November 2, 2007

Mr. John T. Carlin
Vice President, R. E. Ginna Nuclear Power Plant
R. E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT - NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000244/2007006

Dear Mr. Carlin:

On September 20, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the R. E. Ginna Nuclear Power Plant. The enclosed inspection report documents the results of the inspection, which were discussed on September 20, 2007, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed Constellation's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents three NRC-identified findings which were of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the violations and because they were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any of the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the R. E. Ginna Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-244
License No. DPR-18

Enclosure: Inspection Report 05000244/2007006

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REGION I

Docket No.	50-244
License No.	DPR-18
Report No.	05000244/2007006
Licensee:	R. E. Ginna Nuclear Power Plant, LLC
Facility:	R. E. Ginna Nuclear Power Plant
Location:	Ontario, New York
Dates:	August 13 to September 20, 2007
Inspectors:	F. Arner, Senior Reactor Inspector (Team Leader) J. Richmond, Senior Reactor Inspector A. Patel, Reactor Inspector T. Sicola, Reactor Inspector G. Skinner, NRC Electrical Contractor W. Sherbin, NRC Mechanical Contractor
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000244/2007006; 08/13/2007 - 09/20/2007; R.E. Ginna Nuclear Power Plant; Component Design Bases Inspection.

The report covers the Component Design Bases Inspection conducted by a team of four NRC inspectors and two NRC contractors. Three findings of very low safety significance (Green) were identified, and considered to be non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, in that, Constellation had not provided adequate design control measures to verify the adequacy of Residual Heat Removal (RHR) pump net positive suction head (NPSH) margin for the containment sump recirculation mode of operation. Specifically, under certain loss-of-coolant accident (LOCA) conditions, the team determined that additional RHR pump flowpaths would exist through idle containment spray and safety injection pumps which had not been previously evaluated for in the design basis NPSH analysis of record. Constellation entered the issue into their corrective action program and revised the emergency operating procedures to ensure consistency between the implementing procedure and the design analysis.

  The finding was more than minor because the deficient NPSH analysis resulted in a condition where there was reasonable doubt with respect to the operability of the RHR pumps. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that did not result in a loss of RHR pump operability. (Section 1R21.2.1.2)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the maximum expected differential pressure established as a design input in the containment sump suction valve (MOV 850A/B) thrust margin evaluation, had not been verified to be a conservative value during the recirculation phase of operation for a small break loss-of-coolant accident (SBLOCA). Constellation entered the issue into their corrective action program and performed a detailed engineering evaluation to ensure valve thrust margin remained for SBLOCA scenarios. Additionally, Constellation revised the emergency operating procedures to ensure that potential pressurization of the residual heat removal system is monitored and depressurization performed prior to initial opening of the containment sump suction valves.

The finding was more than minor because the design analysis deficiency resulted in a condition where there was reasonable doubt regarding the operability of the containment sump valves. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, the team conducted a Phase I SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that did not result in a loss of operability.

The finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program. Constellation had not taken appropriate corrective actions to address the adequacy of the design bases maximum expected differential pressure assumption for the containment sump suction valve in previous evaluations of the issue. (Section 1R21.2.1.8) (IMC0305, aspect P.1 (d))

- Green. The team identified a finding of very low safety significance involving a non-cited violation (NCV) of Technical Specification 5.4.1, Procedures. Specifically, an emergency operating procedure (EOP) figure referenced in loss-of-coolant accident procedures had not been revised, as required, following plant modifications for an extended power uprate (EPU). The EPU analysis increased the minimum injection flow needed to provide decay heat removal during a loss-of-coolant accident. Constellation entered the issue into their corrective action program, revised the affected EOP figure and performed a preliminary extent of condition review of other operating procedures affected by EPU.

  The finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, if used during a LOCA event response, EOP Figure-6 would have allowed operators to reduce RHR injection flow to a value below that required for minimum decay heat removal. The deficiency was assessed in accordance with NRC (IMC) 0609, Appendix A, Attachment 1, Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations. The team determined this deficiency was of very low safety significance (Green) because it did not represent a loss of system safety function. Operators would have had multiple indications of inadequate decay heat removal, such as core exit thermocouples and reactor vessel level indicating system (RVLIS), and would have had adequate time to respond and increase injection flow.

The finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because Constellation's supervisory and management oversight of contractor work activities on the power uprate project was not adequate to ensure in-progress work was completed. (Section 1R21.2.2.1) (IMC0305, aspect H.4 (c))

B. Licensee-identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Ginna Probabilistic Risk Assessment (PRA) and the U.S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the R. E. Ginna Nuclear Power Plant Significance Determination Process (SDP) Phase 2 Notebook, Revision 2.1, was referenced in the selection of potential components and actions for review. In general, the selection process focused on components and operator actions that had a risk achievement worth (RAW) factor greater than 2.0 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected included a variety of components such as pumps, valves, transformers and batteries.

The team initially compiled a list of a nominal 40 components and 10 operator actions based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 16 components (10 mechanical and 6 electrical) and 4 operator actions. The team's evaluation of possible low design margin included consideration of original design issues and margin reductions due to modifications, including those associated with the extended power uprate (EPU), or as a result of material condition/equipment reliability issues. The assessment included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)1 status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, plant personnel input of equipment issues, and industry operating experience. Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity, available time and extent of training on the action.

This inspection effort included walkdowns of selected components, including a review of selected simulator scenarios. It also included interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet both design bases and risk informed beyond design basis functions. A summary of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Design Reviews (16 Samples)

.2.1.1 Refueling Water Storage Tank (RWST)

a. Inspection Scope

The team reviewed the design basis information and supporting calculations and drawings to identify and verify the design assumptions regarding level and volume of water within the RWST. These design inputs were critical relative to the emergency core cooling system (ECCS) pumps taking suction from the RWST and included available net positive suction head (NPSH), vortexing potential, instrument uncertainty in tank level instruments, and maximum flowrates. Additionally, the volume of the RWST tank contents transferred to the containment sump was reviewed to verify adequate NPSH was available for the ECCS pumps during the recirculation phase of a loss-of-coolant accident (LOCA). The team reviewed documentation of the seismic qualification of the RWST and performed a walkdown to assess its general condition.

b. Findings

No findings of significance were identified.

.2.1.2 Residual Heat Removal (RHR) Pump ‘B’, PAC01B

a. Inspection Scope

The ‘B’ RHR pump was reviewed to verify its ability to meet its design basis head and flowrate requirements in response to accident conditions. This included a review of the NPSH analysis for both the RWST and reactor building sump suction flowpaths. The team verified that design inputs were properly translated into system procedures and tests, and reviewed completed surveillance tests associated with the demonstration of pump operability. The team reviewed applicable emergency operating procedures to verify consistency between system flowpaths and assumptions used in the applicable design analyses. The team reviewed related operating experience as documented in selected condition reports to assess the impact on pump performance capabilities. The team interviewed Constellation engineers to discuss historical pump performance, pump modifications, and associated corrective actions. The team reviewed equipment service conditions and qualification documentation to determine whether the associated motor could operate under postulated abnormal and accident environmental conditions.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, in that, Constellation had not provided adequate design control measures to verify the adequacy of RHR pump net positive suction head (NPSH) margin for the containment sump recirculation mode of operation. Specifically, under certain loss-of-coolant accident (LOCA) conditions, the team determined that additional RHR pump flowpaths would exist through idle containment spray and safety injection pumps which had not been previously evaluated in the design basis NPSH analysis of record.

Description: In 2006, as part of an extended power uprate (EPU) implementation, Constellation revised calculation DA-ME-2005-085, NPSH for ECCS Pumps During Injection and Sump Recirculation, to establish the NPSH margin for the ECCS pumps during the injection and sump recirculation phase of operation. The EPU evaluated a revised post LOCA sump water temperature profile and established a more restrictive time limit (from a nominal 20 hours to 6.5 hours) for re-establishing cold leg safety injection to mitigate the effects of boron acid precipitation within the core.

Procedure ES 1.3, Transfer to Cold Leg Recirculation, Rev. 41, would be used post LOCA to establish the alignment of equipment for containment sump recirculation following the depletion of the refueling water storage tank (RWST) at an indicated level of 28% for a postulated LOCA. This procedure started one RHR pump with suction aligned to the containment sump. After the RWST volume is further reduced from the remaining pumps in operation (containment spray and safety injection pumps), these pumps are secured at an indicated level of 15% in the RWST. The step for securing the containment spray (CS) pump had the operator verify that containment pressure is less than 28 psig so that the containment spray signal can be reset and the CS pump discharge motor operated valves (MOV 860A/B/C/D) closed. If containment pressure was not below 28 psig then the CS injection valves were left opened.

