February 19, 2000

EA 99-280

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657 - 4702

SUBJECT: NRC TEAM INSPECTION REPORT NOS. 05000317/1999008 AND 05000318/1999008

Dear Mr. Cruse:

This letter transmits the results of the NRC team inspection conducted at BGE's Calvert Cliffs Nuclear Power Plant (CCNPP) reactor facility from October 4 to October 22, 1999, and continuing in the Region I office until January 6, 2000. The NRC team focused on your engineering activities and included an in-depth review of the operability of the emergency diesel generator 1A. The preliminary findings were discussed with your staff on October 22, 1999, during a subsequent visit on November 11, 1999, and during telephone conversations on December 30, 1999, and January 6, 2000.

Overall, the NRC team found the quality of technical work to be generally good as seen in system availability and performance assessments. There was notably strong performance in the system engineering organization. Additionally, organizational self-assessments were consistently implemented and effective in identifying program and process deficiencies. While the quality of engineering work appeared acceptable, there were several deficiencies noted.

An apparent violation was identified and is being considered for enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. BGE did not document an analysis of the degraded operability of the 1A emergency diesel generator (EDG), and the corrective actions taken to remedy the degraded performance were not evaluated. The inspection team noted BGE did not assure proper documentation and follow up for the degraded condition of the diesel which resulted in an apparent violation of 10 CFR 50.59. Specifically BGE implemented a change to procedures described in the Updated Final Safety Analysis Report (UFSAR) regarding how the plant would respond to accidents and transients when using EDG 1A. The change would have manually added safety and non-safety loads to the 1A EDG in the event of a design basis accident. That change was made without analyzing if the consequence of the use or malfunction of these loads would have increased the probability of occurrence or the consequence of an accident as previously described in the UFSAR. No Notice of Violation is presently being issued for this inspection finding. Please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

We also found several examples where procedures were not followed. The examples are described in the enclosed report. The NRC has determined that one Severity Level IV violation of NRC requirements occurred. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999, 64 FR 61142). The NCV involves not promptly implementing a modification to prevent the effects of water-hammer. If you contest this violation or the severity level of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Calvert Cliffs Nuclear Power Plant facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room. Should you have any questions regarding this report, please contact Mr. William H. Ruland at 610-337-5376.

We appreciate your cooperation in this matter. No response to this letter is required.

Sincerely,

/RA Brian E. Holian
Acting For/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000317, 05000318

Enclosure: Inspection Report Nos. 05000317 and 05000318/1999008

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REGION I

Docket Nos:	05000317, 05000318
License Nos:	DPR-53, DPR-69
Report Nos:	05000317/1999008, 05000318/1999008
Licensee:	Baltimore Gas & Electric Company
Facility:	Calvert Cliffs Nuclear Power Plant
Dates:	October 4 - 22, 1999 October 22, 1999 - January 6, 2000 (In-Office)
Inspectors:	M. Modes, Team Leader D. Dempsey, Reactor Inspector C. Cahill, Reactor Inspector L. James, Reactor Inspector T. Hoeg, CCNP Resident Inspector
Approved by:	William H. Ruland, Chief Engineering Support Branch Division of Reactor Safety

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Overall, the NRC team found the quality of technical work to be generally good as seen in system availability and performance assessments. There was notably strong performance in the system engineering organization. Additionally, organizational self-assessments were consistently implemented and effective in identifying program and process deficiencies. While the quality of engineering work appeared acceptable, there were several weaknesses noted.

- BGE made changes to procedures, that allowed the manual addition of safety and non-safety loads to EDG 1A to satisfy light loading concerns after April 9, 1998, that affect the way the plant would be operated in response to accidents and transients when using EDG 1A. BGE did not document a safety evaluation that adequately provided the basis that the change was not an unreviewed safety question, resulting in an apparent violation of 10 CFR 50.59. (E1.1)

- The reviewed temporary alterations were generally complete and technically adequate. Three examples were identified in which temporary modification documentation was not completed in accordance with procedure MD-1-100 “Temporary Alterations”. These omissions are being treated as a minor violation. (E1.2)

- BGE’s process for assessing and disseminating industry operating experience was acceptable. For the NRC information notices that were reviewed, the assessments were technically sound and well documented. The backlog of open items was adequately managed. (E1.3)

- The water hammer caused by quick closure of the EDG service water control valve was a condition adverse to quality that was not addressed by promptly implementing a 1994 modification package. This Severity Level IV violation is being treated as a non-cited violation. (E1.4)

- The NRC previously identified that the requirements to document the interval and extent of system walk-downs was not documented. The Principal Engineer of a group is required to document in a memo to the General Supervisor of Plant Engineering Section the “interval and extent of the periodic walk downs”. The absence of guidance in the form of this memo led to inconsistent implementation of walk-downs among the various groups in engineering. (E2.1)

- Plant engineering effectively communicates with station personnel. There was strong representation from site management at the daily manager meetings to discuss plant status and priorities. PORSC meetings were well organized and system engineering provided detailed insights into the subject matter at the meetings. (E2.2)

- Modification change packages were completed in accordance with applicable plant procedures, were technically accurate, and were supported by analysis and safety evaluations. (E3.1)

- The procedures controlling 10 CFR 50.59 safety evaluation screenings and safety evaluations were properly written and controlled. The procedures provided the required information for individuals to prepare, process, and approve engineering services, design changes, and safety evaluations. BGE had a site-wide LAN system where all of the latest procedures could be accessed. (E3.2)

- The thoroughness, structure, and depth of analysis in root cause depends on the importance the requester puts on the problem being evaluated. For all root cause evaluations reviewed by the NRC, it was the requester that performed the root cause evaluation. This process makes an independent root cause more difficult to obtain. (3.3)

- The engineering self-assessments were acceptable. The self-assessment performed by Plant Engineering Section was good with a creative approach and insightful conclusions. (E3.4)

- The OSSRC is regularly engaged in reviewing plant activities. The OSSRC reports did not contain detailed safety-related comments. (E3.5)

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Operability of 1A Emergency Diesel Generator (EDG)

a. Inspection Scope (37550)

To evaluate engineering involvement with the resolution of technical issues, site engineering communications with other departments, and the extent and quality of engineering involvement in site activities, the NRC team reviewed, in depth, the operability determinations performed on the 1A EDG.

b. Observations and Findings

Background

The licensee determined, during conversations with the vendor in the fall of 1997, that the 1A and OC EDGs had the potential to operate below the recommended low load limit as specified by the vendor technical manual. Specifically, the vendor communicated that they "consider that 30% of rated load is the minimum step of smart operation for this kind of engine." The vendor also stated that new or low operating hours engines are more sensitive in cases of low load running, because the piston rings are not perfectly matched with the liner surface. The NRC team interviewed the diesel system and design engineers and reviewed design documents to determine the low-load/ no-load operability status of the 1A and OC EDGs.

