1.8 GENERIC ISSUES

1.8.1 THREE MILE ISLAND ACTION ITEMS

On March 28, 1979 a serious accident occurred in Unit 2 of Three Mile Island (TMI) Nuclear Power Plant in Pennsylvania. In response to the lessons learned from the accident, the NRC issued NUREG-0578, "TMI-2 Lessons Learned Task Force Report (Short-Term and Final)" and NUREG-0737, "Clarification of TMI Action Plan Requirements". What follows in this section is a list of the action items from NUREG-0737 and the BGE actions taken to respond to each of them. Baltimore Gas and Electric Company provided to the NRC the implementation status of each of the TMI action items identifying each item as either complete or not applicable to Calvert Cliffs or, in the case of three items, gave the completion status (Reference 1).

The action items are divided into two sections. The first is a statement of the NRC requirement. Documents appearing at the end of this section are NRC issuances on the same subject in addition to NUREG-0737. The second section is the BGE response to the item and the NRC approval. At the end of this section may be identified some sections of the UFSAR where more information can be found. Likewise, where there is information on the subject in the plant Technical Specifications, the words “Technical Specifications” appear. PLEASE NOTE: These references are starting points. They do not represent all the places that information may be found.

I.A.1.1 SHIFT TECHNICAL ADVISOR

Provide an on-shift technical advisor to the shift supervisor. This person may serve more than one unit if qualified on both units. A bachelor's or equivalent degree in a scientific or engineering discipline and plant-specific training is required. (Generic Letter 86-04)

Baltimore Gas and Electric Company submitted the training and qualifications of a Shift Technical Advisor (Reference 2). The NRC concurred with the program, but did not concur with the use of non-technical degree SROs in the program (Reference 3). Baltimore Gas and Electric Company later restated the position that equivalent qualifications were allowed to substitute for a technical degree (Reference 4). Plant procedures and Technical Specifications define the requirements for the on-shift presence of the Shift Technical Advisor and the qualifications necessary for holding the position.

I.A.1.2 SHIFT SUPERVISOR RESPONSIBILITIES

Delegate the non-safety duties of the shift supervisor to another position.

In 1979, the duties of tagging authority and operations refueling outage coordinator were assigned to other positions. This item was reported complete to the NRC (Reference 1). (Sections 12.1.1, 12.1.3; Technical Specifications)

I.A.1.3 SHIFT MANNING

Provisions governing shift staffing shall be included in plant administrative procedures. These procedures shall also restrict the use of overtime for personnel who perform safety-related functions (e.g., senior reactor operators [SROs], reactor operators, health physicists, auxiliary operators, Instrument and Control technicians and key maintenance personnel. (Generic Letter 82-12)

Baltimore Gas and Electric Company took exception to the NRC overtime rules and informed the NRC that procedures had been established to describe the requirements for shift manning and to control overtime for shift operators

(Reference 5). The NRC accepted the shift manning procedure (Reference 6). The issuance of Generic Letter 82-12 superseded the NUREG-0737 rules on shift manning. Baltimore Gas and Electric Company responded with a license amendment request which stated BGE compliance with the generic letter. The NRC approved the license amendment (Reference 7). (Technical Specifications)

I.A.2.1 <u>IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS</u>

An applicant for SRO who does not have an engineering degree must have been a Control Room operator for four years with one year as a licensed operator. An applicant who is a degreed engineer must have two years of nuclear plant experience as an engineer, participate in an SRO training program and have three months on shift as an SRO in training.

Modifications to the SRO training program were submitted to the NRC (References 8 and 9). The NRC approved the BGE approach to Item I.A.2.1.4, “Upgrading RO and SRO Training” (Reference 10). (Section 12.2.1.6; Technical Specifications)

I.A.2.3 <u>ADMINISTRATION OF TRAINING PROGRAMS</u>

Permanent operations instructors must demonstrate SRO qualifications and be enrolled in the appropriate requalification program until the operations training program is accredited.

The operations training program was first accredited by the National Academy for Nuclear Training in 1984 and has been continuously accredited since then. (Sections 12.1.1, 12.1.3, 12.2.1.6; Technical Specifications)

I.A.3.1 <u>REVISE SCOPE AND CRITERIA FOR LICENSING EXAMS</u>

Simulator exams must be included as part of licensing examinations. (Generic Letter 81-29)

Baltimore Gas and Electric Company informed the NRC that simulator examinations are included in the BGE licensing examinations (Reference 11).

I.C.1 <u>SHORT-TERM ACCIDENT AND PROCEDURES REVIEW</u>

Analyses must be performed to address transients and accidents and inadequate core cooling. Technical guidelines must be developed from these analyses. Emergency operating procedures (EOPs) must be upgraded to the level of the technical guidelines and a writer's guide must be provided. Upgraded EOPs must be implemented and personnel trained on them. (Generic Letter 83-23)

Calvert Cliffs adopted the Combustion Engineering Emergency Procedures Guidelines (CEN-152) for the development of EOPs (Reference 12). The guidelines contain the EOP Writer’s Guide, the EOP Verification/Validation Plan and the EOP Training Plan. The NRC accepted the guidelines for implementation with some comments (References 13 and 14). Baltimore Gas and Electric Company submitted the EOP Procedures Generation Package for NRC review (Reference 15). The NRC gave a critique of the package (Reference 16), recommended that it be reviewed by BGE for compliance with the concepts set forth in CEN-152, and stated that the revision did not need to be submitted to the NRC.

I.C.2 SHIFT AND RELIEF TURNOVER PROCEDURES (NUREG-0578 Item 2.2.1.c)

Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the on-coming and off-going individuals responsible for command of the operations in the Control Room. Supplementary checklists and shift logs should be developed for the entire operations organization, including instrument technicians, auxiliary operators and maintenance personnel.

Shift turnover is controlled by plant procedures which requires a turnover checklist.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITIES

Plant procedures shall clearly define the duties, responsibilities and authority of shift supervisors and operators. The highest level of plant management shall periodically reissue a directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant on his or her shift and that clearly establishes that person's authority.

The duties, responsibilities and authority of the operators and shift supervisors are defined in plant procedures. (Section 12.1.1)

I.C.4 CONTROL ROOM ACCESS

Limit access to the Control Room to those individuals responsible for the direct operation of the plant, certain technical supervisors and certain NRC personnel. Establish authority of the person in charge of the Control Room and line of authority and responsibility in the Control Room in an emergency.

Control Room access and watchstander authority and responsibilities are defined in plant procedures.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE

Carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. Implement procedures to ensure that important information on operating experience relating to plant safety inside and outside the plant is continually supplied to operators and others.

