



CHARLES CENTER • P. O. BOX 1475 • BALTIMORE, MARYLAND 21203

GEORGE C. CREEL
VICE PRESIDENT
NUCLEAR ENERGY
(301) 260-4455

March 1, 1991

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Report of Changes, Tests, and Experiments

REFERENCE: (a) 10 CFR 50, Paragraph 50.59(b)

Gentlemen:

As required by the above reference, please find enclosed our annual report of changes, tests, and experiments completed on Calvert Cliffs Unit 1 and/or 2 under the provisions of 10 CFR 50.59(a), including a summary of the safety evaluation for each. This report covers the period from January 1, 1990 through December 31, 1990.

Items in the report are referred to by Facility Change Request (FCR), Field Engineering Change (FEC), Temporary Modification or Miscellaneous Activity number.

Should you have any questions regarding the contents of this report, we will be pleased to discuss them with you.

Very truly yours,

GCC/JBB/BSM:d1m

Enclosure 1: Annual Report of Changes, Tests and Experiments (20 pages)

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R. A. Capra, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
L. E. Nicholson, NRC
R. I. McLean, DNR
J. H. Walter, PSC

bcc: f. s. Tiernan/A. J. Slusark
Miernicki
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C. H. Cruse/P. E. Katz
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ENCLOSURE (1)

ANNUAL REPORT OF CHANGES, TESTS AND
EXPERIMENTS

CALVERT CLIFFS NUCLEAR POWER PLANT

1990

FCR 82-79

FCR 82-79 moved power cables for #11 Containment Air Cooler, but did not reflect the change on FSAR Figure 8-3. This Safety Evaluation was performed to incorporate the as-built change from penetration 12EB1 to 12EA4 for #11 Containment Air Cooler on FSAR Figure 8-3. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 84-1086

This change replaced the existing unqualified thermocouple assembly, signal converter and temperature indicator for Unit 2 Containment Dome Temperature with environmentally qualified units in accordance with Reg. Guide 1.97, Category 2 requirements. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 85-1022 Supplement No. 15

This change removed the tubing and piping that was abandoned in place after removal of the original screen wash system. In addition, two cables that were abandoned in place will be retagged as spares. This activity does not affect any safety related equipment. However, because it is a change to FSAR figures 9-23-2 and 9-28A, it required a Safety Evaluation. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 87-69 Supplement 2

This activity is necessary to correctly identify the containment isolation valves for penetration Nos. 9 and 10 and to bring the FSAR into agreement with the Technical Specifications.

This activity revises FSAR Figure 5-10 Sheet 8 to correctly identify the containment isolation valves for penetration Nos. 9 and 10 and to further clarify this in Section 5.2.2. This Figure identified valves SI-326, CV-4151, and CV-4160 for penetration No. 9 and SI-316, CV-4150, and CV-4159 for penetration No. 10 as being the respective penetration's containment isolation valves. Also, Section 5.2.2 will be revised to further clarify that it does not apply to penetrations Nos. 9 and 10 since the containment spray system is an accident mitigation system which will not be isolated post-accident until containment pressure is reduced or a failure of the spray system occurs. Per Technical Specification Table 3.6-1 and the following discussion, the correct isolation valve numbers for penetration No. 9 are SI-326 and SI-340 and for penetration No. 10 are SI-316 and SI-330. During normal plant operation, check valves SI-326 and SI-340 for Penetration 9 and SI-316 and SI-330 for Penetration 10 remain closed as there is no flow in the containment spray lines. For both Penetrations 9 and 10, the isolation valves for the spray

header (CV-4150, CV-4151) and the charcoal spray system (CV-4159, CV-4160) are both located downstream of check valves SI-340 for Penetration No. 9 and SI-330 for Penetration No. 10 and are normally closed. Upon receipt of a Safety Injection Actuation Signal (SIAS), valves CV-4150 and CV-4151 open to allow flow to the spray ring nozzles. These two valves also fail open upon loss of power or air. In addition, valves CV-4159 and CV-4160 are opened by a handswitch and also fail open on a loss of power or air. With CV-4150 and CV-4151 open after a SIAS, the check valves provide containment isolation since the spray headers are open to the containment atmosphere. Technical Specification Table 3.6-1 correctly lists SI-326 and SI-340 for Penetration No. 9 and SI-316 and SI-330 for Penetration No. 10 as the containment isolation valves. FSAR Figure 5-10 Sheet 8 and Section 5.2.2 will be revised to reflect this. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 87-129

This change repowered the 72 foot computer room air conditioning units such that a minimum of one unit will be operable during a Loss of Offsite Power (LOOP). This change included new power cable installation from the computer room to switch gear rooms. The reason for the change is because the Safety Parameter Display System (SPDS) must be available during normal and abnormal events including LOOP as required by NUREG-0737 Supplement 1. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 88-11 Supp.5

As described in the FSAR, the original chlorination design, was to inject a chlorine solution into the intake weir just downstream of the traveling screens. The chlorine concentration was controlled at this point rather than at the plant effluent as required by the NPDES permit.

