensee during the inspection to show that the diesel generator cables routed through duct lines were adequately sized and derated for their installation. These calculations demonstrated that the power cables for the CCW pump motors and the diesel generator feeder cables were capable of performing their electrical function. The licensee stated that the ampacity of other power cables would be evaluated under EWR-5298, "Cable Ampacities", which would refine EDG-4A, evaluate the ampacity of the 480V and 4KV cables, and review specific

low voltage AC and 125V DC power circuits. AR 97-1221 was initiated for fire wrap derating concerns.

The existing ampacity analysis for cables wrapped with HEMYC wrap is currently being revised and is scheduled to be completed by 6/30/98. In conjunction with this effort, Electrical Design Guideline EDG-4A, "Cable Sizing Analysis for Cables Installed in Conduit and Cable Trays", is being replaced with a design analysis that specifies specific criteria for evaluating cable ampacity for the various cable installation configurations such as but not excluded to conduits, cable trays, duct banks, effects of fire stops and fire wrap. This effort is scheduled to be completed by 11/30/98. The evaluation of the ampacity of power cables being performed under EWR-5298 requires an extensive review of the routing of all power cables. The review of cable routings is being performed in conjunction with our Fire PSA effort. Following the review of the routing of all power cables the ampacity of the cables will be evaluation. This effort is scheduled to be completed by 12/31/99.

6. *NRC Item 50-244/97-201-06 (IFI) Electrical Calculation Discrepancies*

The sizing and testing of the DC battery system was reviewed by the team to ensure that adequate battery capacity was available to support the loads during accident conditions. The team noted that EEA 09004, "Sizing of Vital Batteries," Revision 0, did not take into account the TS surveillance requirement, SR 3.8.6.5, which required a battery temperature of $\geq$55°F; did not take into account the manufacturer's ampacity reduction due to an effect known as Coup de Fouet; and did not verify the utilization of battery capacity factors (Kt) against manufacturer's data. The calculation stated that the analysis would be revised yearly or if load changes occurred to affect the design margin, the calculation had not been revised since April 25, 1991. The licensee issued calculation DA-EE-97-069, "Sizing of Vital Batteries A and B," Revision 0, during the inspection which adjusted the electrical loads, reduced the battery operating temperature to 55°F, accounted for the manufacturer's revised on e minute discharge rates, and provided a design margin for future load increases. Review of calculation DA-EE-97-069 by the team verified that the A and B station batteries were adequately sized to perform their design function under normal and accident conditions. The licensee stated that, as a result of this revised calculation, other design analyses and UFSAR Table 8.3-5 required revision, the Station Blackout Program required updating, and the battery testing procedures required updating. The licensee issued Corrective Action Tracking System (CATS) items M06273, M06275, and M8672, respectively, to track these revisions and updates. Additionally the licensee had initiated EWR-10360 prior to the inspection which would upgrade the DC load study, DC voltage regulation fuse sizing, and coordination analyses.

EWR-10360 was initiated to address the revision of the battery sizing analysis. This effort has been completed and the EWR closed. Under TSR 97-190, a review of the existing electrical DC system analyses such as voltage regulation, fuse sizing, and coordination will be performed to ensure the design basis for the DC system is adequately documented. Existing analyses will be revised and/or developed to support this review. The completion of this effort is scheduled for 9/30/99.

7. *NRC Item 50-244/97-201-07 (IFI) TS Discrepency*

The following battery testing and maintenance procedures were reviewed by the team:

* **PT-10.2, "Station Battery B Service Test," Revision 20, completed on 5/20/96;**
* **PT-10.2, "Station Battery B Service Test," Revision 19, completed on 4/6/95;**
* **PT-10.3, "Station Battery A Service Test," Revision 24, completed on 5/20/96;**
* **PT-10.3, "Station Battery A Service Test," Revision 23, completed on 4/5/95;**
* **PT-10.4, "A Station Battery Performance Test," Revision 11, completed on 4/8/92;**
* **PT-10.5, "1B Station Battery Performance Test,"Revision 10, completed on 4/26/93; and**
* **PT-11, "60 Cell Battery Banks "A" & "B" and Spare Cells", Revision 36.**

The team verified that the batteries were tested to the load profile as presented in design analysis EEA 09004 and that the batteries met both their performance and service duty cycle requirements. It was noted that the A battery had not been tested within the 60 month interval required by TS surveillance requirement SR 3.8.4.3. The licensee stated that TS SR 3.0.2 allowed a testing extension of 25 percent and that the battery performance testing had been scheduled to be performed in the 1997 fall outage.