ES 1.3 then directed the operators to align the safety injection (SI) for high-head recirculation alignment by opening the applicable MOV's (857A&C/B) that connect the RHR train to the SI/CS suction header. This was in preparation for starting a SI pump if symptoms indicated that RHR flowrate was inadequate through monitoring of the core exit thermocouples and reactor vessel level indicating system (RVLIS) indication. Containment Spray pumps are not required in design basis recirculation scenarios but may be restarted for scenarios where containment pressure remains above a nominal 43 psig. Thus containment spray pumps were not likely to be started for any postulated large break LOCA scenarios during the recirculation phase of emergency core cooling.

The team questioned Constellation personnel about the potential for flow through the two idle CS and three idle SI pumps during the recirculation phase in a scenario where containment pressure would be greater than 28 psig at the time when the pumps were initially secured from the injection phase at a 15% RWST level. The team noted that these flowpaths bypassed the flow indicator (FI-626) referenced in ES 1.3 for monitoring and limiting flowrate from an operating RHR pump to 1500 gpm. The team also questioned whether an NPSH analysis would have accounted for flow through these idle lines. Additionally, the team noted that ES 1.3 was revised in support of EPU for boron precipitation considerations. This required a SI pump to be started for cold leg injection within 4 to 5.5 hours after initial alignment for sump recirculation. The team was concerned that NPSH analysis had not evaluated the potential lineup of one RHR pump simultaneously supplying the core deluge lines, the SI pump suction to support the cold leg flowpath, and flow through the two CS ring headers through the idle CS pumps and open CS injection valves.

Constellation personnel confirmed that the existing NPSH analysis of record, DA-ME-2005-085, had not evaluated the above system alignments (i.e., low head recirculation to the core upper plenum with open flowpaths to the reactor coolant system through idle SI pumps and containment through idle CS pumps). Constellation initiated condition report CR-2007-006039 to document that the configuration allowed by ES 1.3 had not been evaluated with respect to NPSH margin. An emergency operating procedure revision to ES 1.3 was initiated to keep the SI and CS piping header isolated from the RHR system during low head recirculation and allowed alignment only if high head recirculation is needed to ensure cooldown (i.e. higher reactor backpressure scenarios). Additionally, the revision allowed for SI suction alignment later into the event to preclude boron precipitation and established a continuous action step to close the CS pump discharge isolation valves when containment pressure drops below 28 psig. Constellation performed an operability determination to evaluate the impact on previous NPSH margin. This review demonstrated with reasonable assurance historical operability of the RHR pumps. The team reviewed the evaluation and agreed with Constellation's conclusion. Constellation also was considering actions to track a formal revision to the RHR pump margin analysis. The team determined the completed and proposed corrective actions were reasonable.

Analysis: The team determined a performance deficiency existed in that design control measures were inadequate with respect to ensuring an adequate evaluation of RHR pump NPSH had been performed for postulated design basis accidents. Specifically, a design basis calculation to verify adequate RHR pump NPSH had not included all RHR pump flowpaths for the recirculation phase of operation during postulated large break loss-of-coolant accident conditions.

The finding was more than minor because it was similar to NRC Inspection Manual Chapter (IMC) 0612, Appendix E, Examples of Minor Issues, Example 3.j, in that the deficient NPSH analysis resulted in a condition where there was a reasonable doubt with respect to the operability of the RHR pumps. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that did not result in a loss of RHR pump operability.

Enforcement: 10 CFR 50, Appendix B, Criterion III, Design Control, requires, in part, that design control measures shall provide for verifying the adequacy of design.

Contrary to the above, as of September 7, 2007, Constellation had not verified the adequacy of design with respect to the NPSH analysis for the RHR pumps. Specifically, during certain post EPU LOCA scenarios where containment pressure was predicted to be greater than 28 psig, additional RHR pump flow would be diverted through idle CS and SI pumps to the containment and reactor coolant system respectively. The affect of that additional RHR flow had not been accounted for in engineering calculation DA-ME-2005-085, NPSH for ECCS Pumps during injection and Sump Recirculation. Because this violation was of very low safety significance and was entered into Constellation's corrective action program (CR-2007-006039), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000244/2007006-01, Inadequate Evaluation of Residual Heat Removal Pump NPSH for Containment Sump Recirculation Scenarios)**

.2.1.3 Safety Injection (SI) Pump 'A', PSI01A

a. Inspection Scope

The team reviewed hydraulic calculations, technical specifications, accident analysis pump performance assumptions and drawings to ensure the SI pump was capable of meeting system functional and design basis requirements. The team verified that design inputs such as pump head and flowrate requirements were properly translated into system procedures and tests, and reviewed completed surveillance tests associated with the demonstration of pump operability. The team also reviewed system health reports, and corrective action documents to verify SI pump design margins were maintained. The team performed a walkdown of the SI pump area to assess the general condition of the pump. Lastly, the team reviewed equipment service conditions and qualification documentation to determine whether the associated motor could operate under postulated accident environmental condtions.

b. Findings

No findings of significance were identified.

.2.1.4 Service Water (SW) Pump 'A', PSW01A

a. Inspection Scope

Service Water Pump, PSW01A, was reviewed, as a representative sample of the SW pumps, to verify its ability to meet the assumed design basis flowrate requirements in response to transient and accident conditions. The team reviewed design information and historical corrective action documents to determine whether there was the potential for a common cause failure mechanism of the pumps. The team reviewed design documents, including drawings, calculations, procedures, tests and modifications. This review was performed to ensure that, considering allowable pump degradation, the pumps were capable of meeting design flowrates. The team interviewed the system engineer to assess the current condition of the pump and reviewed the system health report. The team performed walkdowns of the SW pump area to assess the general condition of the pump. Lastly, the team reviewed equipment service conditions and qualification documents to determine whether the motors could operate under worst case environmental conditions.

b. Findings

No findings of significance were identified.

.2.1.5 Standby Auxiliary Feedwater (SAFW) Pump ‘C’, PSF01A

a. Inspection Scope

The team reviewed the SAFW pump design, focusing on the ability for the pump to meet extended power uprate (EPU) requirements, which resulted in increased AFW flow requirements for the steam generators. The team reviewed design inputs to ensure they were properly translated into system procedures and tests, and reviewed completed surveillance tests associated with the demonstration of pump operability for EPU conditions. The team reviewed recent results of inspection and flushing of service water pipe line segments to the pump suction to ensure the lines would support full design flow when required. The frequency of pipe line inspections was also reviewed to ensure there was no accumulation of silt or biofouling in the lines that could affect design capabilities of the pump. The team reviewed inputs and assumptions in hydraulic calculations that were performed to ensure backup water sources (i.e. firewater) could be aligned as a suction source for the pump in the event of a loss of service water. SAFW pump room ventilation calculations were reviewed to ensure equipment would operate within design temperature limits. The team performed a walkdown to assess the general condition of the SAFW pump. Additionally, the team reviewed equipment service conditions and qualification documentation to determine whether the motor could operate under worst case environmental conditions.

b. Findings

No findings of significance were identified.

.2.1.6 Turbine Driven Auxiliary Feedwater (TDAFW) Pump, PAF03

a. Inspection Scope

The TDAFW pump was reviewed to verify its ability to meet its design basis head and flowrate requirements in response to transient and accident events. The team verified that design inputs were properly translated into system procedures and tests, and reviewed completed surveillance tests associated with the demonstration of pump operability. Accident analysis evaluations for loss-of-normal feedwater were reviewed to ensure appropriate design criteria for the TDAFW pump were used. The adequacy of the TDAFW pump for operation during a station blackout (SBO) condition was reviewed. The team reviewed the design capacity of the condensate storage tank (CST), and the potential for vortexing at the suction source to ensure the availability of the preferred water source. The design and operating procedures for the service water system were reviewed with respect to supporting operability of the TDAFW pump when the normal pump suction source (CST) is depleted. The team also reviewed room temperature requirements and equipment thermal design requirements to ensure the TDAFW pump would operate within design temperature limits. Lastly, the team performed a walkdown to assess the general condition of the TDAFW pump.

b. Findings

No findings of significance were identified.