In order to meet the requirements of the NPDES permit, it was decided in 1976 to redirect the chlorination system to serve only the Salt Water (SW) system. Although the physical changes to the plant have been made, the one paragraph description of the Chlorination system in the FSAR has not been revised to reflect these changes. Additionally, after reviewing the safety analyses in FCR 76-99 and associated FEC's, although adequate for that time period, it is prudent to revisit the changes made under FCR 76-99. Therefore, this 80.59 evaluation was written to address the overall differences between the FSAR description and the as-built Chlorination system. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 88-031

This change replaced the Durametalllic mechanical seal on Low Pressure Safety Injection (LPSI) pumps 11, 12, 21, and 22 with a Borg-Warner cartridge type seal assembly due to unacceptable amounts of leakage past the shaft. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 88-3000-1

During the refueling outage following completion of Unit 1 cycle 9, all the reload fuel for cycle 10 operation were examined. As a result, four fuel pins were removed and replaced with stainless steel dummy rods. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 88-3002

The previous maximum enrichment limit for the fuel handling equipment (fuel upender, fuel inspection elevator, and new fuel elevator) was 4.1 wt% U-235. The fuel handling equipment was upgraded to 5.0 wt% U-235 to enable movement of higher enriched fuel assemblies between the spent fuel storage racks and the reactor core, and the new fuel storage racks and the spent fuel pool. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 89-089 Supplement 5

Boric Acid deposits were discovered at several pressurizer heater penetrations on Unit 2. It was established through destructive and non-destructive examinations that these sleeves were leaking as a result of Primary Water Stress Corrosion Cracking (PWSCC). This change is to replace Unit 2 Pressurizer Heater Sleeves except at penetration H-3. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-089 Supplement 6

This change plugged Unit 2 Pressurizer heater penetration H-3. This evaluation only addresses the mechanical engineering aspects of this change. Boric acid deposits were discovered at several penetrations. A large diameter core sample was bored out from the pressurizer at penetration H-3 to include the weld, and destructively examine it for root cause analysis. Heater reinstallation may encounter alignment problems, therefore it was decided to plug the penetration. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-089 Supplement 7

This modification changed the power supply to the pressurizer heater at location C3 from Non-1E MCC 210PH to a diesel backed (1E) power supply to the heater at location H3 and reconnecting it to the heater at location C3.

This change was necessary as pressurizer heater penetration H3 was permanently plugged during the Unit 2 pressurizer repairs. In order to maintain a redundant 450 kw heater capacity powered from a 1E power supply (reference letter from R. Reid - NRC to A. Lundvall - BG&E dated April 7, 1980) after the present 1E heater location H3 is permanently plugged, the pressurizer heater in location C3 must be powered from MCC 211PH, instead of MCC 210PH which is non-1E. This change will not place any additional loading on the diesel generator as heater C3 requires 12.5 kw which were the same power requirements as heater H3. The Safety Evaluation concludes there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-98

This change was made to replace the Unit 1 and 2 Safety Injection Tank "non-code" relief valves with new Crosby Model JOS ASME Section VIII relief valves. The FSAR states that the SITs were designed, constructed, and have overpressure protection in accordance with the ASME Code, Section III, Class C. Additionally, the nitrogen relieving capacity of the Safety Injection Tank relief valves, were insufficient if the nitrogen supply line control valve would fail in the open position. The new Section VIII relief valves satisfy the FSAR code requirements. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-0171

This modification will eliminate an unwanted shutdown of Diesel Generator #11 from a short circuit caused by a postulated control room fire. This is in agreement with the Diesel Generator #12 and Diesel Generator #21 stop circuits isolation feature initiated by FCR 81-1052. In addition, the requirements of AOP-9A (Control Room Evacuation and Safe Shutdown due to a Severe Control Room Fire) are also being met. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 89-180 Supplement 2

Many Saltwater Control Valves must remain fully operational post-LOCI. During performance of an Engineering Test Procedure, it was discovered that many of the air operated control valves which utilize Safety Related (SR) air accumulators would not have performed as expected after a loss of normal Non-Safety Related (NSR) air supply. Of specific concern are the valves which are inaccessible post-LOCI and are required to align the Saltwater System to the overboard mode of operation upon a rupture of a Saltwater Discharge line after a Recirculation Actuation Signal (RAS) occurs. This change upgrades electrical circuits, supplies SR air to control valves, provides a cross tie between SR and NSR air, and modifies air tubing. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 89-139