Vendor Manual VTD-G185-4001, "GNB Station Battery Installation and Operating Instructions," stated that the battery specific gravity may vary +/- .010 points. The normal specific gravity of a fully charged battery was 1.215 at 77°F. Maintenance procedure PT-11 did incorporate the manufacturer's requirement of +/- .010 points; however, TS surveillance requirement SR 3.8.6.6 allowed a less conservative value of +/- .020 points. The licensee initiated AR 97-1170 to resolve this discrepancy.

The TS Surveillance SR 3.8.6.6 values of $\pm$ .020 points are taken directly from NUREG-1431, Standard Technical Specifications for Westinghouse Plants, and cannot be changed during the ITS conversion process. There are allowances for individual battery cells and the limits at which the battery bank is declared inoperable. This surveillance was approved by the NRC as part of the ITS conversion process. RG&E will evaluate providing additional wording to the Bases for SR 3.8.6.6 to discuss the manufacturer's recommendations regarding the $\pm$ .010 point allowance.

8. *NRC Item 50-244/97-201-08 (IFI) Battery Rack Configuration*
The team performed a walkdown of the CCW system. The team verified that nameplate loadings were used in the plant design analyses. The auxiliary building area containing the system was clean and the raceway systems were properly identified.

During the walkdown of the A and B battery rooms, the team noted that spacers were installed between the battery cells, however, drawing 33013-1120, "Battery Room Racks Seismic Battery Restraint," Revision 6, did not reflect the installation of these spacers. The licensee determined that the installed condition was seismically acceptable. The A battery rack also had metal standoffs that come into contact with the battery room wall which were not shown on drawing 33013-1120. The licensee previously recognized that these structural members existed and they had been evaluated as acceptable under the Seismic Qualification Utility Group (SQUG) program; however, drawing 33013-1120 was not updated to reflect this condition. The licensee issued AR 97-1170 to address the spacers and initiated Plant Change Record (PCR) 97-038 to modify the existing spacers and update the battery rack drawings.

PCR 97-038, "A&B Battery Rack Upgrade", was completed during the 1997 outage. Closeout of this modification will correct the identified concerns. Closeout is scheduled to be completed by January 31, 1998.

9. *NRC Item 50-244/97-201-09 (IFI) Battery Rack Grounding*
The team noted that the B battery rack had no visible ground connection. This was previously identified in Safety System Functional Inspection 89-81 as a concern. The licensee had previously evaluated this installation, determined that the rack was actually grounded, and recommended that a visible ground be installed. The visible ground connection was not installed and the item was closed. The licensee stated that a visible ground would be installed as recommended.

A visible ground was installed on the battery rack as part of PCR 97-038, "A&B Battery Rack Upgrade", during the 1997 Refueling Outage. Therefore, this item is complete.

10. *NRC Item 50-244/97-201-10 (URI) Relief Valve Design Basis*
During the team's walkdown of the CCW system, various instrument configurations were inspected. The team identified a concern with modification EWR 10037, "Evaluation of Setpoint Adjustment for Relief Valve 10020," Revision 0. The team observed that the modification had not been implemented. Relief valve RV 10020 was installed to prevent an overpressure condition in the shell side of the post-accident sampling system (PASS) coolers. The PASS coolers were cooled by CCW. The setpoint for RV 10020 was recommended to be changed from 200 psig to 150 psig to match the design pressure of the shell side of the PASS coolers and of the relief valve body. The EWR recommended that the relief valve setpoint be re-adjusted to its proper lifting pressure. An inter-office correspondence dated September 13, 1993, to the Technical Engineering Manager recommended that a technical staff request (TSR) should be initiated. The licensee stated that the TSR was never formalized to change the set pressure from 200 psig to 150 psig. The team reviewed the relief valve setpoint and verified it was still listed in the Equipment File Maintenance Program as 200 psig. The team inspected the calibration data of the relief valve and it was also set at 200 psig.