.2.1.7 RHR Heat Exchanger (HX) ‘B’ Outlet Valve, AOV-625

a. Inspection Scope

The team reviewed RHR system hydraulic calculations to determine the required opening valve disc position to ensure adequate flow is delivered to the vessel while ensuring sufficient RHR pump NPSH margin is maintained. Inservice testing (IST) procedures were reviewed to verify the valve is tested consistent with the ASME Code requirements. The team conducted a walkdown of AOV-625 to assess the general condition of the valve and to verify the locked position of its associated handwheel.

b. Findings

No findings of significance were identified.

.2.1.8 Residual Heat Removal (RHR) Containment Sump Isolation Valve, MOV 850A

a. Inspection Scope

The team reviewed the RHR containment sump isolation valve, MOV 850A, to verify that it was capable of meeting its design basis requirement of opening during transfer to sump recirculation during postulated small to intermediate break LOCA’s. The review included system calculations and motor operated valve (MOV) analyses to verify that thrust and torque limits and actuator settings were appropriately evaluated. The team reviewed surveillance testing results to verify that the stroke time acceptance criteria were in accordance with the design bases and accident analysis assumptions. Additionally, condition reports relative to the valve were reviewed to ensure conditions did not exist which would invalidate previous assumptions for the capability of the valve. The team performed walkdowns of accessible areas to assess the current material condition of the valve. The team also verified that the current valve configuration was not susceptible to thermal binding or pressure locking conditions. Lastly, the team reviewed load flow and MOV voltage drop calculations to determine whether appropriate voltage values were used in the thrust calculation.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the maximum expected differential pressure established as a design input in the containment sump suction valve (MOV 850A/B) thrust margin evaluation, had not been verified to be a conservative value during the recirculation phase of operation for a small break loss-of-coolant accident (SBLOCA).

Description: NSL-5080-0002 EWR 5080, Rev. 13, Ginna Motor Operated Valve 89-10 Program Scope Evaluation, used 150 psid as a bounding maximum expected differential pressure (MEDP) to verify the capability of the containment sump suction valve to open in support of the establishment of recirculation operation. This input was used to determine the thrust capability of the valve and the associated margin. The team had several concerns that this established MEDP had not been verified to be a conservative value for all postulated accidents such as a SBLOCA.

First, NRC event notification (EN) report (42004), applicable to another facility, had identified a condition where RHR system pressure may increase above previously assumed pressures during RHR system minimum flow operation for a postulated SBLOCA. The team determined that a similar design configuration existed at Ginna. The Ginna RHR system was designed with independent recirculation lines located downstream of each RHR heat exchanger unit, and upstream of a check valve. Each of the two lines included an orifice sized for a minimum flowrate of 200 gpm. The two lines tie together and connect to the common suction line (outlet of the RWST). For a SBLOCA, the safety injection system would be initiated and operators would enter EOP E-0, Reactor Trip or Safety Injection. At the onset of the accident, both RHR pumps would be operating; however, component cooling water would not be initiated to either RHR heat exchanger as this is a manual action performed later in the event, prior to establishing recirculation from the sump. A SBLOCA scenario would result in reactor coolant system pressure remaining at a pressure higher than the shutoff head of the RHR pumps (140 psig). Therefore, as water recirculates through the heat exchangers and back to the pump suction, heat is added to the water proportional to the pumps horsepower, efficiency and piping friction. As the RHR system fluid increases in temperature, a corresponding increase in system pressure will potentially occur. The team determined that there was no verified design analysis for this configuration and was concerned that system pressure may exceed the previously established 150 psid design input MEDP for the containment sump suction valves.

Secondly, Westinghouse Topical Report, WCAP-13097, System Operating Basis for Motor Operated Valves, revision 0, December 1991, had established a plant generic MEDP for containment sump valves of a nominal 585 psid. The evaluated scenario maximized the valve outlet pressure by assuming that during a postulated SBLOCA, the check valve between the refueling water storage tank (RWST) and RHR suction header is leak-tight, and that normal back-leakage through the RHR deluge lines to the reactor coolant system check valves pressurizes the RHR piping up to the system discharge relief valve setting of a nominal 600 psig. The team noted that the MOV in the deluge line from the RHR system to the RCS opens on a safety injection (SI) signal and therefore the check valves (853A/B) in the deluge lines would be relied on to prevent pressurization of the RHR system to its relief valve setting in a SBLOCA scenario. The team requested the leak testing criteria for these valves. During their review of the issue Constellation determined that the existing leakage acceptance criteria did not support the 150 MEDP input utilized in the MOV 850A/B thrust margin analysis.

Finally, the establishment of 150 psid as the design input MEDP was based on emergency operating procedure (EOP) attachment 14.6, RHR Pressure Reduction, established in the mid 1990s. The team determined that procedures E-1, Loss Of Reactor or Secondary Coolant, Rev. 35, and ES 1.3, Transfer to Cold Leg Recirculation, Rev. 41, did not ensure that RHR pressure would be consistent with the150 psid MEDP assumption prior to attempting to initially open the containment sump suction valves. Attachment 14.6 would be performed only if the valves failed to operate. Additionally, this compensatory action in Attachment 14.6 utilized non-safety related instrument air to support the depressurization of the system. The team determined that the existing procedures had not supported or verified 150 psid to be an adequate input for the MOV design analysis. The team noted Constellation had identified this concern of relying on non-safety related instrument air in their review of operating experience related to RHR pressurization during SBLOCAs within a condition report (CR 2005-4966) evaluated in 2005. However, actions to address this concern had not been implemented as of the time of this inspection.

Constellation entered these concerns into their corrective action system as condition reports 2007-6351, 2007-6293, and 2007-6292, respectively. Constellation performed a detailed evaluation of several potential SBLOCA scenarios to demonstrate MOV 850A/B operating margin. The analysis determined an MEDP across the valves of 310 psid. The actuator capability supported a maximum allowable differential pressure across the valve of 335 psid and the team noted additional conservatisms which were applied in the analysis such as the assumption of degraded voltage conditions. Constellation also reviewed actual leak testing data from the deluge line check valves to account for this potential pressurization pathway. Additionally, EOP changes were implemented to ensure that potential pressurization is monitored by operations and depressurization of the system is performed prior to attempting to initially open the valve. Constellation was evaluating further EOP changes or RHR system modifications to increase operating margins. The team determined that the completed and proposed corrective actions were reasonable and agreed with Constellation’s conclusion that the valves remained operable.

Analysis: The performance deficiency associated with this finding was that Constellation had not verified the adequacy of their design analysis with respect to ensuring the capability of their containment sump valves to operate for all postulated accident conditions. This finding was more than minor because it was similar to NRC IMC 0612, Appendix E, Examples of Minor Issues, Example 3.j, in that the design analysis deficiency resulted in a condition where there was reasonable doubt regarding the operability of the containment sump suction valves. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, the team conducted a Phase I SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that did not result in a loss of operability.

The team determined that this issue had a crosscutting performance aspect in the area of Problem Identification and Resolution, Corrective Action Program. Specifically, Constellation had not taken appropriate corrective actions to address the adequacy of their design bases maximum expected differential pressure assumption for the containment sump suction valves in a previous corrective action review within condition report 2005-4966. (IMC0305, aspect P.1 (d))

Enforcement: 10 CFR 50 Appendix B, Criterion III, Design Control, requires, in part, that design control measures provide for verifying or checking the adequacy of design. Contrary to the above, as of September 6, 2007, measures had not been established to verify that the MEDP established in the MOV 850A/B design analysis was a conservative value for all postulated accident conditions. Because this violation is of very low safety significance and has been entered into Constellation's corrective action program (CR-2007-6292,CR-2007-6351, CR-2007-6293), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000244/2007006-02, Nonconservative Differential Pressure Value Used In Motor Operated Valves 850A/B Design Analysis)**

.2.1.9 Residual Heat Removal (RHR) to SI/CS Suction Header Valve, MOV 857B

a. Inspection Scope

The team reviewed the RHR train to SI/CS suction header valve, MOV 857B, to verify that it was capable of meeting its design basis requirement of opening to support high head injection during sump recirculation for postulated small to intermediate break LOCA's. The review included system calculations and motor operated valve (MOV) analyses to verify that thrust and torque limits and actuator settings were appropriately evaluated. The team reviewed surveillance testing results to verify that the stroke time acceptance criteria were in accordance with the design bases and accident analysis assumptions. Additionally, condition reports relative to the valve were reviewed to ensure conditions did not exist which would invalidate previous assumptions for the capability of the valve.

b. Findings

No findings of significance were identified.