A 1977 LTOP commitment stated that both PT-103 and PT-103-1 (pressurizer pressure transmitters) would provide a computer generated high pressure alarm and type written printout. This modification allowed PT-103 and PT-103-1 to provide the required alarm and printout function. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-3001

This change removed the following sentence from the UFSAR, "At no time during the transfer from the reactor core to the spent fuel storage rack is there less than 112 inches of water above a fuel assembly." This sentence was incorrect, had no basis and unnecessarily restricted fuel assembly movement. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 90-10

The text of the UFSAR, in a number of cases, is too vague to adequately define separation of cables and raceways in redundant separation groups. This evaluation was written to modify Section 8.5 in the UFSAR, allowing the use of evaluations when determining acceptable separation. The probability of failure will not increase by evaluating the actual separation required. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-010 Supplement 3

The FSAR implies that only silicate separation barriers are used for raceway/cable tray separation in the plant. This is not the case as sheet metal tray covers are used also. This change will clarify the FSAR, to specify which materials may be used as separation barriers. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FCR 90-011

The Component Cooling Water (CCW) flow-rate to the Shutdown Cooling Heat Exchanger (SDCHx) was reduced during all modes of operation, due to previous manual valve position causing heat exchanger tube rattling during normal shutdown cooling operation. Performance Engineering determined that at shell side flows of 3000 gpm, considerable tube to tube and tube to baffle contact occurs, and increases in frequency and intensity as flow increases. Engineering Test procedure 89-77 set the 11 and 12 SDCHx inlet manual valves to provide a flow of 2500 gpm to each heat exchanger when two CCW pumps are operating and all major CCW loads (CCW flow through the shell side of the SDCHx) are secured (condensent, evap, rator, letdown). With this position held, the CCW system was aligned for normal and accident operation. Given a LOCA with a LOOP and one DG failure, the flow through each SDCHx is 1800 gpm and this flow rate was used in the LOCA analysis.

As a result of the change in CCW flow to the SDCHxs, two calculations were performed that affect information in the FSAR.

1. LOCA
The LOCA analysis was reviewed to determine the post-RAS containment response given a CCW flow of 1800 gpm. The results show that it will take a longer period of time for containment to be brought back to 120°F (<30 days). The EQ group of EAU was able to demonstrate acceptable qualification for a 30-day cool-down time period (see EQDR 14, Rev. 1).
2. Normal Cool-down
The FSAR states that the RCS can be brought from 300°F to refueling temperature in 27 1/2 hours after shutdown. Because of the reduced CCW flow, the cool-down extends to 36 hours. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-21

Prior to 1986, the manufacturer's technical manual did not specify system accuracy for the Delphi Model K-IV Hydrogen Analyzers. Then in 1986, the manufacturer's representative established a system accuracy of 5 percent of full scale (0.5 percent H₂) which was then documented in a revised technical manual (12-277-24). During surveillance testing of the Hydrogen Analyzers, required by Technical Specifications 4.6.5.1 and 4.6.5.2, maintenance could not achieve the recorder or indicator loop accuracy stated in BG&E March 14, 1983 letter to the NRC. Therefor, calculation I90-16 was performed by I&C Engineering to determine the total loop uncertainty (accuracy) associated with the recorders and indicators of the hydrogen analyzers using present day methodology.

The methodology used in this calculation is conservative in that little use of square root sum of the squares (SRSS) is used. A margin above STP tolerance is allowed without approaching the calculated loop uncertainty.

Loop Uncertainty for Indicator Loop = $\pm 10.26 \%$ (1.0% H₂)
Loop Uncertainty for Recorder Loop = $\pm 9.40 \%$ (0.94% H₂)

The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-036

The previously installed Unit 1 and 2 SG blowdown tank RVs had insufficient capacity. This change replaced the SG blowdown tank RVs with new Crosby style JBS RVs to comply with Section VIII of the ASME code. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-64

This change enhanced the reliability of the steam driven Auxiliary Feedwater System by allowing the turbine governor to accelerate the AFW pump in a more controlled manner in order to eliminate turbine feed pump trip due to initial over speed. The change modifies the control schemes of the AFW Main Steam Admission Valves (MSAV) 1-CV-4070 and 1-CV-4071. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-68