The 1A and OC EDGs are described in the Updated Final Safety Analysis Report (UFSAR) section 8.4.2 as 4.16 kV, three-phase, 60 cycle, tandem-engine Societe Alsacienne De Constructions Mecaniques De Mulhouse (SACM) diesel generator sets which have a continuous rating of 5400 kW each. The 1A EDG is safety-related and powers the Unit 1 bus 11. The OC diesel is the station blackout (SBO) diesel generator. The design of the power connections from the SBO diesel generator allow for manual power diversion to any of the safety-related trains in either unit via a class 1E engineered safety features (ESF) bus. Any subsequent discussion of the 1A diesel is applicable to the OC diesel.

Baltimore Gas and Electric (BGE) in a letter to the NRC, dated January 29, 1993, committed to design and qualify the 1A EDG in accordance with Regulatory Guide 1.9, Revision 3 (draft) and IEEE 387-1984. IEEE 387-1984 provides specific guidance for light load and no load operation and qualification. The qualification requirements as defined in section 7.2.1(4) of IEEE 387-1984 state, "Light or no load capability as described in 5.1.2(3) shall be demonstrated by test. Light or no load operation shall be followed by a load application greater than or equal to 50% of the continuous kilowatt rating for a minimum of 0.5 h." The standard also states in section 5.2.1, *Operations,* that, "The diesel-generator units may be utilized to the limit of their power capabilities, as defined by the continuous and short term rating. Unless time and load parameters for light load and no load operation are established by test and documentation, the following precautions shall be taken: (1) When 4 h operation at 30% or less of the continuous rating has been accumulated without at least 0.5 h operation above 50% of the continuous rating, the unit shall be operated at a load of at least 50% of continuous rating for a minimum of 0.5 h." At the time this inspection concluded the licensee had not been able to produce documentation to support that the 1A EDG meets all of the qualification requirements of IEEE 387-1984. This issue will remain as an Unresolved Item pending NRC review of the documentation. **(URI 05000317, 05000318/199908-001)**

The licensee entered the low-load/ no-load concern into their corrective action system as issue report (IR) number IR1-050-947, on December 22, 1997. The IR stated, "Recent discussions with SACM defines unloaded operations as < 30% (1620 KW). Under certain scenarios, DG loading may not get over 30% (ex: Loss-of-Offsite Power (LOOP) w/ modes 5 & 6)." The IR recommended a complete review of loading scenarios. ES199702025-000 was initiated to perform an evaluation of minimum expected long term load on the 1A and OC EDGs for the various accident scenarios. As identified in the Minimum Load, Root Cause Assessment dated June 1, 1999, ES199702025-000 was canceled by the System Manager and Manager of Design Engineering because, "Calculation of Min Load will not resolve the issue."

BGE identified on April 7, 1998, that the SACM engines could fail if operated at < 30% for an extended period of time. The operability concerns raised by CCNPP Plant Engineering Section (PES) resulted in the initiation of IR3-009-582 and Functional Evaluation 98-012. BGE initiated compensatory actions in the form of an administrative load list by memorandum to Shift Managers from the Principal Engineer Electrical and Systems. The list provided four options, listed in order of preference, which would provide adequate minimum safety loads for the 1A EDG. Note 2 on the list stated, "If these loads are not considered available diesel generator should be considered inoperable. If other loads are not available contact PES for direction." The NRC inspector did not question the appropriateness of decisive action taken to remedy a potentially degraded condition, identified in a safety-related component, system, or structure. This degraded operability was evaluated within the guidance of Generic Letter 91-18. However, the subsequent failure to perform an analysis of the impact of adding the loads on the plant configuration during an accident and the possible unreviewed safety question that might result is an apparent violation of 10 CFR 50.59.

During the spring 1998 outage BGE determined that the loads on the load list may not be available to support 1A EDG operability. The 1A EDG is required by technical specifications to support the operations of Unit 2. Specifically, the loss of the 1A EDG would result in the loss of the 11 control room heating, ventilation, air conditioning (CR HVAC) if a LOOP event occurred. To address this concern, BGE initiated Operations Evaluation 98-013 and installed a nonsafety-related load bank to support the minimum load operation of the 1A EDG.

Upon exiting the outage, BGE initiated a change to procedure NO-1-207, Nuclear Shift Turnover, that added a step to ensure sufficient loads were available to meet the 1A EDG minimum load requirement, and added an attachment that listed the loads available to be used. This list of loads contained both safety-related (SR) and nonsafety-related (NSR) loads.

In addition, to alert the operators to a 1A EDG minimum load condition, BGE installed a minimum load alarm under modification ES199800728-000. BGE conducted Operation Evaluation 98-019 and concluded the 1A EDG was degraded, but operable because of the compensatory actions in the form of a load list. Again, BGE failed to analyze if the consequence of the use or malfunction of these loads (which now included NSR loads) would have increased the probability or the consequence of an accident as previously described in the UFSAR. This is another occasion when a documented evaluation would have supported the use of these loads and precipitated consideration of the impact of the loads on the plant configuration during an event.

After the onsite inspection was completed, BGE conducted a review of UFSAR Chapter 14 events for EDG 1A minimum load concerns and documented them in a memo from R. A. Buttner to T. R. Lupold dated November 3, 1999. Attachment 1 of the evaluation showed that normal accident loading, without operator action, would be below the required 1620 kW (30%) on 1A EDG for the following scenarios:

(1) Salt water header outage, -585 KW
(2) Loss of Feedwater Flow, -28 KW
(3) Loss of Non-Emergency AC, -28 KW
(4) Steam Line Break, -10 KW
(5) Large Break LOCA, -60 KW.