The licensee stated that a similar situation as existed for RV 10020 would not be likely to occur now as the plant change process has been revised. The team sampled on additional relief valve setpoint modification, EWR 10107, "RV 1817 & RV 861 Setpoint Change," Revision 0. RV 861 functions to provide overpressure protection for the piping and components on the suction side of the CS pumps. RV 1817 functions to provide overpressure protection for the piping and components on the suction side of the SI pumps. Both setpoint changes were performed based on a recommendation from Mechanical Engineering.

The team reviewed the EWR documents, TSR 93-072, "Relief Valves 861 and 1817 Setpoint Change," Revision 0, and Design Analysis DA-ME-92-099, "Relief Valve 861 and 1817 Setpoint Evaluation," Revision 0. The team verified that the implementing documents were appropriate, the change was implemented correctly, and the data sheets were consistent with the design.

The licensee issued AR 97-1203 to evaluate this item. The design basis for RV 10020 was not implemented in the plant installation as required by 10CFR50, Appendix B, Criterion III, "Design Control".

The operability of the CCW system with a setpoint of 200 psig has been previously documented. Setting of the valve to 150 psig is considered more appropriate, but requires installation of a new valve. PCR 97-084 has been initiated to accomplish this with a targeted completion of June 30, 1998.

11. *NRC Item 50-244/97-201-11 (IFI) SI Transfer Procedure*
Procedure ES-1.3, "Transfer to Cold Leg Injection," Revision 20, provided the operating instructions for transferring the SI system and containment spray (CS) system to the recirculation mode of operation. This procedure would be entered from various other procedures, or whenever the RWST level reached the low level setpoint of 28 percent under post-accident conditions. The procedure contained various steps that would have to be completed during the RHR pump suction transfer from the RWST to the containment sump in a limited time, which could be as little as 8.5 minutes. The note on page 3 of Procedure ES-1.3 stated that steps 2 through 12 of the procedure should be performed without delay.

Westinghouse report FSD/SS-M-2083, "Ginna Nuclear Station Switchover to Recirculation," Revision 1, addressed both the operator time allowances to transfer the SI system to the recirculation mode and the SI flow required to keep the core covered during transition. This evaluation (Figure IIIA) determined that the four operator actions required to transfer the RHR pump suction from the RWST to the containment sump could be completed in less than 8 minutes. The four actions addressed in the report were:

* Stop two RHR pumps, one SI pump, and one CS pump;
* Close motor operated valves (MOV) 704A & B;
* Open MOVs 850A & B; and
* Individually start two RHR pumps.

The Westinghouse evaluation determined that, based on maximum safeguards pump flows, the shortest time to pump the RWST down from the low level alarm setpoint of 28 percent to the low-low level alarm setpoint of 15 percent would be 8.5 minutes during a large break LOCA. Therefore, to avoid a complete interruption of SI flow during the transfer, the operators would have to complete the required actions in less than 8.5 minutes.

In addition to the operator actions addressed in the Westinghouse evaluation, steps 2 through 12 of procedure ES-1.3 contained several additional actions. These actions included verification that at least two SW pumps were running, dispatching an auxiliary operator to verify SW flow to the CCW heat exchangers, and dispatching an auxiliary operator to manually adjust RHR flow if the air-operated control valves were not available. The team asked the licensee to verify that steps 2 through 12 of procedure ES-1.3 (including CCW and SW system realignments) would be performed prior to reaching the RWST low level setpoint and entering ES-1.3. This direction was not included in the Emergency

Operating Procedures (EOP), but was included in operator training and had been demonstrated during simulator exercises. The licensee stated that a note would be added to procedure E-1, "Loss of Reactor or Secondary Coolant," Revision 14, and to the EOP users' guide, A-503.1, recommending early entry into procedure ES-1.3 when transfer to cold leg recirculation was imminent, provided the injection flowrates were not altered until the RWST level reached the switchover setpoint. Operations Change/Clarification Form 97-73 was initiated to change these documents. Based on discussions with the licensee and observing a simulator training exercise involving this transfer scenario, the team concluded that the proposed procedure change would resolve this concern.