Baltimore Gas and Electric Company proposed to create the Plant Operating Experience Assessment Committee to satisfy this requirement (Reference 17). Approval was given by the NRC (Reference 6). Subsequently, BGE proposed to dissolve the committee (Reference 18) and form a new entity called the Industry Operating Experience Review Unit. The unit would perform the same operating experience review function as the committee. This change was approved by the NRC (Reference 19). (Section 12.1.3)

I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Establish an effective system for verifying the correct performance of operating activities.

Baltimore Gas and Electric Company stated that plant procedures would control activities associated with operating activities (Reference 17). Some activities are done by tagging equipment with the knowledge and concurrence of a senior licensed person and including independent verification. The NRC found the program to be acceptable (Reference 6).

I.D.1 CONTROL ROOM DESIGN REVIEWS

Conduct a detailed Control Room design review considering human engineering requirements and correct discrepancies resulting from the review. The objective of this action is to improve the ability of nuclear power plant operators to prevent accidents by improving the information provided to them. (NUREG-700, Generic Letter 82-33)

This was a long-term project. Baltimore Gas and Electric Company submitted the Control Room Design Review program plan (Reference 20), as well as a supplementary report (Reference 21). The BGE approach to resolving this issue was approved by the NRC (Reference 22). (Technical Specifications)

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Install a safety parameter display which displays to operating personnel the minimum set of parameters which defines the safety status of the plant. (NUREG-0696, Generic Letter 82-33)

Baltimore Gas and Electric Company presented the plan for installing the Safety Parameter Display System (References 23 and 24). The NRC approved the system for installation (Reference 25). Baltimore Gas and Electric Company later sent a summarizing report to the NRC (Reference 26). (Section 7.5.5.3)

II.B.1 REACTOR COOLANT SYSTEM VENTS

Install high point vents, remotely operated from the Control Room, in the ARCS and the reactor vessel head. The important safety function of the vents is to enhance core cooling. The vents must not lead to an unacceptable increase in the probability of a LOCA or a challenge to containment integrity.

The proposed design and operating procedure guidelines were presented to the NRC (References 27 and 28). Later, the NRC stated this TMI requirement was superseded by 10 CFR 50.44(c)(3)(iii) and, therefore, the item was considered complete (Reference 29). The work was completed on December 18, 1987. (Section 4.1.2, 4.1.3.6, Technical Specifications)

II.B.2 PLANT SHIELDING

Conduct a radiation and shielding review of plant areas around systems that may contain highly radioactive material as result of an accident. Identify vital areas and equipment. Design and install shielding where the review shows it is necessary to allow access to vital areas and equipment. (Generic Letter 83-37)

The NRC approved the plant modifications and inspected them (References 30 and 31). (Sections 5.1.5.6, 5A.6, 11.2.1, 11.2.2.5)

II.B.3 POST-ACCIDENT SAMPLING

Provide the capability for personnel to obtain a sample of reactor coolant and containment atmosphere under accident conditions. Personnel must be able to take these samples with limited radiation exposure.

Baltimore Gas and Electric Company detailed compliance with the PASS requirements of this item (Reference 32). Compliance was approved by the NRC (Reference 33). Baltimore Gas and Electric Company later proposed a different approach to meeting the requirements of this item (Reference 34). The

NRC accepted the modified system (Reference 35). The NRC subsequently determined (Reference 103) that the PASS was not needed to support emergency response decision making during the initial phases of an accident. They no longer require dedicated equipment to perform PASS functions and the PASS sampling requirements are eliminated from the licensing basis. However, the NRC believes that there are benefits in having post-accident information previously provided by the PASS. Therefore, we committed to maintaining contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump and containment atmosphere, and for monitoring radioactive iodines released to offsite environs. In addition, we committed to maintain a capability for classifying fuel damage events at the Alert level threshold. (Table 5-3; Sections 9.6.2.2, 7.3.2.2, 7.5.8, 9.6.3, 11.2.1)

II.B.4 <u>TRAINING FOR MITIGATING CORE DAMAGE</u>

Develop and implement a program which teaches the use of installed equipment and systems to control and mitigate accidents. This training must be presented to operations personnel from the plant manager through licensed operators.

The NRC observed the BGE operator training and approved the implementation of the program (Reference 36). Development of the training program was approved by the NRC (Reference 37).

II.D.1 <u>RELIEF AND SAFETY VALVE TEST REQUIREMENTS</u>

Test the ARCS relief and safety valves to ensure they will operate under expected conditions created by design basis transients and accidents.

Baltimore Gas and Electric Company informed the NRC (Reference 38) of participation in a PWR utilities response to the NRC recommendations for safety and relief valve testing. The NRC endorsed the BGE response (Reference 39), leaving only operability testing of the PORV block valves as an open item. Technical Specifications require periodic testing of the PORV block valves but do not satisfy this open item. (Section 12.2.1.6)

II.D.3 <u>VALVE POSITION INDICATION</u>

Provide the operator with unambiguous indication of ARCS safety and relief valve position so that appropriate operator actions can be taken.

Baltimore Gas and Electric Company proposed to utilize acoustic monitors with indication in the Control Room to satisfy this requirement (Reference 40). This arrangement was reviewed and accepted by the NRC (Reference 41).

II.E.1 <u>AUXILIARY FEEDWATER SYSTEM EVALUATION</u>

Perform a simplified Auxiliary Feedwater (AFW) System reliability analysis, review of the AFW System using the criteria of Standard Review Plan 10.4.9, and reevaluate the design basis of the AFW flowrate.

The NRC detailed the requirements for this plant (Reference 42) and approved the BGE responses (Reference 43). What follows here are the requirements on which BGE took action.

<u>Short-Term Recommendations</u>

X.2.3.1

GS-2 Perform routine inspections to verify locked open valves are still open.

Originally BGE instituted a Technical Specification surveillance to verify the position of these locked valves. In Technical Specification Amendment Nos. 178 and 152 the inspection of locked valves was dropped because the valves are locked and access to the lock is controlled by procedure. These factors give assurance that the valve will stay in the designated position.

GS-4 Provide plant operators with emergency procedures for transferring to an alternate AFW supply.

Baltimore Gas and Electric Company proposed that the direction for transferring between condensate storage tanks be included in plant operating instructions (Reference 44). The NRC approved this as part of a license amendment (Reference 43).

GS-5 Without an AC power source, provide the required AFW flow for at least two hours from one AFW train.

