BALTIMORE
GAS AND
ELECTRIC

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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, 1991 through December 31, 1991.

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,

for
G. C. Creel
Vice President - Nuclear Energy

Cert No P 807 376 322

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Enclosure 1: Annual Report of Changes, Tests and Experiments (22 pages)

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ENCLOSURE (1)

# ANNUAL REPORT OF CHANGES, TESTS, AND EXPERIMENTS
# CALVERT CLIFFS NUCLEAR POWER PLANT
# 1991

CCI-117 S/N 1-91-1

This temporary modification was to gag open flow control valve 1-SRW-1627CV in its present operating position while the associated flow controller 1-PIC-1627 was repaired. 1-SRW-1627CV is the service water supply pressure control valve for the Condenser Vacuum Pump Seal Water Coolers.

The flow controller 1-PIC-1627 had malfunctioned and was creating operational problems with maintaining normal service water flow to the coolers. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

CCI-117 S/N 1-91-5

The temporary modification authorized the removal of the inboard and outboard journal bearing temperature elements from the No. 11 Component Cooling Water Pump Motor (CCPP); thereby, disabling the respective inputs to the Data Acquisition System (DAS/Plant Computer).

The originally installed motor for #11 CCPP failed and was removed for repairs. The currently installed replacement motor (per FEC 90-03-120) does not have the required end-bell or bearing housing penetrations to allow the subject temperature elements to be installed. This temporary modification supports plant operation during ALL MODES.

This 50.59 was required because the temperature elements (1TE3813A & B) were shown on FSAR Fig. 9-6 (M-51, Sh. 1 of 3, Rev. 28, BG&E Dwg. No. 60-235-E). The temperature elements and their respective inputs to the DAS/Plant Computer are not otherwise discussed in the FSAR or the Technical Specifications. However, the temperature elements are identified by the Q-List as part of the Non-Safety Related portion of the Component Cooling Water System.

The temporary modification was processed to support continued Unit 1 operation and service during ALL MODES. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

CCI-117 S/N 1-91-25

1-TE-156 is a temperature element which monitors the upper guide bearing temperature for No. 11A Reactor Coolant Pump motor. The RTD is spring loaded within the motor housing approximately 4 inches from the point of contact. The lube oil was leaking, at the spring sleeve, through the RTD sheath and carried onto the RTD leads and into a junction box where it was accumulating. Temporary Modification (CCI-117) S/N 1-91-25 severed the RTD outside the motor housing and sealing the sheath with a cap device. A similar modification was used to stop leakage from 1-TE-176 on No. 12A Reactor Coolant Pump in 1987.

The leaking RTD cannot be replaced without partial disassembly of the RCP motor. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

CCI-117 S/N 2-91-6

This Temporary Modification substituted flasks of nitrogen gas in place of instrument air to operate the safety-related pressurizer auxiliary spray line control valve, 2-CV-517. Instrument air needed to be secured in order to conduct the repair/replacement of instrument air check valve, 2-IA-175, which was leaking.

Operability of the auxiliary spray line control valve 2-CV-517 was maintained to provide pressurizer spray cooling of the RCS during plant heatup or if the reactor cooling pumps were secured. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

CCI-117 S/N 2-91-016

This Temporary Modification isolated LPSI loop check valve, 2-SI-144, from the line that taps into the cold leg of reactor coolant loop 22A by freeze sealing a section of the injection line downstream of the valve. The freeze seal resulted in isolating both the HPSI and LPSI headers from reactor coolant loop 22A.

A freeze seal was required to isolate reactor coolant loop 22A from 2-SI-144 and to maintain reactor coolant pressure boundary so that the valve may be repaired in place. Valve 2-SI-144 required repair due to improper seating of the disc resulting in backleakage. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

DCR 90-694

This activity revised drawing M-65 Sh.4, Rev. 7, Ventilation Systems Control Room HVAC to reflect the as built condition.

The DCR was a result of walkdowns performed to verify "as built" vent drain and instrumentation configurations versus existing OM and P&ID drawings.