This analysis shows that manual operator action is necessary to meet EDG minimum load requirements. NRC review of this analysis also revealed that previous load list assumptions did not account for operational constraints placed on the Unit 1 “swing pumps” and therefore their ability to provide the assumed full load to the 1A EDG. Specifically, as identified by BGE:

(a) 2 salt water pumps cannot be operated on the same header due to flow limitations (resulting from the heat exchanger replacements).
(b) Running the 13 service water pump on the same header only adds 126 kW, not the 254 kW as previously assumed.
(c) The 11 and 13 HPSI pumps cannot be powered from the same bus without entering a technical specification action statement.

BGE’s action with respect to the swing pumps will be addressed as an inspector follow-up item by the resident staff. **(IFI 05000317, 05000318/199908-002)**

In response to the NRC team’s concern about use of NSR loads to maintain the operability of the 1A EDG, BGE stated, in the GS-NPO Notes and Instructions dated October 21, 1999, “When completing Attachment 30 of NO-1-207, ensure that a minimum of 1620 KW of SR load is available. If unable to meet this criterion using only SR loads, then notify the Shift Manager and GSO-NPO immediately to evaluate operability of the 1A EDG.”

<u>Apparent Violation</u>

In accordance with 10 CFR 50.59 “Changes, Tests and Experiments,” BGE may make changes to Calvert Cliffs Nuclear Power Plant’s design as it is described in the UFSAR, without prior Commission approval, unless the change involves an unreviewed safety question. A proposed change, test or experiment involves an unreviewed safety question if:

(1) the probability of occurrence an accident may be increased or,
(2) the consequence of an accident or the malfunction of equipment important to safety previously evaluated in the UFSAR may be increased or,
(3) if a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR may be created.

The licensee shall maintain records of changes in the facility and of changes to procedures made pursuant to 10 CFR 50.59, to the extent that these changes constitute changes in the facility as described in the updated final safety analysis report or to the extent that they constitute changes in procedures as described in the safety analysis report. These records must include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question.

Contrary to the above, on or about April 9, 1998, and subsequently on April 28, 1998, BGE made changes to procedures described in the Final Safety Analysis Report and did not document a safety evaluation that adequately provided the basis that the change was not an unreviewed safety question.

Specifically, BGE implemented a change to procedures described in the UFSAR, namely the description of how the plant would respond to accidents and transients. The change manually added safety and non-safety loads to the 1A EDG in the event of a design basis accident. That change was made without analyzing if the consequence of the use or malfunction of these loads would have increased the probability of occurrence or the consequence of an accident or malfunction as previously described in the Final Safety Analysis Report.

In implementing the provisions of 10 CFR 50.59, BGE is required to maintain records of changes in the facility and of changes to procedures, to the extent that these changes constitute changes in the facility as described in the UFSAR or to the extent that they constitute changes in procedures as described in the UFSAR. These records must include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question. Contrary to the above, on April 9, 1998, and subsequently on April 28, 1998, BGE made changes to procedures that affect the way the plant responds during design basis events and did not document a safety evaluation that adequately provided the basis that the change was not an unreviewed safety question. This is an apparent violation of 10 CFR 50.59 **(EEI 05000317/19908-01)**

c. Conclusion

In implementing the provisions of 10 CFR 50.59, BGE did not have a written safety evaluation which provides the basis for the determination that the change does not involve an unreviewed safety question. BGE made changes to procedures, requiring the manual addition of safety and non-safety loads to EDG 1A, after April 9, 1998, that affect the way the plant would be operated in response to accidents and transients when using EDG 1A and did not document a safety evaluation that adequately provided the basis that the change was not an unreviewed safety question resulting in an apparent violation.

E1.2 Temporary Alterations

a. Inspection Scope (37550)

The NRC team reviewed procedure MD-1-100, "Temporary Alterations," interviewed several system engineers, reviewed approximately six temporary modifications, and walked down temporary alteration 1-99-0053, which used a jumper to by-pass cell # 37 on the No. 12 battery, to determine the adequacy of BGE's control and implementation of temporary alterations.

b. Observations and Findings

BGE discovered on September 9, 1999, that cell # 37 on battery 12 was below it's technical specification (TS) required individual cell voltage (ICV) of 2.10 volts. BGE initiated temporary alteration 1-99-0053 to jumper out the deficient cell. The NRC team reviewed the temporary alteration design justifications and found them technically adequate.

The NRC team walked-down the temporary alteration and found that it was installed and marked in accordance with the design requirements. The NRC team noted that the required post-installation test completion signature was missing from procedure MD-1-100, Attachment 3, section D. Temporary Alterations procedure MD-1-100, section 5.3.D requires in part that "The Installer or Verifier shall perform required Post-Installation Testing and complete the Post-Installation Testing portion of Section D, Attachment 3." The NRC team reviewed several other open temporary alterations Attachment 3, and found that temporary alterations # 1-99-0048 and 2-99-043 were also missing the required post-installation test completion signature. The NRC team verified, by reviewing maintenance order (MO) # 1199903710, that the required retest had been completed on the battery. The NRC team brought this to the attention of the control room. BGE issued IR3-058-232 to identify and track this issue and BGE will review the other temporary alterations.

c. Conclusion

The reviewed temporary alterations were generally complete and technically adequate. Three examples were identified in which temporary modification documentation was not completed in accordance with procedure MD-1-100 “Temporary Alterations.” This is a minor violation of 10 CFR Part 50, Appendix B, Criterion V.