Baltimore Gas and Electric Company stated the following: 1) that the only components in the steam-driven trains that would lose power on loss of AC power are the motor-operated steam supply valves to the AFW turbines (Reference 45); 2) that there are procedures instructing the operators to manually open the valves on loss of AC power (Reference 46); and 3) that emergency lighting is provided in the area of the handles for these valves (Reference 44). The NRC approved these responses (Reference 43). (Modified by Recommendation GL-3)

GS-6 Confirm AFW flow path availability after a train has been out of service for periodic testing or maintenance.

Baltimore Gas and Electric Company proposed a change to the Technical Specifications to require that the flow path be verified after a cold shutdown of 14 days or greater (Reference 47). Baltimore Gas and Electric Company also proposed that any valve in the flow path that has been repositioned must be returned to the original position and the position must be independently verified (Reference 44). The NRC approved the responses (Reference 43).

GS-8 Install a system to automatically initiate AFW.

Baltimore Gas and Electric Company stated that all criteria had been met (Reference 44). The NRC approved the response (Reference 43). Installation of an automated system was approved as a control-grade installation by the NRC (Reference 43). This approval included only the automatic start of the AFW pumps. The safety-grade system was evaluated under Recommendation GL-1.

<u>Additional Short-Term Recommendations</u>

2.4.2

1. Provide redundant AFW primary supply level indications and low level alarms in the Control Room. The alarm setpoint should allow the operator at least 20 minutes to anticipate the need to make up water or to transfer to an alternate supply.

Baltimore Gas and Electric Company stated that the primary source of AFW (Condensate Storage Tank No. 12) has redundant level indication and the alarm in the Control Room will provide 20 minutes warning (Reference 45). The NRC approved the response (Reference 43). (Sections 10.3.2 and 10.3.3; Bechtel Design Criteria 4.3.5, Bullet 4)

2. Perform a 72-hour endurance test of all AFW system pumps. After a cool-down, run the pumps for one hour.

   This test was conducted on Pump 11. Following that test, the NRC changed the requirement to a 48-hour test. Pump Nos. 12, 21 and 22 were tested to the new standard. The results of the tests were evaluated as acceptable by the NRC (Reference 43).

3. Provide safety-grade indication of AFW flow to each SG in the Control Room. The flow instrument channels shall be powered from the emergency busses consistent with emergency power diversity requirements of Auxiliary Systems Branch Technical Position 10-1 of Standard Review Plan, Section 10.4.9.

   Baltimore Gas and Electric Company stated that the flow indication system was upgraded to safety-related (Reference 17). The NRC found this to be acceptable (Reference 43).

4. Provide a dedicated individual at the manual AFW valves when they are shut for testing.

   There is a technical specification requirement for the presence of this individual.

<u>Long-Term Recommendations</u>

2.4.3

GL-1 Provide a system to automatically start AFW flow. Design and install the system to meet safety-grade requirements.

Control grade circuitry to automatically initiate feedwater flow was installed and was approved by the NRC (Reference 43). The upgrade of the circuitry to safety-related for automatic initiation and for flow indication was approved by the NRC (Reference 48). (Sections 7.2.3.4 (Low SG Water Level), 7.4.5.2 (Manual Operation); 7.10 (Actuation), 7.12 (Diverse AFW Actuation System), 10.3 (AFW System), 14.4.2.2 (Full Power Case), 14.6 (Loss of Feedwater Flow Event), and 14.10.2 (Loss of Non-Emergency Power); Technical Specifications)

GL-2 Install a redundant parallel flow path for the water supply to the AFW system.

Baltimore Gas and Electric Company requested an exemption to this requirement, citing the other changes being made to the system and the results of a cost and reliability study (Reference 49). The study showed that the addition of this requirement to the other modifications would result in a cost increase with little gain in reliability. The NRC agreed (Reference 105).

GL-3 Provide for automatic initiation of at least one AFW pump, its associated flow path and essential instrumentation. Maintain flow for at least two hours independent of AC power.

Automatic initiation is discussed in Recommendation GL-1. Baltimore Gas and Electric Company agreed to replace the motor-operated valves in the steam supply to the AFW pump turbine with air-operated, fail open valves (Reference 45). Flow can be maintained independent of AC power for two hours as discussed in Section 10.3.3. These modifications were found acceptable by the NRC (Reference 43).

Other Long-Term Recommendations

1. Environmentally qualify motor operated steam inlet valves and other equipment affected by main steam and feedwater line breaks.

   As stated in GL-3, the steam supply valves were changed to air operated control valves. Baltimore Gas and Electric Company stated that the solenoid valves associated with the control valves were environmentally qualified (Reference 44). This was approved by the NRC (Reference 43).

2. Because the steam supply to the AFW pump turbines comes from a single pipe source and the discharge from the pumps is directed to a single pipe, the NRC recommended an evaluation of the systems to determine if any changes were necessary to protect the SGs from boiling dry or to prove that the plant could be brought to a safe shutdown.

   Baltimore Gas and Electric Company stated that plant design and procedures were reviewed to ensure that they maintained the capability to provide adequate flow to the SGs in the case of an AFW pipe break (Reference 44). This was approved by the NRC (Reference 43).

   The NRC required flow design basis information for design basis transients and accidents, therefore, the minimum long-term flow rate was reanalyzed (References 51 and 52).

II.E.1.2 AUXILIARY FEEDWATER INITIATION AND FLOW

1. Design and install a safety-grade AFW system with the following features:

   A. The design shall provide for the automatic initiation of the AFW System;

   B. The automatic initiation signals and circuits shall be designed to withstand a single failure;

   C. The initiating signals and circuits shall be testable;

   D. The initiating signals and circuits shall be powered from the emergency busses;

   E. Manual capability to initiate the AFW System from the Control Room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function;

   F. The electrically operated pumps and valves shall be included in the automatic actuation of the loads onto the emergency busses;

G. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW System from the Control Room; and

2. Provide for safety-grade indication of feedwater flow in the Control Room.

See Item GL-1 in Section II.E.1.1 of this discussion.

Baltimore Gas and Electric Company described modifications proposed to meet the control grade automatic start requirement (Reference 53); provided additional information on instrument power sources (Reference 54); and committed to installation of a third AFW train with electrically-driven pumps in each unit, as well as safety-grade AFW flow initiation and automatic initiation systems. Baltimore Gas and Electric Company later provided a more detailed description of the AFW proposed changes (Reference 55).

Compliance with this requirement was approved by the NRC (Reference 48). The Technical Specifications give limiting conditions for operation and surveillance requirements for the automatic operation of the AFW System. Although it was not relied upon by the NRC in their review, Bechtel Design Criteria (Reference 56) provides further information on how BGE met the requirements to ensure adequate flow, provide unit separation, provide auxiliary shutdown capability, improve overall system reliability and incorporate human engineering considerations into control boards and panels.