Changes to procedures E-1 and A-503.1 to add a recommendation for early entry into ES1.3 were made during the A/E Inspection. Therefore, this item is complete.

12. *NRC Item 50-244/97-201-12 (IFI) RHR Pump NPSH*
The team reviewed the available licensing, design, and operations documents related to the required and available Net Positive Suction Head (NPSH) of the SI and RHR pumps operating under accident conditions for both the injection phase of the SI system from the RWST and the recirculation phase of the SI system from the containment building sump. Gilbert Associates Report Number 428-4824-027-2R, Revision 0, dated March 11, 1982, addressed the required and available NPSH during the injection mode of SI operation, and determined that the SI and RHR pumps would have adequate NPSH.

The RHR pump NPSH available from the containment sump during recirculation was calculated by Design Analysis NSL-0000-DA027, "Residual Heat Removal Pump NPSH Calculations During Accident Conditions," Revision 0. This calculation used a minimum post-accident sump water level at switchover of 4 feet above the containment floor. This water level was determined by an informal calculation in 1982 as part of the evaluation of SEP topic VI-7B, Sump Switchover. Prior to the inspection, the licensee determined that a formal calculation was required to verify and document the minimum containment sump level under post-accident conditions. Design Analysis DA-NS-97-065, "Post-LOCA Sump "B" Level," Revision 0, was issued on July 7, 1997. This new calculation determined that the minimum post-accident sump level would be 2.78 feet above the containment floor, as opposed to the value of 4 feet above the containment floor used in the NPSH calculation. The licensee initiated AR 97-1167 to evaluate the impact of the reduction in calculated sump level on the RHR pump NPSH and performed an operability assessment to evaluate the RHR pumps.

The operability assessment addressed the containment sump level inconsistency in the RHR pump NPSH analysis as well as a discrepancy identified by the licensee

in the limiting RHR flowrate to be considered during one pump operation. The assessment determined that in the limiting condition ("A" pump operation with only one containment sump flow path available due to the single failure of MOV 850A or 850B to open) a NPSH deficit of approximately 0.6 feet would exist, based on saturated water conditions in containment sump B. The licensee determined that these conditions would not occur until several hours after the accident, and that a NPSH deficit would not be expected to occur if credit were taken for subcooling of the containment sump water based on the calculated containment pressure and temperature profiles. Furthermore, the licensee received confirmation from the pump vendor that operation with the calculated NPSH deficit was acceptable. The assessment, therefore, concluded that the RHR pumps were operable and had adequate NPSH to provide the required long term cooling function required of the ECCS system under the limiting condition and utilizing the current procedure guidance.

The licensee also issued and approved Safety Evaluation SEV-1101, "Alignment of MOV-857A, 857B, and 857C During Sump Recirculation in ES-1.3," during the inspection. This Safety Evaluation supported a change to Step 11e of Procedure ES-1.3 so that if only one RHR pump were operating, only the associated discharge valve(s), 857A and 857C (Train A) or 857B (Train B), would be opened. The licensee stated that this procedure change would limit RHR flow under the limiting post-accident conditions and eliminate the calculated NPSH deficit. The licensee also stated that Design Analysis NSL-0000-DA027 would be revised to incorporate the correct minimum water level.

As discussed above, all required procedure changes were made during the A/E Inspection and are therefore complete. A revision to design analysis incorporating these changes, NSL-0000-DA027 is scheduled to be completed by June 30, 1998.

13. *NRC Item 50-244/97-201-13 (IFI) Auxiliary Building Post-Accident Environment* UFSAR section 9.4.9.1 states that the Engineered Safety Features (ESF) Ventilation system is not required for the operation of the SI and RHR pumps and section 9.4.9.4 states the capabilities of the CCW pumps will not be exceeded if the auxiliary building air handling unit are inoperable (this unit is nonsafety-related). The basis for these assumptions was contained in two analyses, one that calculated the maximum expected post LOCA ambient temperatures in the auxiliary building, and one that calculated the corresponding qualified life of the effected equipment.