This activity accepted the "as-found" plant condition for the loose Calvert Cliffs Unit 2 core support barrel (CSB) snubber pins. During the 1989 outage, a visual examination was performed on the core stabilizing lugs. This examination revealed that two of the snubber pins were partially sticking out. The first pin is located in the snubber assembly at the 120° location and was sticking out approximately 3/8". BG&E attempted to pull the pin out using a robot submarine. The submarine accidentally bumped into the pin, knocking the pin back into the hole. The pin is now in its proper location. A second questionable pin is located in the snubber assembly at the 180° orientation. It appears as if this pin is slightly out of its proper position. There is a possibility that the small locking pins located in the CSB snubbers in the reactor vessel downcomer may loosen and fall into the flow-stream. This "as-found" plant configuration introduces concerns with loose particles in the reactor coolant system (should the pin come completely out) and increases the possibility that a shim retaining bolt can loosen and back out. In conclusion the Safety Evaluation showed that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. The Safety Evaluation also concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-87

This change revises FSAR Section 9.4 to allow the four removable Spent Fuel Pool (SFP) spool pieces to remain installed in the SFP system during all modes of operation. FSAR Section 9.4 refers to the spool pieces as temporary tie-ins to be used as a means of augmenting the heat removal capacity to SFP cooling when 25/3 cores are in the pool. The design intent for the temporary spool pieces was to keep the systems separate while allowing for hook-up for a very rare occurrence of cooling augmentation. This independence can be satisfactorily achieved via isolation valves without having to remove the spool pieces, thereby eliminating a valid ALARA concern. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-095

This activity resolved minor inconsistencies between the defined quality standards for field erected tanks as described in FSAR Section 6.3.5.1 and the quality standards imposed in the purchase specification for the field erected tanks. Review of FSAR Section 6.3.5.1, applicable P&IDs, related documents and applicable specifications indicated the change to the FSAR was necessary and no other hardware problems exist. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-146

This change was necessary to show in the FSAR the corrected postulated dose rates from Units 1 and 2 Emergency Personnel Air Locks and to include the corresponding information for the Units 1 and 2 equipment hatches. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 85-1041-37

This modification removed differential pressure indicator 2-PDI-2420 including all of the associated process tubing and supports for the indicator. This was necessary as the differential pressure indicator and process tubing interfered with the installation of Main Steam line support 20" EB-3-2002-R14. These components are no longer in service and removing them will facilitate the installation of this main steam line support. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 87-0074-011

This change rerouted Unit 2 Reactor Coolant Pump vapor seal leak off lines from the Reactor Coolant Drain Tank (RCDT) to the containment sump. This prevents spray at the seal due to back pressure at the tank. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 89-01-149

This change replaced the existing shaft packing seal with a mechanical seal in Condensate Pump No. 11. Minor piping modifications were also done to accommodate this change. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 89-01-473

This change installed drain lines off existing drain taps on all Emergency Core Cooling System (ECCS) basket strainers. This will allow draining of the strainers to the Saltwater system instead of into floor drains, thereby lessening the amount of liquid waste. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 89-03-35

This activity approved the change of Salt Water pump component materials. This was requested by the vendor due to market availability. Some of the components involved included, impeller shaft, impeller shaft nut, impeller shaft sleeve and thrust bearing shaft ring. This activity found the substitute materials acceptable. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 89-98-1 and 2

This activity relocated the safety relief valves 1(2)-RV-211, 221, 231, and 241 for all the Safety Injection Tanks (SITs) on Units 1 and 2. This was required because of the failure of seal welds at pipe connections for the SIT vent lines upstream of 1-RV-221 and 241. The relief valves were relocated from the SITs vent lines to the nitrogen fill lines upstream of the SIT, near the base of the tank. This will reduce the local stresses on the vent lines for the SITs where the relief valves were attached. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-17

This change replaces the body of 2-CV-5209 and the two flanged spool pieces in the 30" piping downstream of the service water heat exchanger salt water discharge throttle valve, 2-CV-5210, with a single, rubber lined spool piece.

The salt water system was sized for a loss of coolant accident. During normal operation, the flow through the service water heat exchanger is throttled at the heat exchanger discharge control valves 2-CV-5210 and 2-CV-5212. The turbulent flow associated with throttling these butterfly valves caused erosion of the valves and the downstream piping. To mitigate this effect, orifice valves, 2-CV-5209 and 2-CV-5214, were installed downstream of the throttle valves (FCR 75-1100). These valves could be opened to provide full flow during accident conditions and closed during normal operations to move the turbulence away from the discharge control valves. Unfortunately, erosion was even more severe after installation of the new valves.