II.E.3.1 EMERGENCY POWER FOR PRESSURIZER HEATERS

Provide the capability to supply the pressurizer heaters from emergency power in a timely manner consistent with the with safety-related devices. Provide training for operators in the use of the heaters in natural circulation.

Two sets of pressurizer heaters on each unit are connected to emergency busses. The emergency power supply is also a technical specification requirement which was approved by the NRC (Reference 57). Operator training was implemented and was reviewed by the NRC (Reference 58). Baltimore Gas and Electric Company informed the NRC that instructions to the operators regarding timely connection of the heaters to emergency busses are in the Calvert Cliffs emergency procedures and that training programs are implemented. At the same time, BGE also stated that the interfaces between the heaters and the emergency busses are safety-related (Reference 59). The NRC acknowledged receipt of this information (Reference 60).

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

Revise and review procedures used for the control of combustible gas.

Operating and test procedures are reviewed and updated periodically.

Control of hydrogen in Containment during and following a DBE is no longer required. On March 2, 2004, the NRC issued a license amendment that allows removal of the hydrogen recombiners and hydrogen analyzers from the Technical Specifications (Reference 102). Since control of hydrogen in Containment is no longer necessary, the hydrogen recombiners and procedures for them are not required to be maintained. The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond DBE. Maintenance and testing procedures for the analyzers continue to be maintained.

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

The approval of related Technical Specification license amendments (Reference 61) concluded the NRC's review of Item II.E.4.2.

1. Containment isolation systems shall have diverse isolation parameters.

   Baltimore Gas and Electric Company stated that all containment penetrations by non-essential systems are locked shut, close on containment isolation signal, close on safety injection actuation signal, close on SG isolation signal or close on containment radiation signal (Reference 40). (Sections 5.2, 7.3.2.2 and 9.8.2.2; Technical Specifications)

2. Define essential and non-essential systems.

   Baltimore Gas and Electric Company defined essential fluid systems as those that are actively required during the early stages of an accident to control and mitigate the consequences of an accident such that exposure to offsite individuals is not in excess of the limits specified in 10 CFR Part 20 (References 62 and 63). The only exception to the basis for the "essential" designation of essential systems, CEN-125, is containment pressure sensing. Baltimore Gas and Electric Company determined that this system is necessary to monitor containment pressure throughout the accident evolution and is, therefore, an essential system.

3. All non-essential systems shall be automatically isolated.

   See Number 1 in this item. (Table 5-3, Section 5.2)

4. Design of the isolation scheme will not permit reopening of the containment isolation valves without operator action.

   Baltimore Gas and Electric Company stated that the system, as designed, does not allow automatic reset of the containment isolation valves and administrative procedures and controls are in place to ensure continued operability of the isolation system (Reference 40).

5. The isolation signal setpoint pressure that initiates containment isolation for nonessential penetrations must be the minimum compatible with normal conditions.

   Baltimore Gas and Electric Company provided the NRC with a description of the containment isolation pressure setpoint (Reference 2). The setpoint is also supported by BGE design calculations (Reference 64).

6. Containment purge valves that do not satisfy the operability criteria in Branch Technical Position CSB 6-4 (attached to Standard Review Plan 6.2.4) must be sealed closed as defined in Standard Review Plan 6.2.4, item II.3.f during Modes 1-4. Sealed closed valves must be under administrative control to ensure they cannot be inadvertently opened.

   Baltimore Gas and Electric Company applied for a license amendment to stipulate that the containment purge valve operators would be disconnected in Modes 1-4 and the valves would be verified shut (Reference 65). This was approved by the NRC (Reference 61). (Tables 5-3 and 7-4, Technical Specifications)

7. Containment purge valves must close on a high radiation signal.

   The NRC stated the valves were not required to have a high radiation closure signal for containment isolation (Reference 61). The logic behind this position was that the valves would be closed in Modes 1-4 and therefore automatic closure was not necessary. (Table 5-3, Section 7.3.2.2, Table 7-4, Technical Specifications)

II.F.1 <u>GASEOUS EFFLUENT MONITORS</u>

1. Provide monitors that are capable of measuring concentrations of noble gas fission products in plant gaseous effluents during normal operations and during and following an accident. Monitor all potential accident release paths. The detection range shall be from ALARA [As Low As Reasonably Achievable] concentrations to a maximum of 10^5 micro Ci/cc of XE-133.

   Baltimore Gas and Electric Company proposed to incorporate the main vent noble gas wide range gas effluent monitors into the Technical Specifications (References 66, 67, and 68). The NRC concurred with this request and stated that the Technical Specification part of this item was satisfied. Baltimore Gas and Electric Company notified the NRC that the noble gas monitors were installed in the main steam headers (Reference 69). The NRC found this acceptable (Reference 70). (Sections 11.2.3.2.1, 11.2.3.2.11, 11.2.3.2.12; GL 83-37; Technical Specifications)

2. Provide the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. An administrative program should be established, implemented and maintained to ensure this capability (Generic Letter 83-37).

   Baltimore Gas and Electric Company reported that a new wide range noble gas monitor had been installed and that procedures for its use were implemented (Reference 66). The NRC approved the BGE request (Reference 68). (Section 11.2.3.2.1; Technical Specifications).

3. A minimum of two containment high-range radiation monitors shall be provided for each unit. The monitors shall be physically separated and function to measure radiation within the reactor containment during and following an accident.

   The NRC conducted an inspection of the high-range radiation monitors and closed the item (Reference 71). The NRC acknowledged the installation and approved the related technical specifications (Reference 68). (Section 11.2.3.3, Table 11-11; Technical Specifications)

4. Continuous indication of each unit's containment pressure shall be provided in the Control Room during power operation, startup and hot standby. (Generic Letter 83-37) The NRC also required that the measurement and indication capability include three times the design pressure of the containment and minus 5 psig, including a requirement that indication meet the design provisions of Regulatory Guide 1.97, including qualification, redundancy and testability.

   The NRC declared the equipment and installation requirements of this item were satisfied (Reference 7). The NRC conducted an inspection of the containment pressure measurement and indication and closed the item (Reference 72). The NRC acknowledged the installation and approved the related technical specifications (Reference 68). (Sections 4.3.2.1, 7.3.2.2, 7.5.8; Technical Specifications)

5. Continuous indication of each unit's containment water level shall be provided in the Control Room during power operation, startup and hot standby. (Generic Letter 83-37) The instrumentation shall consist of a wide-range indicator covering the range from the bottom of the containment to an elevation equivalent to a 600,000 gallon capacity and a narrow-range indicator which covers the range from the bottom of the containment to the top of the containment sump. The NRC also required that the wide-range instruments shall meet the requirements of Regulatory Guide 1.97 and the narrow-range instruments shall meet the requirements of Regulatory Guide 1.89 (Reference 73).