The calculation of ambient temperature was contained in "Engineering Evaluation of R.E. Ginna Nuclear Power Plant Ventilation System," Revision 1, which determined the thermal environment in which the SI, RHR, and CCW pumps and other safety-related equipment must operate after a LOCA assuming no ESF

cooling. The team identified that this evaluation contained several non-conservative assumptions: first, an initial auxiliary building temperature of 85 degrees F was used instead of the maximum design basis temperature of 104 degrees F listed in UFSAR Table 3.11-1; second, a water temperature of 80 degrees F for the RWST was used instead of 104 degrees F, which corresponded to the maximum design basis temperature in the auxiliary building in which the RWST was located; and third, the evaluation did not consider the effect of the design basis 50 gpm seal leak from a RHR pump at the sump water temperature of 155 degrees F used in the evaluation. The team also noted that the evaluation contained conservative assumptions to simplify the analysis. These assumptions included use of a 75 degrees F ground temperature, no mixing between the east and west portions of the bottom floor of the auxiliary building, and not considering piping colder than the atmosphere as a heat sink. The licensee had not quantified the effect of these assumptions on the analysis.

The environmental qualification for the RHR pump motors was contained in EWR-4237.30, "Qualified Life Calculation for RHR Pump Motor S/O 67C68831, S/N 1," Revision 1. The EWR determined that the RHR pump and motor had a qualified life of approximately 28 years utilizing a 120 degrees F normal ambient and a 149 degrees F post-LOCA ambient for 200 days. The team also reviewed EWR-4991-EQ1, "Verification of Environmental Qualification of the Rewound RHR Pump 1B Motor (S/O 67C68831, S/N 2)," Revision 0, which determined that the rewound ambient temperature of 104 degrees F and a post-LOCA ambient of 160 degrees F for 40 hours followed by 134 degrees F for the rest of the assumed 1 year post-LOCA period. This calculation was not revised to include the results of the Devonrue evaluation; however, the team observed that the long qualified life calculated for pump 1B motor appeared to provide adequate margin for a post-LOCA temperature of 149 degrees F for 200 days.

As a result of the team's questions concerning the non-conservatisms in the ambient temperature analysis, the licensee performed EWR 4237.30, "Qualified Life Calculation for RHR Pump Motor S/O 67C68831, S/N," Revision 2, during the inspection which documented the acceptability of the RHR pump A motor in a post-LOCA ambient temperature of 195 degrees F, which was a temperature that would be considered high enough to envelope the concerns raised with the non-conservatisms in the original analysis. The results of this analysis indicated that the RHR pump motors would still have a sufficient qualified life even at the higher ambient temperatures. The licensee also issued AR 97-1226 to evaluate any other effects of the discrepancies in the analysis.

Similar concerns were also raised for the SI pumps and motors. Assuming that the post LOCA ambient temperature increased by the 19 degrees F difference between a starting ambient of 85 degrees F and one of 104 degrees F, a post-LOCA ambient temperature of 127 degrees F would be calculated. The team

determined that the qualified life under this condition would still be on the order of 400 years and that the SI pump motors were therefore qualified for their design basis service conditions.

The CCW pump motors were not evaluated for qualification in a harsh environment as the ambient temperature analysis for the corresponding area had calculated a post-LOCA temperature less than the 40 degrees C motor design basis. However, if the maximum design building ambient of 104 degrees F were used as an initial condition, the calculated post-LOCA temperature might subject the CCW pump motors to temperatures in excess of design.

The licensee was in the process of evaluating whether the CCW pump motors or any other associated equipment would require qualification as a result of the non-conservatisms identified by the team with the ambient temperature analysis.

Based on preliminary calculations, RG&E is confident that the Auxiliary Building will remain a mild environment in accordance with the definition in 10CFR50.49, even with initial conditions of 104°F. RG&E is in the process of building a computer model of the Auxiliary Building to reassess the post accident environment in that building. This effort is scheduled to be completed by 12/31/98.