FCR 80-0017 removed the internals and actuators of 2-CV-5209 and 2-CV-5214 and installed rubber lined spool pieces downstream of the discharge control valves. FCR 82-1012 replaced valves 2-CV-5210 and 2-CV-5214 with rubber lined valves. These modifications appear to have resolved the localized pipe erosion problem. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-028

While working on one of the Main Feed Regulating Valves (MFRV) it was determined that some washout of the valve body had occurred. This washout was repaired by a weld buildup. A radiographic examination (RT) of the repair weld was required but was not possible to perform on the entire weld build-up due to the non-uniform valve geometry. This Safety Evaluation was performed so that a magnetic particle (MT) examination could be done instead. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-143

The installed Unit 1 and 2 SG blow-down tank R.V.s had insufficient capacity to satisfy Section VIII of the ASME code. The procurement and installation of a new SG blow-down tank R.V. could not be accomplished in a timely manner. BG&E opted for a temporary fix, installing flow limiting orifices in the SG blow-down lines to limit the mass flow rate to the blow-down tank. Limiting the mass flow-rate to the tank will ensure the blow-down tank will not become overpressurized and will retain its structural integrity. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-173

Installation of isolation valves upstream and downstream of the boundary check valve 2-IA-310 has been completed. Also, a pressure point consists of teeing off of the main line with an isolation valve and a drain. This change was made because it is required to periodically test all safety-related and non-safety related boundary check valves. By adding these isolation valves and pressure points the check valve 2-IA-310 can be tested with minimum impact to the plant. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-189

Logic diagrams 61058A (1LD58A) and 63058A (2LD58A) were revised to more accurately reflect SIAS and RAS operation for Component Cooling and Service Water Heat Exchangers' salt water valves 1(2) CV-5160, 5163, 5206, 5208, 5210, 5212. These revisions to the logic diagrams reflect information from the text of FSAR Section 9.5 and existing schematics. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-238

This activity added low point drain lines to the Unit 2 feedwater recirculation lines, upstream of the condenser. These low point drains are necessary to allow for draining and flushing of the feedwater piping which the condenser is unavailable and isolated. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-250

Auxiliary Boiler Steam system steam trap O-ST-1855 was modified from two inches to one inch size due to excessive, unnecessary loss of steam, reducing system efficiency. The Auxiliary Boiler Steam system is a non-safety related system. However, this FEC requires a change to FSAR figure 10-6 and therefore, this Safety Evaluation was performed. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-265

This change changes BG&E drawing 60-227-E to reflect manual drain and vent valves 1-BD-187, 188, 189, 190, and 191 and isolation valves 1-BD-192 and 193 in Unit 1 Steam Generator Blowdown and Recovery System piping. All of the valves are in Non-Safety Related piping and not described in the text of the FSAR. However, this Safety Evaluation is required because 60-227-E is shown in the FSAR. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-1-298

This FEC proposed the replacement of the original "G" type buffer spring with a vendor recommended stiffer "F" type buffer springs, in the Auxiliary Feedwater (AFW) system turbine governor compensatory unit. The replacement buffer springs are proposed for both Unit 1 and 2 AFW system turbine governors. The "F" type buffer spring will help dampen the sudden minor load changes experienced by the governor, but allow it to respond to major load changes to govern the turbine speed. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-406

This activity involves updating Unit 1 elevation 12 feet Turbine Building location drawing (FSAR Figure 1-20) to reflect the as-built location of the Mechanical Chiller for the Turbine plant samples and removal of the chemistry lab. This activity is non-safety related and does not affect any safety related equipment, but a Safety Evaluation is necessary since the activity requires a revision to FSAR figure 1-20. The Safety Evaluation concluded

there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-491

This change removed and reinstalled the air components on the Unit 1 feedwater regulating control valves. The air supply components were retubed using flexible hose installed between the control valve and solenoid valves which are mounted on a support frame. The reason for this activity is due to two (2) incidences where a loss of instrument air was due to shear and fatigue caused by the vibration of the feedwater regulating valves. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-537

This change replaces the body of 1-CV-5209 and the flanged spool piece in the 30 inch piping immediately upstream of 1-CV-5209 with a single, rubber lined spool piece.

The salt water system was sized for a loss of coolant accident. During normal operation, the flow through the service water heat exchangers is throttled at the heat exchanger discharge control valves 1-CV-5210 and 1-CV-5212. The turbulent flow associated with throttling these butterfly valves caused erosion of the valves and the downstream piping. To mitigate this effect, orificed valves, 1-CV-5209 and 1-CV-5214, were installed downstream of the throttle valves (FCR 75-1100). These valves could be opened to provide full flow during accident conditions and closed during normal operations to move the turbulence away from the discharge control valves. Unfortunately, erosion was even more severe after installation of the new valves.

FCR 80-0017 removed the internals and actuators of 1-CV-5209 and 1-CV-5214 and installed rubber lined spool pieces downstream of the discharge control valves. FCR 82-1012 replaced valves 1-CV-5210 and 1-CV-5212 with rubber lined valves. These modifications appear to have resolved the localized pipe erosion problem.