   In 1984, BGE advised the NRC of the status of this item. The current status can be found in Section 7.5.8. (Section 4.3.2.1; Technical Specifications; Regulatory Guide 1.97)

6. Continuous indication and recording of each unit's containment hydrogen concentration shall be provided in the Control Room within 30 minutes of the initiation of safety injection.

   The NRC declared that the equipment and installation requirements of this item were satisfied (Reference 7). (Section 6.8.1; Technical Specifications; Regulatory Guide 1.97)

   The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond DBE (Reference 102).

II.F.2 <u>INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING</u>

Provide instrumentation which gives an unambiguous, easy to interpret indication of inadequate core cooling.

In 1984 and 1985, BGE submitted the results of a review of post-accident monitoring instrumentation of which inadequate core cooling is a part (References 74 and 75). The NRC found the instrumentation to be acceptable (Reference 76). Baltimore Gas and Electric Company later revised the 1984 submittal (Reference 77). Section 7.5.9 of the UFSAR contains information resulting from that revision. (Technical Specifications; Regulatory Guide 1.97)

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Motive and control components for the PORVs, PORV block valves and pressurizer level indication must be supplied from vital power supplies when offsite power is not available.

Baltimore Gas and Electric Company stated that the motive components of the PORVs and the PORV block valves are connected to vital 480 Volt motor control centers, which are powered by the emergency diesel generators on loss of offsite power (Reference 40). The control components for the valves are connected to the same motor control centers with a 125 Volt battery backup. Two of the pressurizer level indicators are connected to vital DC busses and one is connected to offsite AC power with emergency diesel generator backup. The NRC found this acceptable (Reference 58). (Section 4.2.2, Technical Specifications)

II.K.2 ANALYSIS AND EVALUATION

II.K.2.13 A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

An analysis (CEN-189) was performed. The analysis was endorsed by BGE (Reference 78) and approved by the NRC (Reference 79). (Section 4.1.4.5.4)

II.K.2.17 Analyze the potential for voiding in the ARCS during anticipated transients.

An analysis (CEN-199) was performed and endorsed by BGE (Reference 80). In that letter exception was taken to the 20-hour cooldown period before depressurization for entry into shutdown cooling given in CEN-199. A 17.5 hour period was proposed instead. The analysis and the BGE exception were approved by NRC (Reference 81). (Sections 3.2.3.4 and 3.2.3.6)

II.K.2.19 Provide a benchmark analysis of sequential feedwater flow to the SGs following a loss of feedwater.

The NRC declared that no action was required for CE-designed NSSSs (Reference 50).

II.K.3 B&O TASK FORCE FINAL RECOMMENDATIONS

II.K.3.1 Provide a system that uses the PORV block valves, in an automated mode, to protect against a small-break LOCA.

Baltimore Gas and Electric Company declared that an automated PORV isolation system would present serious challenges to plant safety and was, therefore, not necessary at this plant (Reference 82). The NRC agreed (Reference 83).

II.K.3.2 Submit a report documenting the various actions taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV and show how those actions constitute sufficient improvement in reactor safety.

An analysis (CEN-145) was performed. The analysis was endorsed by BGE (Reference 84) and approved by the NRC (Reference 83).

II.K.3.3 Report safety valve and PORV failures and challenges.

Technical Specifications used to require all such events be reported annually. The Technical Specifications were amended (reference 104) to remove this requirement.

II.K.3.5 Automatic trip of the RCPs during a LOCA.

An analysis (CEN-268) of a strategy called “trip two/leave two” was performed. Baltimore Gas and Electric Company endorsed this strategy, which was approved for Calvert Cliffs by the NRC (References 85 and 86). (Section 7.2.3.3, Table 14.1-3; Generic Letters 83-10a, 83-10b and 86-06)

II.K.3.17 Report on EGGS outages for the five years prior to issuance of the TMI Action Items.

The report was submitted to the NRC and accepted (Reference 87).

II.K.3.25 Determine the effect of loss of cooling water to the RCP seal coolers due to loss of offsite power.

Baltimore Gas and Electric Company presented argument that automatic initiation of RCP seal cooling on loss of offsite power is not necessary (Reference 88). The NRC approved this position (Reference 89).

II.K.3.30 Revise, document and submit the small-break LOCA analysis to show compliance with 10 CFR Part 50, Appendix K.

An analysis (CEN-203) was performed. The analysis was endorsed by BGE (Reference 90). The NRC advised BGE that use of the topical was satisfactory (Reference 91).

II.K.3.31 Submit the small-break LOCA analysis required by II.K.3.30 to show compliance with 10 CFR 50.46.

The NRC approved the analysis for II.K.3.30 and stated that further analysis to satisfy this item was not necessary (Reference 91).

III.A <u>IMPROVING EMERGENCY PREPAREDNESS</u>

III.A.1.2 Establish a technical support center separate from but in close proximity to the Control Room. Establish an operational support center separate from the Control Room and other emergency response facilities (Reference 92). (NUREG-0696)

Baltimore Gas and Electric Company presented a conceptual design for the Technical Support Center and the Emergency Operations Facility (Reference 93). In addition, BGE provided a progress report for the Emergency Operations Facility (Reference 94). The NRC gave partial approval in 1983 (Reference 95). Final approval of the facilities was to be the subject of an appraisal by the NRC but their appraisal program was canceled and the appraisal was not conducted. (Sections 7.5.5.2, 7.8.2.6, 12.6.2.1)

III.A.2 Upgrade emergency plans to provide reasonable assurance that adequate measures can and will be taken in the event of a radiological emergency. Additionally, NUREG-0654 required that plants have the capability to take meteorological measurements from primary and backup systems. (Regulatory Guide 1.97)

The NRC found the upgrade of the emergency plans satisfactory and the onsite and offsite emergency preparedness adequate (Reference 96). (Section 12.6)

The NRC approved the meteorological data upgrades (Reference 97). (Section 2.3.7)

On November 30, 2004 the Patuxent River Naval Air Station discontinued staffing the weather station on a 24/7 schedule. A 10 CFR 50.54(q) evaluation was performed to replace the backup meteorological information provided by Patuxent River Naval Air Station. The new description states that the Emergency Response Plan Implementing Procedures provide instructions for accessing backup meteorological data in the event the primary meteorological data becomes unavailable.