14. *NRC Item 50-244/97-201-14 (IFI) Relay Cracking*

The team reviewed the diesel generator steady state loading analyses, DA-EE-92-098-01, "Diesel Generator A Steady State Loading Analysis," Revision 1, and DA-EE-92-120-01, "Diesel Generator B Steady State Loading Analysis," Revision 1, and noted that these calculations considered a injection phase operation with the SI pumps aligned to draw from the Boric Acid Storage Tank (BAST). The team noted, however, that the SI pumps are no longer aligned to draw from the BAST. The licensee stated that the alignment in the calculations was incorrect, demonstrated to the team that the actual alignment to the BAST was conservative with respect to diesel loading, and initiated CATS item M06243 to track the analysis revision to correct this discrepancy.

The team reviewed the licensee's evaluation of IN 91-45, Supplement 1, "Possible Malfunction of Westinghouse ARD, BFD, and NBFD Relays, and A200 DC and DPC250 Magnetic Contactors," July 1994. The IN alerted licensees of cracking of the relay housing and potential relay malfunction. The licensee's analysis included a combination of (1) inspection of the installed components; (2) testing of spare relays, relay coils, and starter coils; and (3) addition of steps in procedure M-1306.2, "Periodic Cleaning/Inspection of Relay Cabinets and Related Electrical Components," which required periodic inspection of relays installed in cabinets for degradation. The licensee stated that the latest inspection, which was performed during the 1996 outage, did not identify any cracked relays. During

the inspection, the licensee initiated a walkdown of relay racks and identified 3 relays exhibiting cracked coil cases. The team reviewed procedure M-1306.2, Revision 12, and noted that step 5.4.2 only required that NBFD relays be inspected. The license stated that the procedure would be updated to inspect for the BFD style relay, which was the only other type noted in the IN used by the licensee. The licensee determined that the observed cracking would not degrade the relay function and issued AR 97-1147 to resolve this issue.

Procedure M-1306.2 was revised during the A/E Inspection to include inspection of the BFD relays. Therefore, this item is complete.

15. *NRC Item 50-244/97-201-15 (URI) Control and Review of Accident Analysis*

a. The computer models used to demonstrate acceptable plant response to a postulated Large or Small Break LOCA were run by Westinghouse, using input parameters specific to the Ginna Plant. Some input parameters were provided by Westinghouse, while others were provided by RG&E. The team learned that prior to running the analyses, Westinghouse transmitted a partial set of input data to RG&E for their review and approval. For some parameters, such as the accumulator water and gas temperature, the total axial offset at 100% power, and the parameters related to the new steam generators, Westinghouse provided no data and left it up to RG&E to provide the correct parameters.

The team identified that this process of review and approval of input data had not been specifically proceduralized at RG&E and that key attributes such as independent review, data traceability, and data control were sometimes lacking. Consequently, errors have occurred, most of which were identified by Westinghouse either before or after the analyses were run. For example, RG&E failed to identify that the value for accumulator water volume initially supplied by Westinghouse was in error. Subsequently, after running the analysis, Westinghouse identified this error and reported it to the NRC. In another instance, RG&E provided Westinghouse with an incorrect value for the accumulator total tank volume, but Westinghouse identified the error and used the correct value in the analyses. Also, RG&E provided Westinghouse with a non-conservative value for the refueling water storage tank water temperature. Westinghouse ignored this value and used a conservative value for the analyses.