The flange on the second spool piece between 1-CV-5210 and the body of 1-CV-5209 was leaking. Replacing this spool piece and the body of 1-CV-5209 with a single spool piece stopped the leakage and returned the piping run closer to its original configuration. A rubber lined spool piece was used to mitigate the erosion effects associated with 1-CV-5210 and to provide good corrosion resistance. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-1-578

This change will delete level transmitters 2-LT-311A, 321A, 331A, and 341A from FSAR figure 6-10A. These level transmitters were never installed because of problems associated with purchasing level transmitters compatible with the Safety Injection Tanks existing instrument taps. The Safety Injection system, as described in the FSAR is not affected. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-579

This change involves updating the Unit 2 Instrument Air System Piping and Instrument Diagram (P&ID) to show the as built location of the Instrument air headers which supply MSR Drain Tank 23 and First Stage Reheater Drain Tank 23 pneumatic level control instruments. A safety Evaluation was necessary since the change creates a revision to P&ID M-454 Sh. 2 which is FSAR Figure 9-28A. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

FEC 90-01-819

This change is to allow the design pressure of the feedwater piping, between the containment penetrations and Steam Generators to be reduced from 1500 psig to 1400 psig. The design pressure of 1500 psig for the feedwater piping between the containment penetrations and the Steam Generators is very conservative. This statement is based on the fact that the Steam Generators have pressure relief via the Main Steam Code Safety Relief Valves, which have staggered settings ranging from 1000 psig to 1050 psig. The maximum Steam Generator pressure as found in Chapter 14 of the FSAR, is 1074 psig which occurs during an Asymmetric Steam Generator Event. These are the worst case pressure transients for the feedwater pipe in containment. At normal operating conditions the Steam Generator sustained pressure is about 850 psig. This is the pressure that meets the code definition of Design Pressure, which is the maximum sustained pressure. The pressure drop as given in the original design calculations from the Steam Generator to the flow element is about 40 psi, for a pressure at the FE's of 890 psi. Therefore, a reduction in the Design Pressure to 1400 psig will still envelop the worst case, and design conditions, will not compromise the safety or design basis of the feedwater line. However, this change will allow us to extend the useful service life of the feedwater flow elements which have been found to have wall thinning due to erosion / corrosion of the interior surfaces. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Temporary Modification # 1-90-61

The previously installed Unit 1 and 2 SG blowdown tank RVs had insufficient capacity. The capacity requirements, which must be satisfied, are specified by Section VIII of the ASME Code.

The procurement and installation of a new SG blowdown tank RV could not be accomplished in time to support the April 1990 Unit 1 plant start-up schedule. Thus, BG&E opted for a temporary fix. Flow limiting orifices (1-FO-4017, 4018) were installed in the common SG blowdown lines via FEC 90-01-143. Additionally, CCI-117 Serial #1-90-26 was issued to chain shut the surface blowdown manual isolation valves (1-BD-194, 196, 198, 200). Both activities were implemented so as to limit the maximum mass flowrate that could be supplied to the blowdown tank. Limiting the mass flowrate to the tank will ensure that the blowdown tank will not become overpressurized and will retain its structural integrity.

Unit 1 was shut down for the 1990 Eddy Current Outage. During this outage, the steam generators were drained via the SG blowdown lines. The installed orifice plates overly restrict the draining of the steam generators. Therefore, the orifice plates were replaced with non-flow restricting spacer plates.

The SG blowdown line orifices limit the mass flowrate to the SG blowdown tank so that the tank does not become overpressurized. Removing the orifices from the blowdown lines means that the blowdown tank will no longer have adequate overpressure protection. To ensure that the tank was not placed back in service without overpressure protection, this temporary modification was a Unit 1 mode restraining activity because it limited the RCS temperature to less than 200°F. This temperature limit prohibited the plant from entering mode 4 and ensured that the blowdown tank pressure remained less than its 200 psig design pressure. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

Temporary Modification 1-90-130

This temporary modification was to suppress nuisance alarms on Channel A of the Unit 1 Reactor Vessel Level Monitoring System (RVLMS) resulting from an inoperable heated junction thermocouple (HJTC) on sensor number eight. The control room trouble and low level alarm remain in alarm. The temporary modification substitutes HJTC number seven for number eight, to suppress the alarms. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Temporary Modification 1-90-146

This temporary modification was to suppress nuisance alarms on Channel A of the Unit 1 Reactor Vessel Level Monitoring System (RVLMS) resulting from an inoperable heated junction thermocouple (HJTC) on sensor number five. The Control Room trouble and low level alarm remain in alarm. The temporary modification substitutes HJTC number six for number five, to suppress the alarms. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Temporary Modification 1-90-175