E1.3 Review of Operating Experience

a. Inspection Scope (37550)

The NRC team reviewed BGE’s engineering evaluations of eight NRC Information Notices. The review included administrative procedures for processing industry operating experience, timeliness of licensee evaluations, technical evaluation quality, and corrective actions.

b. Observations and Findings

The process for evaluation and dissemination of industry operating experience is described in procedures NS-1-100, Use of Industry Operating Experience, and NS-1-300, Industry Operating Experience Information Processing. A staff of two engineers, a Nuclear Network Coordinator, and a supervisor reviews and disseminates to the plant staff operating experience information derived from a number of sources, including: the Institute for Nuclear Power Operators (INPO) web page, vendor letters and E-mailings, and the NRC. Approximately 10 to 25 new items are processed each day. Each item is screened for applicability to Calvert Cliffs within three working days to determine whether a detailed assessment is required. Items that are judged to involve conditions adverse to quality are handled by the corrective action program through issue reports. Otherwise, subject matter experts are expected to complete their evaluations within 60 days, with final closure of the review at 120 days. These time lines were established in June 1999 to address program self-assessment observations concerning issue follow-up and backlog management. The current backlog of issues greater than 120 days was 15 (down from approximately 40 in June 1999) and trending downward.

NRC Information Notice 98-45, Cavitation Erosion of Letdown Line Orifices Resulting in Fatigue Cracking of Pipe Welds, dated December 15, 1998, was not captured in the operating experience program. The error appeared to be an isolated case, and the NRC team did not consider the technical issue to be an immediate safety concern. BGE initiated Issue Report IR3-008-258 to identify the cause and determine appropriate corrective actions.

Because Information Notice 97-76, dated October 30, 1997, has not been resolved to date, the response is not considered timely. The notice involved potential degradation of emergency core cooling system throttle valves due to cavitation erosion during the long-term recirculation phase of a loss of coolant accident. BGE’s initial target date of June 3, 1998, was not met because the industry experience staff rejected the initial assessment from plant engineering. The item was re-assigned to design engineering who initiated discussion with the valve vendor. BGE did not anticipate a vendor response that would challenge system operability, but expected that emergency procedures might have to be revised with a caution statement concerning excessive long-term throttling. The NRC team considered BGE's assessment to be reasonable.

c. Conclusions

BGE's process for assessing and disseminating industry operating experience was acceptable. For the NRC Information Notices and GLs reviewed, BGE engineering's assessments and responses were technically sound and well documented. The backlog of open items was adequately managed. The operating experience staff appropriately responded by rejecting a less than acceptable engineering assessment.

E1.4 Water Hammer in Emergency Diesel Generator Service Water Supply

a. Inspection Scope (34550)

The NRC team assessed the corrective actions for water hammer in the six-inch service water supply piping to the heat exchanger stacks of emergency diesel generators 1B, 2B, and 2A. The review included operability determinations, modification packages, and design calculations.

b. Observations and Findings

The relatively quick closing time (approximately two seconds) of the EDG service water supply valves causes water hammer in the supply lines to the heat exchanger stacks. The stacks consist of the jacket water, lubricating oil, and turbo charger after cooler heat exchangers.

In February 1993, BGE found a broken steel strap on the 1B EDG jacket water cooler. Visual examinations of the straps on the 2B and 2A EDGs identified linear indications in the areas of the break transverse to the bends of the straps. Issue Reports IR5-000-746, IR5-028-445, and IR5-028-226 were initiated. The heat exchanger stack is mounted on the EDG skid and is integrally supported by the system piping. The straps provide lateral support to the stack. The condition was evaluated by civil engineering, which concluded that the break probably had been caused by initial bending, tightening (or possibly over tightening) of the clamps combined with years of service. The individuals performing the evaluation concluded the EDGs would remain operable during a seismic event. The strap on the 1B EDG was replaced almost immediately, and the strap on the 2B EDG was replaced in May 1993. However, the strap on the 2A EDG was not replaced until September 1999.

In April 1993, a metallurgical examination concluded that the broken strap had been caused by a combination of a pre-existing defect and sudden loading (such as water hammer), and recommended determining whether conditions existed in the other two Fairbanks Morse EDGs that could lead to water hammer.

To eliminate the cause of the water hammer, BGE developed a modification to increase the stroke time of the service water control valves. The modification package was completed in December 1994 and work orders were initiated in March 1995 to implement the modification. However, due to higher priority work on the diesel generating systems, no attempt was made to implement the modification until Fall 1999 when it was installed on one valve. Modification Change Request MCR-024-014 called for installing Parker fittings and small diameter tubing in the valve actuator vent line to act as a restricting orifice. Although the design was acceptable from a seismic perspective, BGE was not comfortable with the robustness of the installation. During the NRC inspection the modification package was being re-worked to provide a more suitable design.

In April 1999, BGE discovered that an expansion joint on a service water connection to the 2B EDG heat exchanger exceeded the manufacturer's design limit for bellows extension. A similar, but less severe, condition was identified in the expansion joints of the other EDGs. Information regarding the extent to which water hammer contributed to the condition was inconclusive. However, Engineering Service Package (ESP) ES199900616 estimated a pipe displacement of about 0.75-inch acting on the joint due to rapid closure of the service water control valve. Issue report IR3-017-971 and operability determination 99-007 documented the condition. The operability determination evaluated the cycle fatigue on the expansion joint and concluded that the total number of EDG start/stop cycles was well below design limits. The NRC team verified that a compensatory measure to visually inspect the joints for leakage following EDG operation was being implemented. The NRC team found the operability determination to be acceptable.

ES 199900616 replaces the expansion joints and installs a sway strut on the heat exchanger stack of each EDG to reduce the response to flow transients on the service water system. Struts were installed on EDG 2B and 1B in September and October 1999, respectively. Installation of the strut on EDG 2A is scheduled for January 2000.

The design calculation for the struts was described under the modification package section "disposition for acceptability of the change". The NRC team concluded that the calculation lacked rigor because assumptions were not stated or justified, the bases for engineering judgement were not documented, the methodology did not consider valve stroke time or pressure wave velocity, and the independent review, if performed, was not documented. The NRC team performed an independent calculation of water hammer forces using the Joukowski equation, and concluded that BGE's results were acceptable. However, the calculation did not meet the requirements of BGE Engineering Standard ES-22, Calculation Preparation, and Criterion III of 10 CFR Part 50, Appendix B. This failure constitutes a violation of minor significance and is not subject to formal enforcement action. The issue is being captured by the non-cited violation that follows.