III.D <u>PRIMARY COOLANT OUTSIDE CONTAINMENT</u>

III.D.1.1 Implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. The NRC asked for a review of potential release paths due to design and operator deficiencies (Reference 73). This requirement was a result of an incident at North Anna, Unit 1. (NUREG-0578)

The program, in the form of a Surveillance Test Procedure and a procedure for dumping the reactor coolant drain tank into the containment sump, was approved by the NRC (Reference 41). Baltimore Gas and Electric Company conducted the review and declared that no changes to the plant or procedures were necessary (Reference 98). (Technical Specifications)

The systems involved are:

Safety Injection
Containment Spray
Shutdown Cooling
Containment Sump Recirculation
Containment Atmosphere Sampling
Reactor Coolant Sampling

III.D.3.3 Provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas of the facility where personnel may be present during an accident. Effective monitoring of increasing iodine levels under accident conditions must include portable monitoring instruments using sample media that will collect iodine selectively over xenon.

Baltimore Gas and Electric Company declared that all requirements of this item were being met (Reference 17). The NRC accepted that statement (Reference 99). (Sections 4.3.3.1, 11.2.3.2.1; Technical Specifications)

III.D.3.4 Assure that Control Room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the plant can be safely operated or shut down under design accident conditions.

Control Room ventilation is capable of automatic isolation from outside air and filtered to reduce airborne radioactivity concentration. A Control Room Habitability Study was conducted by BGE and reported to the NRC (Reference 2). An evaluation of the available self-contained breathing apparatus and Control Room infiltration was presented to the NRC in 1982 (Reference 100). The NRC concluded that the BGE response to this item was acceptable (Reference 101). Baltimore Gas and Electric Company's current method of complying with General Design Criterion 19 is discussed in Section 9.8.2.3, Auxiliary Building Ventilating Systems. (Sections 1.2.9.7, 1.4.8, 1C.4, 7.6.2, 11.2.3.2.8; Technical Specifications)

1.8.2 STATION BLACKOUT

On July 21, 1988, 10 CFR Part 50 was amended to include a new section 50.63, “Loss of All Alternating Current Power” (Station Blackout [SBO]). The SBO rule requires that each light water cooled nuclear power plant be able to withstand and recover from an SBO of specified duration. It also identifies the factors which must be considered in specifying the SBO duration. Regulatory Guide 1.155, “Station Blackout,” describes a method acceptable to the NRC for meeting the requirements of 10 CFR 50.63. The Regulatory Guide references NUMARC 87-00, Revision 1 as an acceptable means of evaluating our response to the SBO rule. Our response to the SBO rule is briefly described below.

Station Blackout Duration

With the non-safety-related diesel generator for SBO response (the 0C diesel generator), the duration of a SBO is one hour (the maximum time assumed to start and load the non-safety-related diesel). Only one Unit is assumed to be in an SBO condition. The scenario that the rule proposes is as follows. Both Units are at full power when offsite power is lost. Three diesel generators fail to start. The fourth diesel generator starts and loads the shutdown loads for one Unit. The other Unit is in an SBO. The non-safety-related diesel generator is started and loaded with the SBO Unit shutdown loads within an hour. Restoration of AC power after a SBO is assumed to be from an onsite diesel generator.

Ability to Cope with a Station Blackout

The ability of either Unit to cope with an SBO was evaluated. The Regulatory Guide requires the evaluation of several factors. Each of these factors is described below. The information presented below is based on a one hour coping duration.

Condensate Inventory

The minimum required condensate storage tank level, per the Technical Specifications, provides more than enough water for one hour of decay heat removal for one Unit.

Class 1E Battery Capacity

Battery capacity calculations verify that each of the four 125 VDC Class 1E batteries have capacity to carry SBO loads for at least one hour. This is sufficient for SBO since the 0C diesel generator will be available within one hour to supply the battery chargers. Shutdown loads and equipment needed to start a diesel generator and close its breakers were included in the load profile. The four battery duty cycle calculations utilize a 15% design margin in accordance with IEEE-485 recommendation.

Compressed Air

Air-operated valves relied upon to cope with an SBO for one hour can be operated manually. Valves requiring manual operation are identified in plant procedures.

Loss of Ventilation

Detailed room heatup calculations were performed for nine different areas of the plant. These calculations resulted in modifications to the Control Room ceiling and the battery room heating, ventilation and air conditioning system. Operability of the necessary equipment in these rooms was verified.

Containment Isolation

Containment isolation valves which must be closed or cycled during an SBO event are capable of being operated without onsite or offsite power. Valve position indication is provided if necessary.

Reactor Coolant Inventory

An analysis was done to confirm that adequate reactor coolant inventory can be maintained for at least one hour during an SBO event. The analysis assumed the maximum Technical Specification leakage and 25 gpm from each RCP (four pumps).

Procedures

Existing plant procedures address AC power restoration, severe weather response, and the plant response to an SBO. These procedures will be updated as plant conditions change.

Modifications

Plant modifications were required to comply with 10 CFR 50.63. The major modification is the addition of one safety-related and one non-safety-related diesel generator to our onsite distribution system. These modifications were completed and are described in Section 8.4.

1.8.3 MAINTENANCE RULE

Title 10 CFR 50.65 "Requirements for Monitoring the effectiveness of Maintenance at Nuclear Power Plants" was issued on July 10, 1991. Utilities were required to be in full compliance by July 10, 1996. The purpose of the Maintenance Rule is to insure that structures, systems, and components of nuclear power plants are maintained such that plant equipment will perform its intended function when required.

Nuclear Energy Institute [formerly Nuclear Management & Resources Council (NUMARC)] formed a utility group to develop a guideline (NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) to assist utilities in implementing 10 CFR 50.65. NUMARC 93-01 was developed with input from the NRC and later endorsed by the NRC in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" as an acceptable method to meet the requirements of 10 CFR 50.65. Calvert Cliffs used NUMARC 93-01 to implement the requirements of the Maintenance Rule. Existing programs were used when available to meet the requirements of the Rule. New processes were developed and included in the appropriate program procedure to ensure compliance.

1.8.4 INDIVIDUAL PLANT EXAMINATION

In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded that existing plants posed no undue risk to the public health and safety. However, the Commission recognized that systematic examinations were beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. Therefore, each plant was requested to perform such a systematic examination of internally initiated events in Generic Letter 88-20 and report the results to the NRC. One of the purposes of the examination is to determine whether modifications to hardware and procedures were necessary to reduce the frequency of severe accidents or to mitigate their consequences. Supplement 1 to the Generic Letter provided additional guidance concerning the method to be used in the plant examination. Supplement 2 to the Generic Letter addressed severe accident management strategies that could be used in the plant examination. These strategies were developed by the NRC based on experience gained in reviews of probabilistic risk assessments. Calvert Cliffs was requested to evaluate these or similar strategies during our plant examination. Supplement 3 of the Generic Letter provided additional insights about the performance of pressurized water reactor containments. These insights could be used, if appropriate, during the individual plant examination. Supplement 4 addressed the need to evaluate plant vulnerabilities and response to external events. Risk assessments at that time indicated that the risk from external events could be a significant contributor to core damage in some instances. Finally, Supplement 5 to Generic Letter 88-20 provided updated guidance concerning seismic hazard estimates for many plants so licensees could determine the appropriate level of examination for their plants.