The 1/4" test connections were shown on the chilled water piping drawing M-807, 60-564-E. The test connections were not shown on P&ID M-65 Sh. 4. The test connections were shown incorrectly on OM-65 Sh. 4. These drawings were corrected per DCR 90-693. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 84-1072

This modification to the Containment Vent/Hydrogen Purge System involved the replacement and relocation of the existing non-safety-related Foxboro FE-6901 Flowmeters with non-safety-related air flow monitors (e.g., Kurz Model #455-08) with an indication range from 0 - 750 SCFM. The new flow totalizers added to the system as a part of the new air flow monitors were not safety-related. The location where the existing Foxboro flowmeters were installed was replaced with a pipe spoolpiece.

The non-safety-related Foxboro FE-6901 Flowmeters were replaced as they do not meet the sensitivity/accuracy requirements for air flow values described in the FSAR post-accident analysis. The Foxboro Flowmeters were replaced with non-safety-related air flow monitors installing the new flow transmitters in the vertical section of the Hydrogen Purge System

pipe downstream of the outboard MOV and installing the flow indicator/totalizer in the Cryogenics Room. Installing the new flowmeters in the vertical section of pipe meets the vendor's requirement of having a straight pipe length of 10 pipe diameters upstream and 3 pipe diameters downstream from the flow transmitter.

A 50.59 Evaluation was needed because the activity constituted a revision to FSAR figures 6-9, 9-20A and Section 6.8.3.3. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 85-1023

Letter L91-090 documented licensing concerns regarding the capability of instrument channel FT-212 to satisfy licensing criteria from NRC Regulatory Guide 1.97 regarding post-accident monitoring of charging flow. During the past three years efforts have been made to compensate for system flow pulsations caused by the positive displacement charging pumps and thus increased the accuracy of channel FT-212. Although some success has been achieved and further testing is planned, evaluations can be undertaken to demonstrate that the existing plant configuration is technically adequate.

The technical evaluations provided herein demonstrate the capability of utilizing existing plant instrumentation to ascertain proper post-accident operation of the charging system. These evaluations show that the Calvert Cliffs Nuclear Power Plant (CCNPP) meets the intent of Regulatory Guide 1.97 given the current existing plant configuration. Specifically, it is proposed that the charging pump breaker lights be utilized as Category 2 indication of charging system performance while FT-212 be utilized for Category 3 backup indication. The charging pump electric current meters, which are already utilized as Category 2 indication, can be utilized as an additional backup to the charging pump breaker lights. These evaluations demonstrate that plant safety is not compromised given existing available status indicators for charging flow. Charging flow is tested using STP-0-73D-1(2). Test data has shown that a charging pump in good condition will deliver 44-45 gpm, and a pump with degraded packing will deliver 41-42 gpm. With this test data, use of the charging pump breaker lights, (backed up by the electric current indications and FT-212), to demonstrate charging system performance is justified.

Licensing commitments currently specify that control room indicated charging flow rate from channel FT-212 can be designated as a type D, Category 2 variable as defined in Regulatory Guide 1.97. A type D designation means that the prescribed variable is utilized to ascertain the operation of a safety system. Category 2 is the recommended designation for this parameter as per Regulatory Guide 1.97 and thus, the as licensed plant design satisfies the licensing criteria therein via direct compliance. However, channel inaccuracies due to pulsating flow from the positive displacement charging pumps have called into question the reliability of FT-212 and thus, its stand-alone capability to satisfy Category 2 criteria from Regulatory Guide 1.97.

A review of the maintenance history associated with channel FT-212 showed that the channel had consistently produced high flow readings over the life of the plant. Efforts to correct this problem had yielded only partial success. Additionally, a review of applicable CCNPP procedures indicated that operators utilize charging pump breaker lights as the preferred charging system performance indicator (vice FT-212), lending further credence to this proposed change.

The change is technically suitable and meets the intent of the NRC Regulatory Guide 1.97. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 85-1052

In order to meet 10CFR 50.62, this FCR installed an additional means of scramming the reactor on high pressurizer pressure. The design used PT-102ABCD to develop a ESFAS 2/4 actuation signal.

The ESFAS A.. logic cabinet will open the load contactor of the M/G set 27' (1 & 2 G206). The ESFAS FL logic cabinet will open the load contactor of the M/G set on 45' (1 & 2 G306).

During normal operation, a single channel trip would not de-energize the RTS bus and cause rod drop. Rod drop will occur on a single channel trip only if the M/G set load sharing circuit does not operate properly. The load sharing feature has operated successfully in the past to catch the load of a M/G set trip.