.2.1.10 Service Water (SW) to Component Cooling Water (CCW) Heat Exchanger Valve, MOV 4615

a. Inspection Scope

The team reviewed the SW to CCW heat exchanger valve to verify that it was capable of meeting its design basis requirement of opening to support the heat removal required during postulated accident conditions. The review included system calculations and MOV analyses to verify that thrust and torque limits and actuator settings were appropriately evaluated. Surveillance testing results were reviewed to verify that the stroke time acceptance criteria were in accordance with the design basis and accident analysis assumptions. The team performed a walkdown to assess the current material condition of the valve. The team also verified that the current valve configuration was not susceptible to pressure locking conditions. Additionally, the team reviewed the MOV voltage drop calculation to determine that appropriate voltage values were used in the thrust calculation.

b. Findings

No findings of significance were identified.

.2.1.11 480V Vital Bus 16

a. Inspection Scope

The team reviewed alternating current (AC) load flow calculations to determine whether the 480V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed elementary wiring diagrams for bus feeder and load breakers to verify that control logic was consistent with the system design requirements stated in the updated final safety analysis report (UFSAR). The team reviewed bus and load protective relaying to determine whether it afforded adequate protection to equipment and whether there would be any adverse interactions within the protection scheme that would reduce system reliability. The team reviewed undervoltage relay accuracy calculations to ensure that appropriate tolerances had been applied. The team reviewed setpoint and time delay calculations to verify that relays afforded proper undervoltage protection to safety related equipment, and that settings were adequate to prevent spurious separation of Class 1E buses from the preferred (offsite) power supply.

The team reviewed undervoltage relay scheme logic to verify that it would respond as described in the design bases and determine whether there was a potential for adverse interaction with other control schemes such as diesel generator logic. The team

reviewed maintenance procedures to ensure that tasks and acceptance criteria were consistent with vendor recommendations. Completed surveillances for undervoltage relays were reviewed to verify that relays were performing consistent with assumptions in accuracy calculations. The team reviewed system operating procedures to verify that they were adequate to assure reliable sources of power to the buses, and to determine whether the results of design calculations and modifications had been properly incorporated. Lastly, the team reviewed system health data and corrective action documents to determine whether there were any adverse equipment operating trends.

b. Findings

No findings of significance were identified.

.2.1.12 Emergency Diesel Generator (EDG) 'A', (KDG01A)

a. Inspection Scope

The team reviewed diesel starting circuits to verify that the EDG receives the proper signals to respond to accident and bus undervoltage conditions. The team reviewed diesel output breaker logic and load sequencing circuits to verify that loads would be available within the time assumed in the accident analysis. The team reviewed timing relay accuracy calculations and completed surveillances to determine whether the system was demonstrated to perform as required in the accident analysis. The team reviewed static loading calculations to determine whether the maximum loading under accident conditions was within the diesel ratings. The team performed walkdowns of the EDG to assess the material condition and presence of hazards. Additionally, the team reviewed system health data and corrective action documents to determine whether there were any adverse equipment operating trends.

b. Findings

No findings of significance were identified.

.2.1.13 4Kv Bus 12B

a. Inspection Scope

The team reviewed AC load flow calculations to determine whether the 4160V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed elementary wiring diagrams for bus feeder and load breakers to verify that system control logic was consistent with system design requirements stated in the FSAR. The team reviewed bus and load protective relaying to ensure that it afforded adequate protection to the buses and determine whether there would be any adverse interactions within the protection scheme that would reduce system reliability. The team reviewed system operating procedures to verify that they were adequate to assure reliable sources of power to the buses and that

the results of design calculations and modifications had been properly incorporated.

Additionally, the team reviewed system health data and corrective action documents to determine whether there were any adverse equipment operating trends. The team reviewed 4160V circuit breaker maintenance procedures and schedules to verify that they were consistent with vendor recommendations.

b. Findings

No findings of significance were identified.

2.1.14 120 Vac Instrument Bus ‘C’, IBPDPCBCB

a. Inspection Scope

The team reviewed the sources of power to the bus including the 480Vac system and the 125Vdc system to determine whether they afforded adequate input voltage for all required operating and accident conditions. The team reviewed inverter controls to determine whether there were automatic features such as high or low voltage trips that could interfere with proper operation of the equipment. The team reviewed voltage drop calculations to verify that vital loads received adequate voltage when supplied from primary and alternate sources. The team reviewed maintenance schedules, records, and procedures for major components including inverter and molded case circuit breakers to verify periodic maintenance was consistent with vendor recommendations. The team reviewed modification records including equipment equivalence evaluations to determine whether the original design bases had been adversely affected by maintenance or modifications. The team also reviewed corrective action documents to determine whether there were any adverse equipment performance trends and to determine whether operability assessments were adequate.

b. Findings

No findings of significance were identified.

.2.1.15 Station Service Transformer 14 (PXABSS014)

a. Inspection Scope

The team reviewed AC load flow calculations to verify that the transformer had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed transformer protective relaying to determine whether it afforded adequate protection and whether there would be any adverse interactions that would reduce system reliability. The team reviewed maintenance procedures to ensure that tasks and acceptance criteria were consistent with vendor recommendations. Additionally, the team reviewed system health data and corrective action documents to determine whether there were any adverse equipment operating trends.

b. Findings

No findings of significance were identified.

.2.1.16 125 VDC 'B' Battery

a. Inspection Scope

The team reviewed the station 'B' battery and associated 125Vdc switchgear, buses, chargers and inverters. The team reviewed the battery calculations to verify that the battery sizing would satisfy the requirements of the risk significant loads and that the minimum possible voltage was taken into account. Specifically, the evaluation focused on verifying that the battery and battery chargers were adequately sized to supply the design duty cycle of the 125Vdc system, and that adequate voltage would remain available for the individual load devices required to operate during the station blackout (SBO) coping duration. The team reviewed condition reports and maintenance work orders for the associated battery charger and inverters as well as plant change records for the 125Vdc system. Additionally, a walkdown was performed to visually inspect the physical condition of the battery and battery chargers. During the walkdown, the team visually inspected the battery for signs of degradation such as excessive terminal corrosion and electrolyte leaks. The team also verified the battery chargers were properly aligned and had acceptable indicated voltage and current. The team reviewed battery surveillance test results to verify that applicable test acceptance criteria and test frequency requirements specified for the battery were met. The cognizant design and system engineers were interviewed regarding design aspects and operating history for the battery.

b. Findings

No findings of significance were identified.

.2.2 Review of Low Margin Operator Actions (4 samples)

The team performed a margin assessment of expected operator actions, and selected a sample of operator actions for detailed review based upon risk significance, time dependency of the actions, and factors affecting the likelihood of human error. The operator actions were selected from probabilistic risk assessment (PRA) rankings of human action importance based on risk reduction worth (RRW), risk achievement worth (RAW), and other PRA insights. The non-PRA considerations in the selection process included the following factors:

- Environmental conditions or restrictions for performing the actions;
- Extent of actions to be performed outside of the control room;
- Plant procedures that address the actions;
- Complexity of the actions and need for additional personnel or equipment;
- Reliability and/or redundancy of components associated with the actions;
- Information available for diagnosing conditions and initiating actions;
- Ability of operator to recover from errors while performing task;

- Consequences of failure to complete action;
- Margin between the time needed to complete the actions and the time available prior to adverse consequences; and
- Task included in the Systematic Approach to Training (SAT) based training program, and routine training performed.