The team learned that the highest value for the accumulator water temperature had been assessed by RG&E to be 115°F (RG&E Calculation Note of July 13, 1994). This value had initially been transmitted to Westinghouse (RG&E letter W-94-15, July 13, 1994) for use as an input to the LOCA analyses calculations. The resulting large break LOCA peak clad temperature (PCT) exceeded the 10CFR50.46 PCT criterion of 2200°F. Hence, RG&E requested Westinghouse to provide guidance for determining a lower accumulator water temperature. Westinghouse recommended *computing a "maximum expected" value for the*

accumulator water temperature based on the highest two week average of containment air temperature in the vicinity, to subtract 4°F from the highest two week average and to use that temperature as the value for accumulator water temperature (Westinghouse letter NTD-NSRLA-OPL-95-110, March 8, 1995). RG&E determined that the highest two week average was 108.5°F (based on 1993 and 1994 measurements) and consequently provided Westinghouse with a new accumulator water temperature of 105°F to be used in the large break LOCA analysis (RG&E letter W-95-08, March 8, 1995). The use of this new value resulted in a Large Break LOCA calculated PCT less than 2200°F. The team questioned the accumulator water temperature of 105°F due to the fact that the operating containment design temperature was 120°F and that RG&E had not established a correlation between the containment temperature and the accumulator water temperature. In order to respond to the team concerns, RG&E performed temperature measurements during the inspection both in the vicinity of the accumulators and within the accumulators. These measurements provided reasonable assurance that the accumulator water temperature will not exceed 105°F, even if the containment atmosphere reaches its 120°F design limit.

The team also noted that an independent engineering review by RG&E of the document used to transmit the data inputs to Westinghouse had not been performed. Many of the RG&E calculation notes supporting these inputs were also lacking independent verification (i.e. Calculation Note #3, Rev. 0, "Auxiliary Feedwater Purge Volume," 6/22/94; Calculation Note #2, Rev. 0, "Pressurizer Water Volume," 4/21/93; Calculation Note #4, Rev. 0, "Accumulator Discharge Line Volume," 6/30/94). RG&E has, however, implemented an initiative to assimilate the accident analyses inputs into a single database which should allow for better control of the inputs.

b. In accordance with the approved methodology, prior to running the actual LOCA transient, it is first necessary to model the proper steady-state conditions. Accordingly, acceptance criteria were developed for key output parameters in order to demonstrate acceptable steady-state performance of the computer model. The team identified that the parameter for core inlet temperature was outside of the pre-established acceptance band by -.75%; however, no explanation of this anomaly was given. Based upon the accident analysis methodology, this negative difference put the calculated Core Inlet Temperature in an unacceptable range, as the methodology specifies that the steady-state calculated value for the core inlet temperature must be greater than the desired value. Westinghouse and RG&E failed to identify this unacceptable steady-state condition during their respective review of the completed Large Break LOCA analysis report. Westinghouse reviewed the error during the course of the inspection and concluded that it did not change the reported licensing basis peak clad temperature for the Large Break LOCA. The team concurred with Westinghouse's conclusion.

As stated previously, after an unsuccessful computer run with the accumulator temperature at 115°F, the accumulator temperature was lowered to 105°F, and a successful run (i.e. peak clad temperature less than 2200°F) was achieved. Westinghouse did not, however, re-calculate a new containment back pressure transient, which is an input to the WCOBRA/TRAC code. Since accumulator temperature is also an input to the containment pressure transient COCO computer code, a new transient should have been generated. When re-running WCOBRA/TRAC for the 105°F case, Westinghouse used the containment back pressure transient calculated with the previous value of 115°F. Westinghouse did not change the reported licensing basis peak clad temperature for the Large Break LOCA. The team concurred with Westinghouse's conclusion.

The team also identified several errors in the Large Break LOCA engineering report (WCAP-11427, May 1995) that had not been previously identified by either Westinghouse or RG&E. For example:

(1) in Table 4-5 the calculated PCT for the low Tave is indicated to be 2006°F while the correct value is 2050.5°F;

(2) the Core outlet temperature given in P.59 of the report is indicated to be 590.58°F while the correct value is 594.62°F;

(3) Upper head temperatures given in Table 4-1 are incorrect; and

(4) 714.7 psia is given in P.50 of the report for the accumulator nitrogen pressure while the correct value is 714.5 psia.

The above-mentioned errors and inconsistencies indicate that sufficient reviews may not have been performed of the completed LOCA analyses reports and associated supporting documents (calculation notes).