The water hammer caused by quick closure of the EDG service water control valve was a condition adverse to quality identified before mid 1994 that was not addressed until the time of this inspection. 10 CFR Part 50, Appendix B, Criterion XVI requires measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. This Severity Level IV violation of 10 CFR Part 50, Appendix B, Criterion XVI, is being treated as a non-cited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as IR3-017-971 and IR5-013-405. **(NCV 05000317, 05000318/19908-02)**

c. Conclusions

The design calculation for new sway struts for the EDG heat exchanger stacks lacked rigor and was inconsistent with BGE's engineering standard for preparation of design calculations. The water hammer caused a condition adverse to quality that resulted in a non-cited violation.

E2 Engineering Support of Facilities and Equipment

E2.1 Structure and System Walk Downs

a. Inspection Scope (37550)

The NRC team interviewed several system engineers, reviewed procedure MN-1-319, "Structure and System Walk-Downs," and reviewed completed system walk-downs to determine the adequacy of the program.

b. Observations and Findings

The NRC team reviewed 12 months of completed electrical system walk-downs and found there was a large disparity in the frequency and extent of walk-downs for the systems reviewed. Procedure MD-1-100 "Temporary Alterations" section 5.1.7 states, in part, that, "At regular intervals (typically monthly, or as negotiated with the responsible Principal Engineer) the System Engineer shall perform a walk-down of the system. The extent of the walk-down shall be negotiated with the Principal Engineer. The interval and extent of the periodic walk-downs shall be documented to the GS-PES in a memo."

The NRC team interviewed the 125 Volts Direct Current (VDC) system engineer and found he was not aware of the memo required by procedure MD-1-100, section 5.1.7. Additionally, the NRC team found that some accessible portions of the 125 VDC system, specifically the batteries, did not have a documented walk-down in more than six months. As a result of this finding the NRC team expanded the review of system walk-downs and found that other systems did not have documented walk-downs within the periodicity prescribed in procedure MD-1-100, section 5.1.7. The most notable finding was that the component cooling water system did not have a complete walk-down of the accessible portions of the system documented in more than two years.

BGE was unable to locate a copy of the memo from the General Supervisor Plant Engineering Section expressing the walk-down expectations and issued IR3-031-856 to track the issue and memorandum 991008-001 to define the walk-down expectations as required by MN-1-319.

NRC integrated inspection report 50-317/98-08 and 50-318/98-08, section E2.3 had also identified that the requirements to document the interval and extent of the walk-downs was not documented. It is the purpose of a system walk-down to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected which, in-turn, satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XVI. The failure to repeatedly document the expectations and therefore the completion of system walk-downs is being treated as a minor violation of 10 CFR Part 50, Appendix B, Criterion XVI. This condition is in the licensee's corrective action program as IR3-031-856 which established that the walk-downs had been performed, but that it was the expectation, documentation and scheduling that did not conform to requirements.

c. Conclusion

The NRC team concluded that the licensee did not adequately convey the walk-down expectations to the system engineers. As a result the system walk-downs were inconsistently performed and often not documented. The expectations provided in memorandum 991008-001 should provide adequate guidance to the system engineers. The failure of the licensee to monitor the performance or condition of structures, systems or components, against licensee established goals was treated as a minor violation.

E2.2 Engineering Communications with Other Departments

a. Inspection Scope (37550)

The NRC team reviewed the extent and effectiveness of engineering communications between operations, maintenance, plant engineering, and design engineering. The NRC team reviewed various plant documentation including procedures, issue reports, engineering services packages, and other engineering documentation and memorandums. The inspection team interviewed engineers and attended several BGE meetings.

b. Observations and Findings

Engineering Organization and Interfaces

The BGE engineering organization at Calvert Cliffs is split between plant engineering and design engineering. The plant engineering staff consists of system engineering and component engineering. The design engineering staff consists of design engineers and technical services engineers.

Through a series of interviews, the NRC team determined that the system managers were responsible for the overall health and availability of their assigned systems. System engineers reported directly to the system managers and performed the daily activities associated with their assigned systems including reviewing IRs, observing surveillance testing, and performing system walk-downs. The interviewed engineers were knowledgeable of their responsibilities and the design requirements of their assigned systems. The system managers and system engineers communicate daily with operations and maintenance on various topics including system surveillance testing, industry events, system maintenance, and system status. Plant engineering personnel were observed in the control room attending surveillance test briefings with operations. Operations and maintenance personnel were interviewed to evaluate the interface between their organizations and engineering. The two groups both stated that plant engineering generally provided good support and was always available when needed.

Through a series of interviews, the NRC determined design engineers were knowledgeable regarding recent design changes and ESPs. The inspector determined that design engineering personnel do not interact with their counterpart plant engineering personnel on a regular basis. The design engineer waits for plant engineering, operations, and maintenance to bring problems. This was discussed with design engineering management who knew the weakness existed and was contemplating various solutions including performance bonuses being tied to interfacing with system engineers. Significant design changes involving the addition of new EDGs and replacement of service water heat exchangers were continuing to present challenges to the station. Continued compensatory measures and operator work-arounds were still present after the design changes were completed.

Meetings

On October 5, 1999, the members of the NRC team attended a plant operations review safety committee (PORSC) meeting. The purpose of the meeting was to update the PORSC on biological fouling conditions on the saltwater service water heat exchangers. The saltwater system manager updated the committee on recent findings regarding the operation and maintenance of the saltwater service water heat exchangers. The system manager logically explained recent heat exchanger biological fouling events and how BGE engineering, maintenance, and operations were affected. Barnacles and sea grass have fouled the new titanium plate and frame type heat exchangers since the summer 1998. Increased maintenance requiring operator actions have resulted from the bio-fouling conditions. The system manager described a saltwater strainer continuous flush mode of operation in which the heat exchangers were being cleaned. The system manager explained how BGE maintenance developed a device which can be inserted into the heat exchanger inspection hand holes to perform localized flushing thereby minimizing the heat exchanger outage times. The system manager pointed out to the committee that conditions of the Chesapeake Bay had been changing annually with new species of bio-fouling life being identified. BGE has been working with the Academy of Natural Sciences on resolving bio-fouling conditions. Over the past year, BGE was evaluating which chemical injection program was the most effective for the conditions at Calvert Cliffs. A conclusion has not yet been made by BGE and further research is required. The meeting was well attended by BGE management personnel.