.2.2.1 Cold Leg Recirculation during a Large Break LOCA

a. Inspection Scope

The team selected the manual operator actions to establish cold leg recirculation during a large break loss-of-coolant accident (LOCA). Specifically, the actions reviewed were to transfer the emergency core cooling system (ECCS) and containment spray system pump suctions from the refueling water storage tank (RWST) to the containment sump. These actions included:

- Identification of low RWST level
- Stop both residual heat removal (RHR) pumps
- Stop one safety injection and one containment spray pump
- Establish cooling to RHR heat exchangers
- Transfer RHR pump suction from RWST to containment sump
- Start one RHR pump
- Stop all safety injection, containment spray, and charging pumps
- Align safety injection and containment spray pumps for sump recirculation

The team selected this sample because the operator action appeared to have low margin between the time required and the time available to perform the actions. The team reviewed the RWST level detection system accuracies and tolerances, ECCS system design calculations and hydraulic models, and motor operated valve (MOV) design calculations. The team evaluated the available process margins based on fluid flow rates, component design values, instrument tolerances, limiting operational parameters established in engineering analysis and calculations, and tank and sump expected process variables. The team compared the available margins, including margins for NPSH and vortexing, to the predicted or assumed margins in engineering analysis and calculations to verify the reasonableness of the design and operating values.

The team observed an operating crew respond to a large break LOCA in the Ginna simulator; interviewed licensed operators; and reviewed normal, abnormal, and emergency operating procedures (EOPs) to evaluate the time margins to perform the manual actions. In addition, the team performed main control room walkdowns to independently identify operator task complexity. The team compared the available time, based on the identified equipment and operating limits, against the observed operator simulator performance and expected operator response based on nominal procedure usage demonstrated during licensed operator training. The team evaluated those time margins to verify the reasonableness of Constellation's operating and risk assumptions.

b. Findings

Introduction: The team identified a finding of very low safety significance involving a non-cited violation (NCV) of Technical Specification 5.4.1, Procedures. Specifically, an EOP figure referenced in loss-of-coolant accident procedures had not been revised, as required, following plant modifications for an extended power uprate (EPU). The EPU analysis increased the minimum injection flow needed to provide decay heat removal during a loss-of-coolant accident.

Description: Constellation performed an EPU for the Ginna plant which increased reactor thermal power by approximately 17%. In support of the EPU, numerous engineering calculations and analysis were updated, including the EOP setpoint L.2, Minimum Injection Flow for Decay Heat. This setpoint is the basis for EOP Figure-6, Minimum Reactor Coolant System (RCS) Injection.

During the review of operator actions for transfer to cold leg recirculation, the team identified that EOP Figure-6 was revision 0, and had not been revised subsequent to the EPU. EOP Figure-6 is used by the following EOPs:

- ECA-1.1, Loss of Emergency Coolant Recirculation
- ECA-1.3, Response to Sump B Blockage
- ES-1.3, Transfer to Cold Leg Recirculation

During a LOCA event response, EOP Figure-6 provided operators with the minimum RHR flow required to remove decay heat. ECA-1.1 and ECA-1.3 required the operators to reduce RHR flow, using Figure-6, to delay depletion of the RWST when cold leg recirculation capability could not be verified, established, or maintained. ES-1.3 required the operators to reduce RHR flow, using Figure-6, when indication of containment sump blockage was identified. EOP Figure-6, revision 0, would have allowed the operators to reduce RHR flow to approximately 15% less than the minimum RHR flow needed to remove decay heat, post-EPU. Therefore, during a postulated LOCA event that would have required operators to reduce RHR flow to that minimum value, adequate core cooling would not have been maintained. As a result, the event response would have become more complicated because operators would have subsequently identified and had to respond to inadequate RCS makeup flow.

In response to this deficiency, Constellation determined that a new Figure-6 had been prepared by the EPU project, but no procedure change had been performed to incorporate the new figure into the EOPs. Constellation's preliminary extent of condition did not identify any other missed EPU related procedure revisions. Constellation's initial review also determined that a contractor, responsible for implementing EPU procedure changes in operator procedures, had departed the Ginna site prior to issuing a procedure change request for the revised figure and without any apparent turnover of his in-progress work. Constellation entered this issue into their corrective action program as condition report 2007-5998.

Analysis: The team determined this procedure issue was a performance deficiency because EOP Figure-6 had not been revised and was non-conservative with respect to Ginna's EPU analysis. The deficiency was more than minor because it was associated with the procedure quality attribute of mitigating system operation. Specifically, if used during a LOCA event response, EOP Figure-6 would have allowed operators to reduce RHR injection flow to a value below that required for minimum decay heat removal.

The deficiency affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, because it was associated with the cornerstone's attribute for procedure quality. The deficiency was assessed in accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix A, Attachment 1, Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations. The team determined this deficiency was of very low safety significance (Green) because it did not represent a loss of system safety function. Operators would have had multiple indications of inadequate decay heat removal, such as core exit thermocouples and RVLIS, and would have had adequate time to respond and increase injection flow.

The team determined that this issue had a crosscutting performance aspect in the area of Human Performance, Work Practices. Constellation's supervisory and management oversight of contractor work activities on the power uprate project was not adequate to ensure in-progress work was completed after a contractor, who was originally assigned to perform the work, left the Ginna site. (IMC0305, aspect H.4 (c))

Enforcement: Technical Specification 5.4.1, Procedures, required, in part, that written procedures be established, implemented, and maintained for activities including emergency operating procedures. EOP procedure ES-1.3, Transfer to Cold Leg Recirculation, directed emergency operator actions for responding to potential blockage of the containment sump during a loss of coolant accident. The procedure directed the utilization of Figure-6, Minimum RCS Injection, to ensure minimum RCS injection flow requirements were maintained.

Contrary to the above, from approximately November 2006, following implementation of the extended power uprate, until August 2007, Constellation had not maintained Figure-6 of the Ginna EOPs. Constellation entered this finding into their corrective action program as condition report 2007-5998. As an immediate corrective action, Constellation revised the affected figure to correct the deficiency. Because this issue was of very low safety significance, and it was entered into the corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000244/2007006-03, Ginna Emergency Operating Procedures not Updated After Extended Power Uprate)**

.2.2.2 Cold Leg Recirculation during a Small Break LOCA

a. Inspection Scope

The team selected the manual operator actions to establish cold leg recirculation during a small break LOCA. Specifically, the actions reviewed were to transfer the ECCS pump suctions from the RWST to the containment sump. These actions were similar to those for a large break LOCA (see section 2.2.1 above), except that a safety injection pump would be started in the recirculation phase if the reactor pressure were greater than the discharge pressure of an RHR pump.

The team selected this sample because this operator action appeared to have low margin between the time required and the time available to perform the actions. In addition, in this scenario, RHR pumps could be operated for extended periods of time on minimum flow, because the reactor coolant system (RCS) pressure would be greater than the discharge pressure of the RHR pumps. This had the potential to pressurize the isolated RHR system.

The team reviewed the RWST level detection system accuracies and tolerances, ECCS system design calculations and hydraulic models, and MOV design calculations. The team evaluated the available process margins based on fluid flow rates, component design values, instrument tolerances, limiting operational parameters established in engineering analysis and calculations, and tank and sump expected process variables. The team performed limited independent calculations and analyses in several areas, to verify the reasonableness of the design and operating values. The team observed an operating crew respond to a small break LOCA in the Ginna simulator; interviewed licensed operators; and reviewed normal, abnormal, and emergency operating procedures (EOPs) to evaluate the time margins to perform the manual actions. In addition, the team performed main control room walkdowns to independently identify operator task complexity. The team compared the available time, based on the identified equipment and operating limits, against the observed operator simulator performance and expected operator response based on nominal procedure usage demonstrated during licensed operator training. The team evaluated those time margins to verify the reasonableness of Constellation's operating and risk assumptions.

b. Findings

No findings of significance were identified.

.2.2.3 Provide Alternate Cooling to Turbine Driven Auxiliary Feedwater

a. Inspection Scope

The team selected the manual operator actions to establish alternate cooling to the turbine driven auxiliary feedwater (TDAFW) system, during a loss of service water event. Specifically, the actions reviewed were to connect a hose between the fire water system and the TDAFW system, then align valves to supply fire water cooling to the TDAFW lube oil cooler and turbine bearings. The team selected this sample because the operator action was performed under adverse environmental conditions, and the licensee time validation and specific procedure instruction steps did not appear to account for those adverse conditions.

The team reviewed the TDAFW design analysis and surveillance test procedures associated with the fire water system. The team evaluated the available process margins based on fluid pressures and flow rates, component design values, and limiting operational parameters established in engineering analysis and calculations. The team compared the available margins to the predicted or assumed margins in engineering analysis and calculations to verify the reasonableness of the design and operating values. The team observed a non-licensed operator perform an operator training job performance measure for this task, which included a full field simulation of this task. The team interviewed licensed operators and the Ginna safety engineer; and reviewed normal, abnormal, and EOPs to evaluate the time margins to perform the manual actions. The team compared the available time, based on the identified equipment and operating limits, against the observed simulated task performance and expected operator response under adverse environmental conditions. The team evaluated those time margins to verify the reasonableness of Constellation's operating and risk assumptions.

b. Findings

No findings of significance were identified.