The NRC's purpose in requesting these evaluations was for licensees to: (1) develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at the plant under full-power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

In response to these requests, Calvert Cliffs performed an individual plant examination for internally initiated events. Evaluated events included (but are not limited to) transients, LOCAs, anticipated transient without scram, internal flooding, steam generator tube rupture, and interfacing system LOCAs. In addition, the examination searched for decay heat removal vulnerabilities. Based on that examination, the NRC concluded that the resolution of Unresolved Safety Issue A-45, Decay Heat Removal Reliability, was acceptable and closed this issue for Calvert Cliffs. The NRC also concluded that Calvert Cliffs' individual plant examination was complete and the results were reasonable given Calvert Cliffs' design, operation, and history. As a result, the NRC concluded that Calvert Cliffs' individual plant examination process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and Calvert Cliffs has met the intent of Generic Letter 88-20, including Supplements 1, 2, and 3.

Calvert Cliffs also performed an individual plant examination for externally initiated events. This examination included initiating events such as, fires, earthquakes, high winds, and other external events. As part of the plant examination process, a number of generic safety issues (GSI's) were identified and addressed. The NRC has reviewed the response for these GSI's and considers them resolved for Calvert Cliffs: GSI–103, Design for Maximum Probable Precipitation; GSI–57, Effects of Fire Protection System Actuation on Safety Related Equipment; GSI–147, Fire Induced Alternate Shutdown/Control Room Panel Indications; GSI–148, Smoke Control and Manual Fire-Fighting Effectiveness; GSI-156, Systematic Evaluation Program; and GSI–172, Multiple System Responses Program. The NRC found that the Calvert Cliffs individual plant examination for external events was complete and the results were reasonable given Calvert Cliffs' design, operation, and history. As a result, the NRC concluded that the integrated plant evaluation for external events process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and Calvert Cliffs has met the intent of Generic Letter 88-20, Supplements 4 and 5.

1.8.5 GENERIC LETTER 2008-01, MANAGING GAS ACCUMULATION

Gas accumulation in water systems can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Reference 106 to address the issue of gas accumulation in certain systems. The NRC requested that each licensee evaluate its Emergency Core Cooling System (ECCS), decay heat removal system (Shutdown Cooling [SDC] system), and containment spray (CS) system licensing basis, design, testing, and corrective actions to ensure that gas accumulation is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality are identified.

In Reference 107 the following systems were determined to be in the scope of the generic letter for Calvert Cliffs:

- Safety Injection (SI) system., i.e., ECCS
- SDC system
- CS system
- Relevant flow path in the charging system when used for High Pressure Safety Injection (HPSI) - not applicable to Unit 1 due to existing high point vents.

For the purposes of this issue, the term ECCS refers to the combination of SI and SDC systems. The relevant portion of the charging system includes that piping used for providing a HPSI flow path for the hot leg injection via pressurizer spray post-loss-of-coolant-accident (LOCA).

The following are considered gas intrusion mechanisms:

1) The formation of gas upstream of normally shut valves in SI discharge piping caused by leak-by of nitrogen saturated water from the safety injection tanks to the lower pressure SI discharge piping.
2) The “stripping” of gas out of solution due to leakage of the RCS/SI boundary check valves.
3) Gas coming out of solution due to dynamic pressure drops.
4) Gas intrusion due to human error.

In-leakage through vent valves, valve packing, mechanical pump seals, threaded pipe connections, and gasketed flanges, is not considered a valid source of gas intrusion since all piping in the scope of the GL is under positive gauge pressure (excluding the dry CS piping at higher elevations in the Containment).

References 108 and 109 acknowledged that no gas voids were found at any location during the confirmatory walkdowns and evaluations performed on the subject systems.

Reference 110 submitted a license amendment request to add Surveillance Requirements to verify that the locations susceptible to gas accumulation are sufficiently filled with water and to provide allowances which permit performance of the verification.

In Reference 111, the NRC issued a change to the TS to add new Surveillance Requirements to the TS for the SDC system, the ECCS and the CS system. These SRs require periodic verification that gas has not accumulated in the associated piping to a degree that would render the system inoperable.

1.8.6 RISK INFORMED CATEGORIZATION AND TREATMENT OF SSCS, 10 CFR 50.69

1.8.6.1 Introduction

10 CFR 50.69 provides a risk-informed process for classifying systems, structures and components (SSCs). In the traditional approach, SSCs are categorized as either “safety-related” (as defined in 10 CFR 50.2) or “nonsafety-related.” By applying risk insights, SSCs can be further classified as being either “safety significant” or “low safety significant.” This results in four Risk-Informed Safety Class (RISC) categories:

RISC-1: Safety-related SSCs that perform safety significant functions.
RISC-2: Nonsafety-related SSCs that perform safety significant functions.
RISC-3: Safety-related SSCs that perform low safety-significant functions.
RISC-4: Nonsafety-related SSCs that perform low safety-significant functions.

CCNPP Units 1 and 2 received approval to implement 10 CFR 50.69 by License Amendments 332 and 310 (Reference 112) on February 28th, 2020, in accordance with the methodology described in NEI 00-04 (Reference 113), as endorsed by NRC Regulatory Guide (RG) 1.201 (Reference 114).

1.8.6.2 Scope

This process can be applied to selected systems or structures and implemented over a period of time. Implementation is conducted on entire systems or structures, not selected components within a system. This ensures that all functions for an SSC within a system or structure are appropriately considered when determining safety significance. The following systems have been categorized per the 10CFR 50.59 process:

1) System 077/079: Area / Process Radiation Monitoring System (RMS)
2) System 024: Emergency Diesel Generators System (EDGs)
3) System 114: Post-Accident Monitoring System (PAMs)
4) System 060: Primary Containment HVAC System
5) System 032: Auxiliary Building & Radioactive Waste HVAC System
6) System 012: Salt-water Cooling System
7) System 052/061: Safety Injection System & Containment Spray System
8) System 030: Control Room HVAC System
9) System 015: Component Cooling Water System

1.8.6.3 SSC Categorization

Categorization of SSCs is done in accordance with NEI 00-04. The process consists of the following:

A. Risk Characterization

1. Use of PRA models to evaluate risk associated with internal events, including internal flooding and fire.
2. Apply shutdown safety assessment process (Reference 115) and the station shutdown risk management program to assess shutdown risk
3. Implement the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method (Reference 116) to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports
4. Use non-PRA evaluation methods that are based on the IPEEE Screening Assessment for External Hazards, except seismic (Reference 117).
5. Use alternative seismic approach as described in Exelon's original submittal letter (Reference 118)

B. Defense in Depth (DID) assessment to ensure that adequate redundancy and diversity are retained.

C. Risk Sensitivity Study using PRA methods based on a postulated change in reliability.

D. Review by an Integrated Decision-making Panel (IDP) to ensure appropriate considerations have been made for plant design, operating practices and operating experience.