An October 8 daily morning managers meeting was chaired by the plant general manager and attended by the plant vice president and all site managers or their designated representatives. The meeting discussed recent safety issues, plant status, recent issue reports, industry operating experience, and license renewal activities. The meeting was well attended by BGE management. Good information exchange and good communication between various organizational structures was exhibited.

Procedures

The NRC team reviewed several BGE procedures including EN-1-100, “Engineering Service Process Overview, “EN-1-101, “Design Change and Modification Implementation,” and EN-1-102, “Safety Evaluation Screenings and Safety Evaluations.” The procedures provided the required information for individuals to prepare, process, and approve engineering service packages, design changes, and safety evaluations. BGE had a site-wide LAN system accessible by employees where all of the latest procedures could be read or printed. BGE had a technical library within the security fence for easy access to plant engineering personnel, and another one located outside the security area for the design engineering department. The NRC team verified that both library copies of the above mentioned procedures were of the latest revision and were properly controlled.

Issue Reporting and Assessment

The NRC team reviewed the interaction between the BGE Issues Assessment Unit (IAU), Issue Report Review Group (IRRG), and site engineering. The IRRG, comprised of a multi-disciplined group including engineering, meets periodically to review IRs. In accordance with BGE procedure QL-2-100, “Issue Reporting and Assessment,” the IRRG reviews the risk, screening results, and recommended corrective actions of selected IRs. The selected IRs including IR3-021-942, IR3-026-621, IR3-026-622, and IR3-027-234 were properly entered into the BGE corrective action program. The IRRG reviewed the selected IRs and agreed with the initial report assessment, screening, and assignment given by the IAU. The issue report summary for the IRRG was reviewed at a morning managers’ meeting the following day where the responsible resolution sponsor was represented.

c. Conclusions

Plant engineering is responsible for the everyday engineering activities and effectively communicates with station personnel. Design engineering has had difficulty in resolving service water heat exchanger and EDG design change deficiencies.

BGE conducted daily manager meetings to discuss plant status and priorities which provided a good forum for communicating daily business. A set agenda was followed with representation from site management. BGE PORSC meetings were well organized and system engineering provided detailed insights to the subject matter.

BGE procedures were properly controlled. BGE’s LAN access to procedures and other information is considered a strength. BGE properly responded to NRC generic communication. BGE properly conducted IRRG meetings in accordance with their procedure QL-2-100, “Issue Reporting and Assessment.”

E3 Engineering Procedures and Documentation

E3.1 Plant Modifications

a. Inspection Scope (37550)

The purpose of this portion of the inspection was to assess the effectiveness of the engineering staff performing routine and emergent design activities in support of safe plant operations. The NRC inspector reviewed selected design changes that either have been or will be implemented in Calvert Cliffs Units 1 and 2 to assure their conformance with applicable procedures and NRC requirements.

b. Observations and Findings

Procedure MD-1, Modification Program, Revision 2, September 16, 1998, established the responsibilities, requirements, and guidelines for implementing and controlling design changes. Engineering Services procedures ES-012, Modification Design Scope Documents Preparation, Revision 00, November 8, 1995, and ES-021, Design Input Requirements (DIR) Preparation, Revision 03, September 22, 1997, provided specific guidance on the development of design inputs.

The NRC Team reviewed portions of two modifications to the steam generators' vibration monitoring system: ES199800433, "Replace the Unit 1 General Electric Vibration Monitoring System on 11 and 12 Steam Generator Feed Pump Turbines (STPTs) with New Bently-Nevada 3300 Vibration Monitoring System," and ES1999701201, "Expand Existing Steam Generator Feed Pump (SGFP) and SGFPT Bently-Nevada Vibration Monitoring System." These design changes replaced two obsolete systems to reduce the maintenance and to expand the vibration monitoring capabilities such that one system provides all required system vibration monitoring. The changes included (1) removal of existing monitoring equipment; (2) plugging the holes of the old probe locations; (3) installing new monitoring system, including instrument rack, power supply, system monitoring, vibration monitors, and probes; and (4) providing high vibration alert indication from the new system to the alarm windows. The NRC Team concluded that the design change packages, including the safety evaluations, were technically sound and provided adequate analytical support.

c. Conclusions

The reviewed modification change packages were completed in accordance with applicable plant procedures, were technically accurate, and were supported by analysis and safety evaluations.

E3.2 10 CFR 50.59 Safety Evaluations

a. Inspection Scope (IP 37001)

The NRC team reviewed BGE's procedures and training materials to verify that planned changes to the facility were controlled in accordance with the operating license, the technical specifications, and 10 CFR 50.59. Ten safety evaluations from the past operating cycle were reviewed to verify that the associated changes did not result in unreviewed safety questions that require NRC approval prior to implementation. The NRC team also reviewed several 10 CFR 50.59 applicability screens and operability determinations to verify that safety evaluations were performed when required.

b. Observations and Findings

The procedures that govern the performance of 10 CFR 50.59 safety screens and evaluations are ES-017, "10 CFR 50.59/72.48 Safety Evaluation Screens/Safety Evaluations," and EN 1-102, "Safety Evaluations Screens and Safety Evaluations." The procedures and associated training materials adequately described personnel responsibilities for the preparation, processing, and approval of safety evaluations. The guidance provided by the training materials and procedures was consistent with NRC guidelines, and extensive use was made of lessons learned from NRC generic documents (e.g. Circulars, Information Notices, and Generic Letters) and industry sources.

The screens, operability determinations, and safety evaluations (except for the SACM EDG), that were reviewed were performed in accordance with BGE's administrative procedures and 10 CFR 50.59 requirements. They contained sufficient references and detail to permit an adequate understanding of the proposed changes by independent reviewers. Each evaluation was technically acceptable and provided an adequate basis for determining that no unreviewed safety question was involved.

c. Conclusions

The procedures for the performance of safety evaluation screens and 10 CFR 50.59 safety evaluations were acceptable. Training materials and procedures made good use of lessons learned from NRC generic communications and industry sources. Except for the SACM EDG load list issue, the documentation for the plant changes that were reviewed provided adequate bases for determining that no unreviewed safety questions were involved in the proposed facility changes.