A new by-pass contactor was installed in parallel with the existing load contactor to allow testing of the load contactor while maintaining generator output. The load contactor and by-pass contactor are controlled by different ESFAS channels so that while testing the channel 'A' load contactor, the by-pass contactor is controlling the Channel 'B' and vise-versa. Using this scheme, if the ESFAS channel controlling the by-pass contactor receives a high pressurizer pressure signal, the output of both M/G sets will be lost.

This scheme uses a set of auxiliary contacts on the by-passed condition at control room panel C05. The by-pass contactor is fully rated for the M/G set output and there are no regulatory restrictions against continued operation in the by-pass mode. While in the by-pass mode, a single DSS trip signal of the proper channel will trip the reactor. Each of the four channel bistables is annunciated on 1(2) C08 if a bistable trips. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 85-1052

This modification installed a Diverse Scram System (DSS) in compliance with 10 CFR 50.62, Reduction of Risk from Anticipated Transients without Scram events. The DSS uses existing Pressurizer Pressure Transmitters PT102A, B, C, D, to provide a high pressurizer pressure signal to "high" bistables added to the pressure loops circuitry. Unused isolators, bistables, and logic modules in the ESFAS panels are used as well as existing CEDM Motor Generator controls. The addition of a new bypass contactor in parallel with the existing load contractor for the CEDM Motor-Generator allows testing of ATWS at power.

The four pressure channels provide pressure signals to four high bistables in the ESFAS sensor cabinets. Each bistable provides channel trip annunciation, and input to two isolators. One isolator provides an input to a two-out-of-four logic module in channel "A" of the ESFAS logic cabinets while the other isolator inputs to channel "B". The logic module supply annunciation and data logger input for "Diverse Scram System Trip", each also energize a trip relay in the ESFAS relay cabinets to open a contact in the "load off" portion of the MG set control circuit thereby causing the load contactor (3M) to open. Both Motor-Generator set load contactors must open to cause a reactor trip. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 87-0073

Supplement 9 to FCR 87-0073 was issued to facilita\ the processing of DCR 89-826. This DCR was released to revise P&ID M-56, Sh. 3 of 6 (Plant Fire Protection System, Turbine, and Service Bldgs. & Intake Struct., Units No. 1 & 2) to reflect the addition of a non-safety related fire protection sprinkler system in the North Service Building. P&ID M-56, Sh. 3 of 6, is FSAR Figure 9-22B. Supplement 9 was also issued to update FSAR Table 9-20 to reflect new information associated with the sprinkler system. The FSAR P&ID does not currently reflect the configuration of the sprinkler piping in the North Service Building. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 87-0111

This modification amplified the FSAR description of the PORV Block MOVs to note that electrical separation was not required between the circuitry from MOV 403 and MOV 405.

FSAR Section 8.5 requires six (6) inches of separation or barriers between safety related circuits of different separation groups. The control board handswitches do not meet this criteria. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 88-0054

This change allowed the use of an alternate material (i.e., 15-5PH stainless steel in lieu of 300 series stainless steel) for the Charging Pump Fluid End Cylinders (i.e., blocks).

Leakage from cracked blocks have been experienced in #12 and #23 charging pumps. A new block was installed on #12 pump and it experienced cracks in less than 1 year. A second new block was installed on #12 pump but it is the same design as the one that failed in less than 1 year. The original block from #12 pump has been repaired and was used to replace the #23 block.

The blocks had reached their service life and needed to be replaced on an "As Failed" basis. The proposed change to 15-5PH material enhanced the fatigue endurance of these blocks. The Safety Evaluation concluded there was no unreviewed safety question or ch..nge in the Technical Specifications.

FCR 89-13

This modification removed the internals from the safety-related check valves 1-SRW-321, 1-SRW-322 and 2-SRW-321 located on the service water return headers for the Emergency Diesel Generators (EDGs) 11, 12, and 21 heat exchangers, respectively. The check valve bodies will remain in place as spool pieces.