Instrument channels 1-PT-103 and 1-PT-103-1 monitor pressurizer pressure and provide numerous automatic signals to various plant components. Unit 1 has experienced transient noise on these instruments. Transient noise may cause 1-SI-651-MOV and 1-SI-652-MOV to automatically close and isolate shutdown cooling. This temporary modification was used to temporarily defeat this automatic actuation signal. A dedicated control room operator was used to duplicate the function normally performed by the automatic signal. This 50.59 Safety Evaluation documents the fact that Calvert Cliffs was temporarily in a plant configuration that did not coincide with the FSAR. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Temporary Modification 2-90-61,62

This Safety Evaluation allows the generic installation of temporary blanks or blind flanges in the Salt Water system. These blanks or blind flanges will allow valves and/or spool pieces in the Salt Water System to be removed for various maintenance activities and still allow portions of the system to remain operable. Installation of blanks or blind flanges will be used to return the Salt Water System to a condition that satisfies the operability requirements of the Technical Specifications while the affected valve or spool piece is removed from the system. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

DCR 90-472

This activity is a drawing revision to an FSAR piping and instrument diagram (P&ID). M-36 sheets 1, 3, and 4, will be modified to show additional drain and drain valve details for the Main Steam/Reheat systems. The drain line details being added to the P&ID have always existed in the plant, but were just never explicitly shown on the FSAR P&ID. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

Setpoint Change I-90-41

This Safety Evaluation was written to incorporate setpoint change I-90-41. 12 B Reactor Coolant Pump (RCP) was experiencing vibration levels of 21 mils unfiltered, causing a hanging alarm in the Control Room. Increasing the current alarm setpoints from 18 mils alert, 20 mils danger to 24 mils alert, 26 mils danger, cleared the alarm and allowed operations to detect any degradation in all four RCPs. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-1-045-032-R0

NCR 8584 was issued to document the concern that High Pressure Feedwater Heaters 16 A/B have inadequate tube side over pressure protection per ASME Section VIII. The Feed System relies on the Main Feed Pump High Discharge Pressure Trip to prevent 16 A/B from an overpressure condition. FSAR Section 10.2.3 states that the heaters are designed to ASME Section VIII. However, ASME Section VIII - UG-125 does not allow crediting the main feed pump trip. Therefore, High Pressure Feedwater Heaters 16 A/B are not in compliance with ASME Section VIII. Thus, operation of 16 A/B is prohibited prior to reconciling this discrepancy with both the State of Maryland and the FSAR. This safety evaluation was performed to allow operation of the Feed System with 16 A/B High Pressure Feedwater Heaters isolated and bypassed. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specification.

Miscellaneous Activity 50.59 #90-0-027-037-R0

The purpose of this evaluation is to document the allowed radioactive contamination limits for the Auxiliary Boiler System. Since the auxiliary boiler provides essential functions, it is appropriate to establish allowed contamination limits, so that operation of the system may continue following a possible contamination event. The limits provide a limit of allowable contamination levels for normally non-contaminated systems wherein continued operation is acceptable. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specification.

Miscellaneous Activity 50.59 #90-B-052-039-R1

This change involves the use of the Containment Spray (CS) pump to provide core flush and long term cooling after seven days following a LOCA. The LPSI pump is limited in operation by the effect of radiation on the Teflon O-ring between the gland ring and pump body. Based on radiation levels at the -10 foot elevation of the auxiliary building reaching acceptable levels in order to make valve realignments, the switch to CS pumps can be made after seven days, but must be done at the beginning of day ten. The Safety Evaluation concluded there was no unreviewed safety question or change to the Technical Specification.

Miscellaneous Activity 50.59 90-2-064-056-R0

Mode 2 defines reactor start-up in which criticality is achieved by withdrawing the control rods. Further reactivity control is provided by changing RCS boron concentration. During the transition from Mode 2 to 100 percent power, RCS boron concentration is reduced substantially. This is accomplished by a dilution process that requires a significant reactor coolant change-out. This dilution process is completed in approximately 48 hours. Maintaining RCS Lithium Concentration between 1.0 -2.0 ppm during this dilution process would require excessive adds of LiOH. This activity allows Chemistry to delay the addition of LiOH until this dilution process is complete (provided a minimum Li concentration of .2 ppm is maintained). This will reduce the distraction to plant operators during the start-up evolution. This also reduces the amount of LiOH used and the number of adds required to maintain RCS Lithium concentration. This is viewed as a benefit because LiOH is a caustic in the concentrated form which constitutes a personnel safety hazard. The proposed activity will not have a negative impact on RCS components. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-B-027-088-R0

As a result of changes to the Alternate Safe Shutdown Procedures (AOP-9), eight hour emergency lighting is not provided for all equipment required to be operated for the procedures nor access routes there to. This eight hour emergency lighting is required by Appendix R of 10 CFR 50 Section III. J. This Safety Evaluation concluded there was no unreviewed safety question due to the use of compensatory measures until full compliance can be achieved or did it require a change to the Technical Specifications.