E3.3 Root Causes Analysis

a. Inspection Scope (37550)

The NRC team evaluated the effectiveness of licensee's controls in identifying, resolving, and preventing problems by reviewing a sample of root cause analyses. The intent of the review was to evaluate the strengths or weaknesses in the licensee's controls for the identification and resolution of root cause analysis that could enhance or degrade plant operations or safety.

b. Observations and Findings

Root Cause Analysis IR 199900838 After the modified salt water service water system was started, Unit 1 experienced frequent problems with indicated flow. As a consequence, a plate and frame heat exchanger was disassembled for cleaning only to discover it was clean. Upon further investigation it was determined flow was adequate; however, it was being reported inaccurately. The apparent cause of this inaccuracy was trapped gases in the pressure sensing lines between the flow element and the flow indicating controller. The root cause analysis was requested by the service water system engineer after observing a number of venting evolutions. The service water system engineer was given lead responsibility for the root cause investigation.

The root cause analysis report contains a comprehensive causal analysis including a discussion of previous attempts to solve the problem, significance of the problem, actual and potential consequences, extent of the problem, generic implications, previous related events, corrective actions, and milestones. The report includes an independent review, as an attachment, that includes an evaluation of such things as the consideration of human performance issues, generic implications, corrective and preventative actions, compensatory actions, and risk reduction.

A multidisciplinary team consisting of personnel from operations, instrument maintenance, mechanical engineering design, and plant engineering concluded the flow sensing problems were caused by three related root causes:

(1) The flow sensing lines were becoming clogged with organisms caused by the introduction of oxygenated, nutrient rich, water during weekly venting procedures.

(2) The need for a weekly venting procedure was caused by an inadequate venting procedure.

(3) The venting procedure was inadequate because the instrument lines were poorly routed; especially in the 11 header.

The instrument line routing was part of the modification design of the salt water service water system. In addition to the problems referred to in this root causes analysis, other design problems exist with the salt water system modification. The plate and frame heat exchangers are being cleaned at greater frequency then originally planned, the bypass flow system is less efficient then anticipated, and the intrusion of mussels into the heat exchangers was not anticipated because the heat exchanger test mock-up did not fully model the installed heat exchangers.

Overall there has not been an increase in heat exchanger availability. Because the flow required by the plate and frame heat exchangers is less than the flow required for the original tube heat exchangers, the margin for inlet temperature has been increased. One tube heat exchanger in each train was replaced by two plate and frame heat exchangers. This makes the salt water service water header available at all times when the inlet temperature is below 80° F by only taking out one heat exchanger at a time for maintenance.

Root Cause Analysis IR 199900355 During a planned shut down of Unit Two, on March 12, 1999, the 22 Steam Generator water level dropped to -30 inches because the feed water regulator valve 2-CV-1106 did not maintain steam generator water level while in the automatic mode of operation. The operators manually intervened and, after several attempts, returned the water level to normal. After the transient the feed water valve was returned to automatic control with the observation by an operator that the valve "appeared to be jerky."

The root cause analysis report contains an event narrative, maintenance history, a one sentence description of the problem, followed by a root cause summary. BGE concluded the stem of the valve was originally bent when assembled in May 1, 1993. This conclusion is based on the evidence of the disassembly and the discovery that the upper and lower cage gaskets were reversed which contributed to the bending of the stem. The analysis corrective action summary indicates RPA-1999-1478 was generated to identify the gaskets that were reversed and Action Item 1B199900016 was generated to ensure proper training and supervisory oversight during the next valve evolution. These were plausible causes and the NRC team determining this analysis was thorough.

Root Cause Report IR199800470 This root cause analysis was undertaken because the utility had "...seen a number of failures of Amptectors over the last few years." The report then records the discussion on the subject during a meeting on June 24th of 1998. This discussion is portrayed, in the report, as the problem analysis/validation.

The report then contains a brief historical discussion and a short summary. This is followed by a list of three corrective actions. The short term corrective action is to revise the maintenance procedure. The two long term actions are to identify the aging components and "determine economically the benefits to repair or replace," and to institute a program to repair or replace the old style amptectors. This root cause does not contain any analysis and is not as thorough as the previously discussed root cause.

c. Conclusions

Root causes were acceptable. In the case of the root cause evaluations reviewed by the NRC, it was the requester that performed the root cause evaluation. The NRC team was concerned that if the requester is responsible for performing a root cause, the opportunity to arrive at an independent conclusion is missed.

E3.4 Self Assessments

a. Inspection Scope (37550)

The NRC team evaluated the effectiveness of Calvert Cliffs' engineering self-assessment processes by reviewing implemented self assessments and interviewing engineering management personnel.

b. Observations and Findings

The NRC team reviewed Design Engineering Self-Assessments for 1998 and 1999 and discussed the results of the assessments with the Manager of Design Engineering. The NRC team discussed the manager's evaluation of the self-assessments contained in the "Design Engineering Self-Assessment Review and 1998 Plan," dated May 4, 1998, and the "Design Engineering Self-Assessment Review and 1999 Plan," dated January 4, 1999. These assessments were rigidly structured, concentrated on performance indicators and organization efficiencies, and contained little insight into the organization's weaknesses or strengths. The recommendations addressed the issues raised without offering detailed solutions.

The NRC team reviewed Plant Engineering Section self-assessment dated March 25, 1999. This assessment was creative, insightful and resulted in observations about basic organizational problems that were honest and critical. This assessment was creative and used an assessment tree based on NRC Maintenance Rule as its basic structure. The recommendations contained in the report were also creative and offered specific solutions to the problems.

c. Conclusions

The engineering self assessments were acceptable. The self assessment performed by Plant Engineering Section was good with a creative approach and insightful conclusions.