This modification was performed for the following reasons:

1. Nonconformance Report (NCR) 9255 was initiated to document that a commitment made to the NRC to perform periodic reverse flow testing on check valves 1-SRW-321, 1-SRW-322 and 2-SRW-321 was not being implemented. The cause of this nonconformance condition was a failure to develop a controlled test procedure to adhere to the commitments made to the NRC regarding the IE Bulletin 83-03. The purpose of reverse flow testing is to ensure that check valve disassembly failure has not occurred. Since the check valves do not perform a necessary function, (See

Reason 3) their presence in the system requires unnecessary testing and maintenance. A separate response to NCR 9255 dated March 15, 1990 addressed the failure to perform periodic testing and concluded that it would not create a safety hazard. Removing the check valve internals would eliminate the need for periodic testing and maintenance.

2. The Nuclear Regulatory Commission (NRC) issued IE Bulletin No. 83-03 which is primarily directed at the failure mode of disassembly or partial disassembly of check valve internals in the raw water cooling system of diesel generators. Several utilities have experienced failures of check valves where Diesel Generators have been declared inoperable because of cooling loop blockage. Due to the frequency of check valve failures reported in this bulletin, it is BG&E's position (for safety concerns) to remove unnecessary check valves in the cooling systems of the EDGs. Removal of the check valve internals would prevent the possibility of Service Water blockage to the EDGs' heat exchangers caused by check valve internals failing.

   The intent of NRC Bulletin 83-03 is to prevent blockage of the raw water cooling systems for EDGs by implementing testing programs to detect check valve disassembly failures. The removal of the check valve internals would ensure that the intent of the NRC bulletin is met. The only safety functions of the valves on the cooling systems for the CCNPP EDGs is to maintain the pressure boundary and to permit passage of the design flow. The removal of the check valve internals will not adversely affect the pressure boundary of the valve bodies and will ensure the passage of design flow.

3. The existing check valves 1-SRW-321, 1-SRW-322 and 2-SRW-321 are not required for backflow prevention or separation criteria and do not provide a necessary function other than to maintain the SRW pressure boundary and to permit passage of the design flow. A review of plant history indicated that these check valves were left over as part of the original SRW design. In the original SRW design, the SRW system was not separated into two subsystems (two subsystems for each unit) and the check valves provided double valve isolation for single valve failure criteria, while the pressure boundary for an EDG's heat exchanger was breached during maintenance. However, before the first fuel loading of Unit 1, the SRW System was separated into two subsystems by replacing the cross over valves with blind flanges. The separation was completed around 1972 and is the existing configuration of the SRW system. In this configuration, the check valves are not required for separation criteria. There are presently two isolation valves in series located upstream and two downstream of each EDG train which meets the single valve failure criteria.

   Service water is supplied to each diesel generator through the air operated control valves 1-CV-1587, 2-CV-1587 and 1-CV-1588 for EDGs 11, 21, and 12, respectively. These valves automatically open upon receipt of a signal when the diesel reaches 250 rpm. The valves then modulate (automatically reposition) to maintain a 5 to 7 psig pressure drop across the three diesel generator heat exchangers. Thus, the control valves are closed when the respective EDGs are shutdown and throttle open to maintain a positive differential pressure across the EDG's heat exchangers when the EDGs are operating. Therefore, backflow protection is provided by the operation of the control valves. In addition, the SRW is a steady state system where a positive differential pressure is always available across the EDG heat exchangers. A1/2" bypass line is provided around the control valve and the heat exchangers for each EDG. This bypass line continuously supplies SRW to the safety related air after cooler for the air start system for the respective EDG. The potential for back flow through this line is limited by the relative small size of the piping and the frictional

losses through the system and after-cooler. In addition, these lines bypass the EDGs heat exchangers where if backflow did occur through these lines it would not pose a problem for the EDG heat exchangers. Therefore, the system does not require check valves to prevent backflow through the EDG heat exchangers as backflow prevention is provided by the control valves and the availability of a positive differential pressure.

Additional backflow protection is provided by the elevation difference between the EDGs and the SRW tie-in for the SRW Pump return line. The EDGs are located on the 45' elevation and the check valves are located immediately before the SRW Pump return line tie-in on the 6'-9" elevation. Therefore, the EDGs are additionally protected from backflow by the inertia of an approximate 38' static pressure head.

4. From a maintenance standpoint, this modification will eliminate the need for periodic testing and maintenance on the check valves and will facilitate the refilling of the EDG SRW piping system following maintenance to the system.

5. The removal of the check valve internals will decrease the pressure drop across the check valves and, therefore, will enhance the performance of the system.