The above referenced reports were prepared by Westinghouse as an approved vendor for quality related analytical services. As a result of the noted discrepancies, Westinghouse has undertaken an evaluation to determine the cause of these errors and proposed corrective actions. This evaluation is expected to be completed by December 31, 1997. RG&E is in the process of developing an Engineering Guideline to provide controls for preparation, review and approval of analytical inputs to vendors. This guideline is scheduled to be completed by March 31, 1998.

16. *NRC Item 50-244/97-201-16 (URI) UFSAR Discrepancies*

The team reviewed the Emergency Core Cooling system (ECCS) evaluation reports submitted by RG&E to the NRC between 1992 and 1997 and their supporting documents. The team found that by letter RGE-96-204 of February 9, 1996, Westinghouse had notified RG&E of emergency core cooling system (ECCS) evaluation model errors and changes that affected the licensing basis PCTs of the Ginna plant for the 1995 year. The new values for both the small and the large break LOCA analysis results were 1313°F and 2099° respectively. In accordance with 10CFR50.46, paragraph (a) (3) (ii), RG&E notified the NRC of the changes (RG&E letter to the NRC, July 8, 1996). However, RG&E failed to update the Ginna UFSAR. The December 1996 version of the UFSAR still mentions the old Small Break and Large Break LOCA PCT values of 1308°F and 2051°F. RG&E initiated corrective actions during the inspection to correct the FSAR and to systematically ensure a timely update of the FSAR after notification of changes to the calculated PCT. The team reviewed these corrective actions and considered them appropriate.

The team identified the following discrepancies in the UFSAR:

* **Section 9.2.2.4.3 incorrectly stated that the portion of the CCW loop outside the containment was considered to be part of the containment isolation barrier. TS B 3.7.7 correctly stated that the CCW system outside containment was not required to be a closed system.**

* **Section 9.2.2.2 stated that radiation monitor RM-17 was in the component cooling pump inlet header instead of the discharge piping as installed.**

* **Section 9.2.2.2 stated that one CCW heat exchanger accommodated the loads during normal full-power operation whereas two heat exchangers were actually in use. The licensee issued AR 97-1222 to clarify the UFSAR.**

* **Penetration 124c was missing from UFSAR Table 6.2-15. The licensee issued AR 97-1179 to correct the UFSAR.**

* **Table 7.5-1 incorrectly specified a range of 0-1000 gpm for SI flow instruments FT-924 and 925. The correct range as installed was 0-600 gpm. The licensee issued AR 97-1125 to correct the UFSAR.**

* **Table 15.1-6, Case 2, listed times of 13.3 seconds for the SI pumps to start and 25.3 seconds for the SI system to reach full flow instead of the correct values of 14.3 and 26.3 seconds, respectively. The licensee issued CATS item M06306 to correct the UFSAR.**

* **Section 8.3.2.2 stated that the normal battery operating voltage was 132V instead of 130V. The licensee issued AR 97-1191 to correct the UFSAR.**

* **Section 8.3.2.3 stated that all branch fuses must carry worst-case credible loads without interruption of service under accident conditions, and defined the worst-case credible loads as the sum of all class 1E components within a load group; i.e., all components fed by a branch fuse were assumed to be operating at the same time. This was not the case as the fuses were not sized for all loads operating at the same time. The licensee issued AR 97-1191 to correct the UFSAR.**

* **Section 8.3.2.3 stated that the main and branch fuses used in the DC distribution system must have a minimum DC rating of 140 V; however, some fuses were only rated 125V DC. The licensee issued AR 97-1191 to correct the UFSAR.**

* **Section 8.3.1.4 stated that cable trays were filled greater than 100 percent only where control cable trays intersected; however, trays exceeded 100 percent fill in other instances. The licensee issued AR 97-1220 to resolve this discrepancy.**

The above discrepancies had not been corrected and the UFSAR updated to ensure that the information included in the UFSAR contained the latest material as required by 10CFR50.71(e). The licensee issued ARs and CATS items to correct some of the above discrepancies and stated the others would also be corrected.

RG&E will ensure that all noted discrepancies will be corrected in the next scheduled UFSAR revision to be issued on or before May 21, 1998.