.2.2.4 Provide City Fire Water Alternate Suction to Standby Auxiliary Feedwater

a. Inspection Scope

The team selected the manual operator actions to establish an alternate source of suction water to the standby auxiliary feedwater (SAFW) pumps, during a loss of service water event coincident with an unavailability of the TDAFW pump. Specifically, the actions reviewed were to connect a hose between the city fire water system and the SAFW system, then align valves to supply city fire water to the SAFW pump suctions. The team selected this sample because the action appeared to have low margin between the time required and the time available to perform the actions, and the actions required coordination of multiple operators in different field locations.

The team reviewed SAFW design calculations and analysis, and city fire water hydraulic models. The team evaluated the available process margins based on fluid pressures and flow rates, component design values, and limiting operational parameters established in engineering analysis and calculations. The team compared the available margins to the predicted or assumed margins in engineering analysis and calculations to verify the reasonableness of the design and operating values.

The team observed a non-licensed operator perform an operator training job performance measure for this task, which included a full field simulation of this task. The team interviewed licensed operators and reviewed normal, abnormal, and EOPs to evaluate the time margins to perform the manual actions. The team compared the available time, based on the identified equipment and operating limits, against the observed simulated task performance and expected operator response. The team evaluated those time margins to verify the reasonableness of Constellation's operating and risk assumptions.

b. Findings

No findings of significance were identified.

.3 Review of Industry Operating Experience (OE) and Generic Issues (4 Samples)

a. Inspection Scope

The team reviewed selected OE issues for applicability at the R. E. Ginna Nuclear Power Plant. The team performed a review of the OE issues listed below to verify that the licensee had appropriately assessed potential applicability to site equipment and implemented corrective actions as required.

NRC Information Notice (IN) 2006-21: Entrainment of Air into Emergency Core Cooling and Containment Spray Systems

The team assessed Constellation's review and disposition of NRC IN 2006-21. The basis of the information notice was a concern for circumstances that could result in air entrainment in pump suction lines, potentially affecting the operability of emergency core cooling system pumps. The team reviewed Constellation's evaluation for potential vortexing in the ECCS pump suction lines from the RWST. This review included verifying that the inputs and assumptions used in the original evaluation remained valid.

NRC Information Notice (IN) 1997-90: Use of Non-Conservative Acceptance Criteria in Safety Related Pump Surveillance Tests

The team reviewed Constellation's disposition of NRC Information Notice (IN) 97-90, which discussed the potential for using non-conservative acceptance criteria in safety-related pump surveillance test procedures. The notice identified inadequacies in surveillance test procedure acceptance criteria that had the potential for, and in some cases did result in, pumps not meeting their accident analysis performance criteria, even though they met the degradation limits of the ASME pump test criteria. The team verified that Constellation addressed the industry experience for the safety-related pumps that were in the inspection scope, which included verifying that pump IST criteria for differential pressure and flowrate were conservative with respect to design bases requirements.

NRC Bulletin 1988-04: Potential Safety Related Pump Loss

The team reviewed the applicability and disposition of Bulletin 88-04. This bulletin described conditions where potential design deficiencies in the minimum flow lines and interactions between pumps running in parallel in a system, may lead to permanent pump damage due to prolonged operation at or near shutoff head conditions. The review included verifying that the licensee evaluated the minimum flowrate requirements for the safety related pumps that were in the inspection scope, and that the potential for pump-to-pump interaction for parallel pumps was addressed. The team also reviewed AFW pump testing results of a 48 hour endurance run that was performed for the three main AFW pumps, and the two SAFW pumps under minimum flow conditions.

Generic Letter 2006-02: Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power

The team reviewed the Ginna response to GL 2006-02 to determine whether the responses were adequately supported by operating procedures and analyses for the offsite power supply. At the time the Ginna response to the generic letter was submitted, the station was not relying on the Grid Operator's state estimator to predict post trip voltages on the offsite power supply, but has subsequently transitioned to this method. Consequently, the responses in the GL regarding this topic had largely been marked as not applicable. In order to assess the adequacy of the current practices, the team reviewed several topics originally addressed by the GL, such as procedures for assuring the availability of the state estimator program, and interface agreements between the station and the grid operator.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design or qualification issues. In addition, Condition Reports (CRs) written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4AO6 Meetings, Including Exit

Exit Meeting Summary

On September 20, 2007, the team presented the inspection results to Mr. J. Carlin, Site Vice President, and other members of the R. E. Ginna Nuclear Power Plant staff. The team verified that proprietary information reviewed was returned to the licensee.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Carlin	Vice President, Ginna
S. Kimbrough	PRA Engineering
M. Ruby	Licensing Engineer
E. Groh	Assistant Operations Manager (Shift)
J. Pacher	Engineering Manager
P. Swift	System Engineering Supervisor
D. Wilson	Primary and Reactor Engineering Supervisor

NRC Personnel

K. Kolaczyk	Senior Resident Inspector
M. Marshfield	Resident Inspector
W. Schmidt	Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000244/2007006-01	NCV	Inadequate Evaluation of Residual Heat Removal Pump NPSH for Containment Sump Recirculation Scenarios (Section 1R21.2.1.2)
05000244/2007006-02	NCV	Nonconservative Differential Pressure Value Used In Motor Operated Valves 850A/B Design Analysis (Section 1R21.2.1.8)
05000244/2007006-03	NCV	Ginna Emergency Operating Procedure not Updated After Extended Power Uprate (Section 1R21.2.2.1)

LIST OF DOCUMENTS REVIEWED

Calculations

CN-CRA-04-55, R. E. Ginna EPU Program LOCA Long Term Mass and Energy Release and Containment Integrity Analysis, Rev. 1
CN-CRA-04-74, R. E. Ginna GOTHIC Model for LOCA and MSLB Analysis, Rev. 3
CN-SEE-04-86, R. E. Ginna EPU CST Minimum Volume Requirements, Rev. 0
CN-TA-05-30, R. E. Ginna Feedline Break NOTRUMP/RETRAN Analysis for EPU,Rev. 1
DA-CE-2001-021, RWST Block Wall Seismic Interaction Analysis
DA-EE-2001-047, Instrument Bus Electrical System Evaluation, Rev. 1
DA-EE-92-008-07, Effect of Degraded Voltage on Motor Control Center Safety Related Loads, Rev. 1
DA-EE-92-035-21, RWST Uncertaintly Calculation, Instrument Loop Number (RWST-L920), Rev. 3
DA-EE-92-098-01, EDG A Steady State Loading Analyses, Rev. 5
DA-EE-92-111-01, EDG Dynamic Loading Analysis, Rev. 1
DA-EE-92-120-01, EDG B Steady State Loading Analysis, Rev. 5
DA-EE-92-131-06, A.C. Motor Operated Valve Degraded Voltage, Rev. 16
DA-EE-93-006-08, Instrument Performance Evaluation and Setpoint Verification Undervoltage Relays and Voltmeters on 480V Safeguards Buses, Rev. 4
DA-EE-93-104-07, 480V Coordination and Circuit Protection Study, Rev. 6
DA-EE-93-107-07, 4160 Volt Overcurrent Relays Coordination and Circuit Protection Study, Rev. 5
DA-EE-96-068, Offsite Power Load Flow Study, Rev. 3
DA-EE-97-069, Sizing of Vital Batteries A & B, Rev. 3
DA-EE-99-047, 125 VDC System Loads and Voltages, Rev. 1
DA-EE-99-073-16, Qualified Life for RHR Pump Motor S/O 67C68831 S/N 1, Rev. 0
DA-EE-99-098-16, Verification of EQ for a Rewound RHR Pump Motor, Rev. 0
DA-ME-93-076, Maximum Allowable Blockage in the SW Supply Lines to the SAFW Pumps, Rev. 0
DA-ME-94-065, SAFW Building Transient Heatup Analysis, Rev. 0
DA-ME-98-019, MOV Thrust Calculation for MOV 850A, Rev. 3
DA-ME-98-020, MOV Thrust Calculation for MOV 850B, Rev. 3
DA-ME-98-025, MOV Thrust Calculation for MOV 857B, Rev. 2
DA-ME-98-047, MOV Thrust Limit Calculation for MOV 4615, Rev. 1
DA-ME-98-125, Adjustable Travel Stop Position for HCV-624 and 625, Rev. 0
DA-ME-98-129, Service Water Pump Inlet Strainer Performance Evaluation, Rev. 0
DA-ME-98-150, Motor Operated Valve PPM Evaluation for MOV's 857A/B, Rev. 1
DA-ME-99-067, SAFW Pump Room Cooler Performance Evaluation, Rev. 0
DA-ME-99-087, Motor Operated Valve PPM Evaluation for MOV's 850A/B, Rev. 1
DA-ME-2000-001, City Yard Loop Capability to Supply Cooling Water to EDG, SAFW, and to Fight Screen House Fire With a Loss of Service Water, Rev. 4
DA-ME-2001-053, Minimum Engineering Limit for Differential Pressure for Preferred and Standby Auxiliary Feedwater Pumps During Periodic Test, Rev. 1
DA-ME-2005-041, Hydraulic Analysis of AFW System Using Proto Flo Software, Rev. 0