1.8.6.4 Alternative Treatment

Safety-related SSCs are subject to a specific set of regulations (special treatment) that are not applicable to non-safety related equipment. Compliance with 10 CFR 50.69 provides an alternative to meeting the regulations identified in 10 CFR 50.69(b)(1) for RISC-3 and RISC-4 SSCs.

The performance of RISC-1 and RISC-2 SSCs are monitored to determine if adjustments to the categorization assumptions or treatment processes are necessary. This monitoring can be performed in the same manner as done for 10 CFR 50.65 (Maintenance Rule) except the monitoring addresses all functional failures, not just maintenance preventable functional failures. Since RISC-2 SSCs are non-safety related, enhanced treatment may be warranted to improve the reliability and availability of the SSC in support of its safety significant function.

RISC-3 SSCs must remain capable of performing their safety-related functions under design basis conditions, including seismic and environmental conditions. Periodic inspection and testing activities must be conducted to verify they will remain capable of performing their safety-related functions. Appropriate safety margins must be maintained for RISC-3 SSCs and any increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment must be small, within the sensitivity limits of the risk methods used in categorization.

RISC-4 SSCs have no safety-related or safety significant function. They require no Special nor Alternative Treatments.

1.8.6.5 Periodic Reviews

A periodic review is performed at least once every two refueling outages. The review includes evaluating changes to the plant, operational practices, plant and industry operating experience, SSC performance, impact of updated PRA and other factors that may affect SSC categorization and treatment. This review maintains safety margins for categorized SSCs and any increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment within the sensitivity limits of the risk methods used in categorization.

1.8.7 REFERENCES

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20. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated September 1, 1983, Detailed Control Room Design Review

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54. Letter from A. E. Lundvall, Jr. (BGE) to R. W. Reid (NRC), dated November 23, 1979, Follow-up Actions Resulting from TMI-2 Incident Auxiliary Feedwater System Modification

55. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated March 28, 1980, NRC Requirements for Auxiliary Feedwater Systems

56. Letter from J. C. Ventura (Bechtel) to R. F. Ash (BGE), dated October 16, 1981, Auxiliary Feedwater Design Criteria

57. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated April 21, 1981, Issuance of Amendment Nos. 53 and 36 to Facility Operating License Nos. DPR-53 and DPR-63

58. Letter from E. J. Brunner (NRC) to A. E. Lundvall, Jr. (BGE), dated January 15, 1981, Combined Inspection 50-317/80-16 and 50-318/80-15

59. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated September 3, 1993, Operator Examination Report Nos. 50-317/93-11 and 50-318/93-11, Reply to Unresolved Item Nos. 50-317/93-11-01 and 50-318/93-11-01

60. Letter from G. W. Meyer (NRC) to R. E. Denton (BGE), dated March 13, 1994, Combined Inspection Report 50-317/93-11 and 50-318/93-11

61. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BGE), dated February 1, 1982, Issuance of Amendment Nos. 65 and 47 to Facility Operating License Nos. DPR-53 and DPR-63

62. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated November 20, 1979, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned Short Term)

63. Letter from A. E. Lundvall, Jr. (BGE) to R. W. Reid (NRC), dated February 29, 1980, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned)

64. Design Calculations I-90-307 and I-90-119

65. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated December 7, 1981, Request for Amendments

66. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated June 29, 1984, Request for Amendment

67. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated April 9, 1984, Request for Amendment

68. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BGE), dated February 22, 1985, Issuance of Amendment Nos. 99 and 81 to Facility Operating License Nos. DPR-53 and DPR-63

69. Letter from A. E. Lundvall, Jr. (BGE) to A. C. Thadani, dated April 16, 1986, Request for Amendment

70. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BGE), dated August 6, 1986, Issuance of Amendment Nos. 120 and 102 to Facility Operating License Nos. DPR-53 and DPR-63

71. Letter from R. W. Starostecki (NRC) to A. E. Lundvall, Jr. (BGE), dated March 25, 1983, NRC Resident Inspector Report 50-317/83-05, 50-318/83-05

72. Letter from R. W. Starostecki (NRC) to A. E. Lundvall, Jr. (BGE), dated June 22, 1982, Combined Inspection 50-317/82-12, 50-318/82-10

73. Letter from H. R. Denton to All Reactor Licensees, dated October 30, 1979, Discussion of TMI Lessons Learned Short-Term Requirements

74. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated December 1, 1984, Regulatory Guide 1.97 Review

75. Letter from A. E. Lundvall, Jr. (BGE) to E. J. Butcher, Jr. (NRC), dated July 1, 1985, Inadequate Core Cooling Instrumentation

76. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BGE), dated February 9, 1987, Safety Evaluation of the Inadequate Core Cooling System for Calvert Cliffs Units 1 and 2

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78. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated January 28, 1982, Pressurized Thermal Shock

79. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated June 5, 1984, NUREG-0737, II.K.2.13, Thermal-Mechanical Report

80. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated October 19, 1983, Natural Circulation Cooldown

81. Letter from J. R. Miller (NRC) A. E. Lundvall, Jr. (BGE), dated October 24, 1983, Review of NUREG-0737 Item II.K.2.17, Voiding in the Reactor Coolant System During Transients

82. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated August 11, 1981, Response to NUREG-0737, Item II.K.3.1

83. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated October 12, 1983, NUREG-0737, Item II.K.3.1 – Automatic PORV Isolation and II.K.3.2 – Report on PORVs for Calvert Cliffs Units 1 and 2

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85. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated July 25, 1983, Automatic Trip of Reactor Coolant Pumps

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91. Letter from E. J. Butcher (NRC) to A. E. Lundvall, Jr. (BGE), dated June 28, 1985, NUREG-0737, Item II.K.3.30 Closure

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94. Letter from R. C. L. Olsen (BGE) to R. A. Clark (NRC), dated December 3, 1982, Emergency Operations Facility

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