Miscellaneous Activity 50.59 #90-1-064-119-R0

This activity allowed Reactor Coolant Pump (RCP) 12A to be operated with the seals first two stages failed, for up to fifteen minutes with Reactor Coolant System (RCS) pressure less than 330 psig and temperature less than 180°F. The reason for this activity is that RCP 12B had indication of high vibration during the initial bump and vent. It was desirable to run the pump for several minutes to see if vibration levels would decrease. The Operating Curve requires two pumps in the same loop to be operating. Therefore, allowing the operation of 12A RCP allowed vibration data to be obtained on RCP 12B. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-B-037-120-R2

The purpose of this evaluation is to document the procedure and criteria used to establish allowed radioactive contamination limits for the Condensate Storage Tanks (CST). The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-B-045-129-R0

This activity evaluated the erosion of the thermal sleeve of the Steam Generator Feedwater Nozzles for the Unit 1 and Unit 2 steam generators and how this applies to the susceptibility of the nozzle to sustain a water hammer event. This activity also considers the effects of thermal fatigue and loose parts due to thermal sleeve erosion in the steam generators. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-B-041-131-R0

This activity changes FSAR Table 10A-5, Instrumentation Required to Place the Plant in a Safe Shutdown Condition and Maintain It in a Safe Shutdown Condition, revising the category of some instrumentation for a critical crack in the Sample System or a letdown line break in the Chemical and Volume Control System. The components affected by this change are: 1) 1-FT-212 and 1-FT-212 in the Charging Pumps discharge header; 2) 1-FT-332 and 1-FT-342 which measure the Low Pressure Safety Injection (LPSI) flow to loops 12A and 12B; 3) 1-MOV-616, 1-MOV-626, 1-MOV-636, and 1-MOV-646 which are located in the High Pressure Safety Injection (HPSI) lines to loop 11A, 11B, 12A, and 12B respectively; 4) 1-FT-331 and 1-FT-341 which measure HPSI flow to loop 12A and 12B respectively; and 5) 1-MOV-617, 1-MOV-627, 1-MOV-637, and 1-MOV-647 which are located in the Auxiliary HPSI lines to loops 11A, 11B, 12A, and 12B respectively. The affected components are being revised from Category B (qualified to be operated in a steam environment or not adversely affected by jet impingement) to Category C (not required for a break in this system). The

Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.

Miscellaneous Activity 50.59 #90-B-024-156-R2

The main objectives of this activity are to establish proper steps for a procedure for restoration of Air Conditioning (A/C) Unit No. 12 in the Main Control Room after loss of Control Room A/C Unit No. 11 due to failure of the A/C unit or failure of Diesel Generator (D/G) No. 11 after a loss of offsite power to both Units, and to ensure that this activity does not constitute an unreviewed safety question. This scenario applies when Unit 2 is defueled and D/G No. 21 is inoperable. After a thorough evaluation, this activity required a change to operating procedures but did not constitute an unreviewed safety question or require a change to the Technical Specifications.

LTOP

In the LTOP SER 8/7/78, BG&E committed to cooling the RCS using steam generators until the steam generator temperature reached 220°F. Only then would RCS cooling be switched to the shutdown cooling system. This was intended to minimize the possibility of starting a reactor coolant pump with a secondary to primary temperature difference of 150°F or more.

This change is to procedurally allow the RCS to be cooled on the shut-down cooling system once RCS temperature reaches 300°F. The maximum steam generator temperature would then also be 300°F.

As a compensatory measure, all four reactor coolant pumps (RCPs) shall be secured and tagged (breakers opened) prior to cooling the RCS below 150°F. This ensures that no inadvertent RCP start will occur with a secondary to primary temperature difference greater than 150°F. Prior to starting an RCP, secondary to primary delta T must be verified less than 150°F.

Water Solid operations are not affected, RCPs are secured and tagged per OP1 and OP5 - no RCP starts are permitted during water solid operations.

This change allows operations greater freedom of operations for the plant secondary. It also saves critical path time during each cool-down to cold shutdown.

The purpose of this evaluation was to document that although the proposed action is not in conformance with our 1978 LTOP SER, there is no unreviewed safety question or change to the Technical Specifications.