DA-ME-2005-073, Evaluation of Auxiliary Feedwater Pump NPSH Requirements, Rev. 0
DA-ME-2005-085, NPSH For ECCS Pumps During Injection and Sump Recirculation, Rev. 1
DA-ME-2006-016, Containment Spray Pump Restart Criteria During Sump Recirculation, Rev. 0
DA-ME-2007-020, Containment Air Entrainment into the Emergency Core Cooling and Containment Spray Systems, Rev. 0
DA-NS-92-121, AFW Room SBO Heat Up Calculation, Rev. 1
EWR 4960-1, Time Delay Setpoints ESFAS System, Rev. 4
NSL-5080-002 EWR 5080, MOV Program Scope Evaluation, Rev. 3
3S61-M-10, Ginna ECCS System Hydraulic Analysis-KYPIPE, Rev. 0
109682-M-011, CST Volume and Level Requirements for EPU, Rev. 1
109682-M-012, Appendix R Steam Generator Water Solid Cooldown Analysis with RELAP5 for EPU, Rev. 0
109682-M-022, Impact of EPU on the Standby Auxiliary Feedwater System, Rev. 1
109682-M-027, Evaluation of TDAFW Pump Flow Rate for SBO Under EPU Conditions, Rev. 0

Surveillance Test Procedures (completed)

PR-1.1, Protective Relay Calibration 480V Undervoltage and Ground Alarm Scheme for Buses 14, 15, 16, 17, and 18 (3/31/05, 10/15/05)
PT-2.1Q, Safety Injection System Quarterly Test (4/10/07, 7/8/07)
PT-2.2Q, Residual Heat Removal System - Quarterly (3/28/07, 5/7/07, 6/22/07)
PT-2.3, Safeguard Power Operated Valve Operation (6/21/07)
PT-2.7.1, Service Water Pumps (6/9/07, 6/3/07, 2/28/07, 12/1/06)
PT-2.10.10, RHR System Check Valves Full Flow (10/14/06)
PM074-008, CV-853B Valve Leakage (10/15/00, 4/16/02, 10/1/03, 10/29/06)
PT-16Q-T, Auxiliary Feedwater Pump Turbine Quarterly (3/21/06, 12/8/06, 3/15/07)
PT-36Q-C, Standby Auxiliary Feedwater Pump C Quarterly (11/20/06, 2/23/07, 5/22/07)
RSSP-2.1, Safety Injection Functional Test (10/27/07)
RSSP-2.7A, Train A Safety Injection Sequence Timers (10/09/06, 3/25/07)
RSSP-2.7B, Train B Safety Injection Sequence Timers (10/09/06, 3/25/07)

Completed Work Orders

20503873	20601794	20700894	20100994	20400610
20502386	20605653	20605570	20203208	20505500
20402258	20700419	20700203	20100663	
20700419	20701011	20500824		

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2003-1705	2005-5584	2007-1834	2007-5855*	2007-6392
2004-0284	2006-1731	2007-1835	2007-5898*	2007-6392*
2004-2104	2006-2160	2007-2969	2007-5958*	2007-6395*
2004-2461	2006-3090	2007-3005	2007-5998*	2007-6396*
2004-2983	2006-3917	2007-3009	2007-6008*	2007-6397*
2004-3259	2006-4052	2007-3042	2007-6009*	2007-6407*
2005-0387	2006-4311	2007-3108	2007-6033*	2007-6408*
2005-1844	2006-4929	2007-3108	2007-6035*	2007-6419*
2005-2356	2006-5326	2007-3941	2007-6039*	2007-6424*
2005-3540	2006-5488	2007-4136	2007-6060*	2007-6435*
2005-3703	2006-6400	2007-4378	2007-6258*	2007-6458*
2005-3848	2006-7123	2007-4672	2007-6292*	2007-6469*
2005-4337	2006-7274	2007-4686	2007-6293*	2007-6573*
2005-4909	2007-0020	2007-4960	2007-6351*	2007-6640*
2005-5416	2007-0027	2007-5723*	2007-6360*	
2005-5499	2007-1263	2007-5766*		

* Condition Report written as a result of inspection effort

Drawings

02302-0102, 125 VDC Power Distribution System One-Line Diagram, Rev. 15
03201-0102, 120V AC Instrument Bus One-Line Diagram, Rev. 22
03200-0102, AC Power Distribution Panels One-Line Diagram, Rev. 25
10905-0054, Sht. 3, 480V Bus 16 - Unit 18A PT and UV Relays, Rev. 13
10905-0054, Sht. 2, Undervoltage Scheme, Bus 14, Rev. 11
10905-0055, Sht. 1, Elementary Wiring Diagram Undervoltage Scheme Bus 16, Rev. 13
10905-0055, Sht. 2, 480V Bus 16 - Unit 11A PT and UV Relays, Rev. 9
10905-0055, Sht. 3, 480V Bus 16 - Unit 11A PT and UV Relays, Rev. 11
10905-0056, Sht. 1, Undervoltage Scheme, Bus 17, Rev. 15
10905-0056, Sht. 2, Undervoltage Scheme, Bus 17, Rev. 10
10905-0056, Sht. 3, Undervoltage Scheme, Bus 17, Rev. 11
10905-0057, Sht. 1, Undervoltage Scheme, Bus 18, Rev. 14
10905-0057, Sht. 2, Undervoltage Scheme, Bus 18, Rev. 10
10905-0057, Sht. 3, Undervoltage Scheme, Bus 18, Rev. 11
10905-0061, Elementary Wiring Diagram Bus 16 to Bus 15 Tie 52/BT16-15, Rev. 4
10905-0062, Sht. 2, Elementary Wiring Diagram Bus 16 to Bus 14 Tie 52/BT16-14, Rev. 2
10905-0072B, Elementary Wiring Diagram Residual Heat Removal Pump B, Rev. 4
10905-0073A, Elementary Wiring Diagram Safety Injection Pump A, Rev. 4
10905-0074, Elementary Wiring Diagram Safety Injection Pump C Supply C1 PS101C, Rev. 8
10905-0081A, Elementary Wiring Diagram Service Water Pump A, Rev. 4
10905-0102, Sht. 1, Elementary Wiring Diagram EDG B Supply Breaker to Bus 16, Rev. 8
10905-0445, Elementary Wiring Diagram Standby Aux Feedwater Pump C, Rev. 6