E3.5 Off-Site Safety Review Committee (OSSRC)

a. Inspection Scope (37550)

The NRC team evaluated the overall effectiveness of the independent safety evaluation group at CCNPP, called the Off-Site Safety Review Committee, by reviewing selected reports to determine whether thorough, in-depth reviews of known weak areas were performed, assess the adequacy of the reviews, and to determine if corrective action recommendations were made and implemented effectively.

b. Observations and Findings

The NRC team reviewed reports 98-06, dated November 19, 1998, 99-03, dated May 20, 1999, 99-04, dated July 15, 1999, and 99-07, dated September 16, 1999. The OSSRC regularly meets every two months with each meeting preceded by a plant tour of two members of the committee. The reports reflect presentations made to the committee covering such areas as licensee amendment requests, license renewal status, and follow up on corrective actions. The reports contained references to subcommittees such as "Significant Safety Issues", "Performance Assessment Subcommittee", "Safety Evaluations Subcommittee", etc., with little detail recorded as to the activities undertaken by the subcommittees. Examples of subjects reviewed by the subcommittees include such things as the detailed report transmitting justification for a change to the technical specification on the Service Water system, Alloy 800 sleeving of steam generators, and radiation exposures during a recent outage. The NRC team noted the OSSRC is scheduled, by the management of CCNPP, in reviewing a substantial number of issues.

c. Conclusions

The OSSRC is regularly engaged in reviewing plant activities. The OSSRC reports do not contain detailed safety-related comments.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) IFI 50-317 and 318/98-12-05: EDG Speed Switch Adapter

On January 13, 1998, BGE identified that the 2B emergency diesel generator had been inoperable for 15 days. BGE determined that a speed switch adapter failed during a surveillance run on December 29, 1997. An inspector follow-up item (IFI) was opened to verify implementation of the spring clip preventive inspection. The NRC team interviewed the system engineer and reviewed repetitive maintenance task, ID 20240014, that replaced the speed switch adapter and found the actions to be acceptable. This IFI is closed.

E8.2 (Closed) Licensee Event Report (LER) 1998-006: "Action Time Exceeded Due to Failed Diesel Generator Governor"

As described in inspection report 50-317/98-02 and 50-318/98-02, during preparations for taking the 1A EDG out-of-service for maintenance, the 1B EDG failed to start during an operability test run on March 25, 1998. Troubleshooting identified that the governor was the likely cause of the EDG's failure to start. Failure of the replacement governor is discussed in Violation 50-317&318/98-02-01. Subsequent inspection and root cause analysis by the EDG vendor determined that the problem was a piece of nylon material lodged in the governor control oil system shutdown solenoid valve. The licensee submitted LER 1998-006 to the NRC on April 21, 1998, in response to this event pursuant to the requirements of 10 CFR 50.73(a)(2)(i).

The licensee's corrective actions included a flush of the control oil system in the 1B EDG and an evaluation of the potential for a common mode failure for the five diesel generators at the facility. The licensee consulted with the diesel vendor, Fairbanks-Morse, and found there was no history of similar governor failures involving nylon foreign material in the governor hydraulic oil. The NRC team interviewed the system engineer and reviewed the operational history of diesel generator governor systems. The NRC team concluded that the LER adequately described the root cause for this event, and that it identified the necessary corrective actions to prevent recurrence. This LER is closed **(LER 1998-006).**

V. Management Meetings

X1 Exit Meeting Summary

The preliminary findings were discussed with your staff on October 22, 1999, during a subsequent visit on November 11, 1999 and during telephone conversations on December 30, 1999, and January 6, 2000. BGE acknowledged the findings presented.

The NRC team asked BGE whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

<u>BGE</u>
C. Cruse, Vice President, Nuclear Energy Division
P. Katz, Plant General Manager
K. Cellars, Manager, Nuclear Engineering
M. Navin, Superintendent, Nuclear Operations
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K. Mills, Supervisor, Plant Operations
T. Bukowski, Vendor assessment Unit
S. Collins, Plant Engineering, Electrical Engineering Unit
G. Detter, Design Engineering
G. Dockstader, Plant Engineering
P. Pieringer, Plant Engineering
C. Sly, Nuclear Regulatory Matters
L. Williams, Plant Engineering

<u>NRC</u>

S. Stewart, Senior Resident Inspector
F. Bower, Resident Inspector
M. Evans, Branch Chief, DRP, RI
W. Ruland, Branch Chief, DRS, RI

INSPECTION PROCEDURES USED

IP 37550: Engineering
IP 37001: 10 CFR 50.59 Safety Evaluation Program

ITEMS OPENED AND CLOSED

Opened

IFI 05000317, 05000318/19908-001	IFI	Documentation supporting qualification of 1A EDG
IFI 05000317, 05000318/19908-002	IFI	BGE actions with respect to swing pumps.
EEI 05000317/19908-01	EEI	10 CFR 50.59 Violation for compensatory actions taken to solve the light load issue of EDG 1A.
NCV 05000317, 05000318/19908-02	NCV	A condition adverse to quality that was not addressed by promptly implementing a 1994 modification package.

Closed

IFI 50-317, 318/98-12-05	EDG Speed Switch Adapter
Licensee Event Report (LER) 1998-006	"Action Time Exceeded Due to Failed Diesel Generator Govenor"

LIST OF ACRONYMS

ASTM	American Society for Testing and Materials
BGE	Baltimore Gas and Electric
CCNPP	Calvert Cliffs Nuclear Power Plant
EDG	Emergency Diesel Generator
ESP	Engineering Service Package
GL	Generic Letter
IAU	Issues Assessment Unit
ICV	Individual Cell Voltage
IFI	Inspector Follow-up Item
INPO	Institute for Nuclear Power Operators
IR	Issue Report
IRRG	Issue Report Review Group
MO	Maintenance Order
NSR	Non-safety Related
OSSRC	Off-Site Safety Review Committee
PES	Plant Engineering Section
PORSC	Plant Operations Review Safety Committee
SACM	Societe Alsacienne De Constructions Mecaniques De Mulhouse
SBO	Station Blackout
SR	Safety Related
STFPT	Steam Generator Feed Pump Turbines
TS	Technical Specification
VDC	Volts Direct Current
UFSAR	Updated Final Safety Analysis Report