10905-0605, Elementary Wiring Diagram RHR Pump Suction from Containment Sump B MOV 850A, Rev. 4
10905-0606, Elementary Wiring Diagram RHR Pump Suction from Containment Sump B MOV 850B, Rev. 5
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21946-0021, Sht. 1, Control Schematic Bus 12B to Bus 11B Tie (52/BTB-B), Rev. 3
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21946-0061, Sht. 1, Control Schematic Bus 16 to Bus 15 Tie 52/BT 16-15, Rev. 3
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21946-0570, Sht. 2, Bus 12B Alternate Power Source, Rev. 3
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21946-0605, Sht. 2, Control Schematic RHR Pump Suction from Cnmt Sump B MOV-850A, Rev. 4
21946-0613, Sht. 1, Control Schematic RHR Pump Discharge to SI Pump Suction, MOV-857B, Rev. 4
33013-0623, Sht. 1, Main One Line Operating Diagram, Rev. 22
33013-0623, Sht. 2, Main One Line Operating Diagram, Rev. 10
33013-1237, Auxiliary Feedwater P&ID, Rev. 54
33013-1238, Standby Auxiliary Feedwater P&ID, Rev. 25
33013-1245, Auxiliary Coolant Component Cooling Water P&ID, Rev. 30
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33013-1246, Sht. 2, Auxiliary Coolant Component Cooling Water P&ID, Rev. 12
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33013-1265, Sht. 2, Chemical & Volume Control - Charging P&ID, Rev. 16
33013-1736, Sht. 1, Control Schematic EDG A, Rev. 14
33013-1736, Sht. 2, Control Schematic EDG A, Rev. 14
33013-1949, Electrical Three Line Diagram - 480V Generation Metering and Relaying, Rev. 12
33013-1950, Electrical Three Line Diagram Synchronizing & Phasing, Rev. 5
33013-2539, AC System Plant Load Distribution, Rev. 18

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A-202, Fire Protection Program Staff Responsibilities, Rev. 23
A-601.9, EOP/AOP Support Document Control, Rev. 6
Acceptance Criteria Basis Form Number 2006-2211, SAFW C Pump DP
Acceptance Criteria Basis Form Number 2006-2235, TDAFW Pump DP
Action Report Number 2000-0267, Silt Buildup in SW Supply Line to MDAFW Pumps

Action Report Number 2001-0286, Recurrence of Silt Buildup in TDAFW/MDAFW SW Supply line
DBCOR Number 2004-0021, Small Break LOCA Data Input, 7/13/04
DBCOR Number 2004-0031, Non-LOCA Data Input, 7/13/04
DBCOR Number 2007-0001, Evaluation of the Potential for Vortex Formation in RWST, 1/12/07
DBCOR Number 2007-0024, Evaluation of the Potential for Vortex Formation in Containment Sump B, 9/10/07
Equivalency Eval 2007-0011, Ametek (SolidState) 7.5 KVA Inverter Sync Board Replacement, Rev. 0
Engineering Evaluation Attached to CR 2007-6039
JPM-061.009, Fire Water Cooling to TDAFW Pump, Rev. 4
JPM-061.013, Align Self Cooling to TDAFW Pump, Rev. 1
JPM-061.011, Supply City Water to SAFW Pumps, Rev. 4
Letter R1233554, GL 2006-02 60 Day Response, April 3, 2006
Letter R1253334, GL 2006-02 Response to Request for Additional Information, January 31, 2007
Letter from B. A. Snow (RG&E) to C. Stahl (NRC), Response to NRC Bulletin 88-04, 7/7/88
Letter from Patrick D. Milano (NRC) to Mary G. Korsnick (Constellation), Amendment RE: Revised LOCA Accident Analyses (TAC No. MC6860), dated May 31, 2006
Letter from Patrick D. Milano (NRC) to Mary G. Korsnick (Constellation), Relief Request Number PR-3 Regarding Testing of AFW Pumps (TAC No. MD0316), dated July 19, 2006
SC-3.15.15, Emergency Fire Equipment Inventory and Inspection, Rev. 87
Safety Evaluation SEV-1076, SI Pump Testing with SI Test Line Open, Rev. 0
PTT-23.52, Containment isolation Valve Leak Rate Testing, Rev. 9
OPG-HOSE-Control, Hose Control Program, Rev. 0
NRC Information Notice 93-26, Grease Solidification Causes Molded Case Circuit Breaker Failure to Close
NRC Information Notice 93-64, Periodic Testing and Preventive Maintenance of Molded Case Circuit Breakers
Operability Evaluation, Differential Pressure across MOV 850A/B During a SBLOCA as well as other matters pertaining to RHR Pump Recirculation, 9/10/07
Operations Night Orders for 08-27-2007
Procedure Change Request No. 2007-0056, ES-1.3 Transfer to Cold Leg Recirculation
Procedure Change Request No. 2007-0057, EOP Attachment Containment Spray
Procedure Change Request No. 2007-02842, EOP Attachment 5.2
R0801C LP, EDG Training, Rev. 29
SBO Analysis, Rev. 4
T640003A, Battery Cross Tie Training, Rev. 1
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T640001A, DC Ground Detection Battery Charger 1A, Rev. 0
T640013A, Major Battery Loads, Rev. 1
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94-0586, Material Change for Service Water Pump Impellers and Wear Rings, Rev. 3
98-076, Service Water Pump Inlet Strainers, Rev. 0
PCR 2004-0029, Install Filter Capacitor on 150A Battery Charger Regulator and Modify Arm Circuits, Rev. 0
PCR 2004-0046, Replace Battery Charger A (BYCA) and Battery Charger B (BYCB), and install battery terminal covers for vital batteries BTRYA, BTRYB and BTRYSP, Rev. 0

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GM-50-02-DB50, Westinghouse 480V Air Circuit Breaker Type DB-50 Maintenance for Type DB-50 Breakers, Rev. 22
GNS-RGE, Station Operating Agreement R.E. Ginna Nuclear Power Plant LLC and Rochester Gas and Electric Corporation, February 19, 2007
O-6.9, Operating Limits for Ginna Station Transmission, Rev. 29
PT-2.10.10, RHR System Check Valves Full Flow Operability Verification, Rev. 7
PT-8.0, RHR System Valves-Seat Leakage Test, Rev. 15
PT-10.3, Station Battery A Service Test, Rev. 32
PT-10.2, Station Battery B Service Test, Rev. 29
PT-12.1, Emergency Diesel Generator A, Rev. 13
RSSP-2.7A, Train A Safety Injection Sequence Timers, Rev. 7

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AP-CCW.2, Loss of CCW during Power Operation, Rev. 22
AP-SW-2, Loss of Service Water, Rev. 8
E-0, Reactor Trip or Safety Injection, Rev. 40
E-0 Step Difference, Rev. 12, Reactor Trip or Safety Injection
E-1, Loss of Reactor or Secondary Coolant, Rev. 35
E-1, Loss of Reactor or Secondary Coolant, Rev. 36
E-1 Background Information, Loss of Reactor or Secondary Coolant, Rev. 15
E-1 Step Difference, Loss of Reactor or Secondary Coolant, Rev. 13
ECA-0.0, Loss of All AC, Rev. 32
ECA-0.0 Step Difference, Loss of All AC, Rev. 11
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ER-SH.1, Response to Loss of Screenhouse, Rev. 2
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FR-H.1, Response to Loss of Secondary Heat Sink, Rev. 35
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ATT-5.2, EOP Attachment - TDAFW Pump Alternate Cooling, Rev. 6
ATT-8.3, EOP Attachment - Non-Vital, Rev. 4
ATT-11.2, EOP Attachment - Diesel Air Compressor, Rev. 5
ATT-14.6, EOP Attachment - RHR Pressure Reduction, Rev. 3
ATT-14.6, EOP Attachment - RHR Pressure Reduction, Rev. 4
ATT-21.0, EOP Attachment - RCS Isolation, Rev. 2
ATT-27.0, EOP Attachment - Automatic Action Verification, Rev. 0
ATT-28.0, EOP Attachment - Containment Spray, Rev. 0
ATT-28.0, EOP Attachment - Containment Spray, Rev. 1
FIG-6.0, EOP Figure - Min RCS Injection, Rev. 0
FIG-6.0, EOP Figure - Min RCS Injection, Rev. 1
FIG-21.0, EOP Figure - Containment Spray Restart Criteria, Rev. 1

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VTD-W0120-4170, Instruction Book De-Ion Air Circuit Breaker Type 50DH350, Rev. 2
VTD-A0348-4003, Agastat Nuclear Qualified Timers and Relays, Rev. 1
VTD-W0120-6965, MPM-DB Breaker Maintenance Program Manual for Safety Related Type DB Low Voltage Metal Enclosed Switchgear, Rev. 3
TB-04-13, Replacement Solutions for Obsolete Molded Case Circuit Breakers UL Testing Breaker Design Life and Trip Band Adjustments

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agency-Wide Documents Access and Management System
AOV	Air Operated Valve
App	Appendix
CR	Condition Report
CS	Containment Spray
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
GL	Generic Letter
IN	Information Notice
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
RVLIS	Reactor Vessel Level Indicating System
SBO	Station Black Out
SDP	Significance Determination Process
SI	Safety Injection
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current