The check valves affected by this activity are safety-related. However, removing the internals from these valves did not compromise any of the safety functions of the SRW system as the check valves do not provide a safety-related function other than pressure boundary and opening to allow flow passage. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-0068

NCR 7556 was initiated due to the FSAR description of the RWT water volume and the actual useable volume being in disagreement.

Specifically, this activity will do the following:

1. Provide a better description of the quantity of water supplied by the RWT.

2. Correct the description of the level alarms on the RWT and clarify the purpose of the alarms.

   a. Paragraph 6.3.1 page 6-4

   **Existing wording:**

   These headers are initially supplied with borated water from the Refueling Water Tank and after that tank is 10 percent full, borated water is recirculated from the sump of the containment.

   **Recommended wording:**

   After the headers are initially supplied with at least 360,000 gallons of borated water from the Refueling Water Tank, a Recirculation Actuation Signal (RAS) occurs. The RAS shifts the suction of the headers from the RWT to the containment sump to recirculate the borated water.

b. Paragraph 6.3.1 page 6-6

**Existing wording:**

In the event the automatic transfer fails to occur upon Recirculation Actuation Signal (RAS), the redundant refueling water tank low level alarms are provided to alert the control room operator. The operator can simulate automatic RAS actuation by two manual pushbuttons labeled RAS "A" and RAS "B" on the main control boards."

**Recommended wording:**

In the event the automatic transfer fails to occur Recirculation Actuation Signal (RAS), the operator can manually initiate recirculation using two buttons labeled Recirculation Manual Actuation Channel "A" ("B") on the main control board.

c. Paragraph 6 3.2.4, page 6-12

**Existing wording:**

The Refueling Water Tank is provided with a high level alarm and redundant low level and temperature alarms.

**Recommended wording:**

The Refueling Water Tank is provided with both a wide range and a narrow range level indicator. The narrow range instrument provides both a high level alarm. The wide range instrument provides only a low level alarm. The high level alarm is to alert the operators of an impending overflow of water from the RWT to Miscellaneous Waste Processing System. The low level alarms are used to assist the operator in monitoring for sufficient water inventory in the RWT. Redundant temperature instruments provide both high and low temperature alarms.

d. Paragraph 6.3.3 page 6-14

**Existing wording:**

When the Refueling Water Tank is 10 percent full, a recirculation actuation signal (RAS), opens the isolation valves in the two lines from the containment sump.....

**Recommended wording:**

When the Refueling Water Tank level reaches the RAS setpoint, a Recirculation Actuation signal (RAS) occurs which opens the isolation valves.....

e. Paragraph 6.4.2 page 6-21

**Existing wording:**

The minimum capacity of the RWT is 400,000 gallons whereas the capacity of the reactor coolant system is 77,800 gallons.

**Recommended wording:**

The Technical Specification requires a minimum inventory of 400,000 gallons be maintained in the RWT. Additional RWT data is provided in Table 6-4. The Technical Specification limit and RAS setpoint level have been established to ensure the RWT provides at least 360,000 gallons of usable water before the RAS is actuated. The capacity of the reactor coolant system is 77,800 gallons.

f. Paragraph 6.4.2, page 6-22

**Existing wording:**

When the low liquid level is reached in the Refueling Water Tank.....

**Recommended wording:**

When the Refueling Water Tank level reaches the RAS setpoint, a Recirculation Actuation Signal (RAS) occurs which opens the isolation valves and.....

g. Paragraph 7.3.2.1, page 7-25

**Existing wording:**

In addition, each provides a high level and a low level alarm.

**Recommended wording:**

The full (wide) range indicator provides only a low level alarm. The narrow range indicator provides both a high level and a low level alarm. The high level alarm is to alert the operators of an impending overflow of water from the RWT to the Miscellaneous Waste Processing System. The low level alarms are used to assist the operator in monitoring for sufficient water inventory in the RWT.

The Safety Evaluation concluded there was no unreviewed safety question or a change to the technical specifications.

FCR 89-0026

This FCR initiated the following modifications to the Auxiliary Feedwater System turbine Main Steam supply lines. These lines were designed to ANSI B31.7 Class II from the Main Steam penetrations up to and including the isolation valves and to ANSI B31.1 beyond the valves.

1. Added an air-operated 2" bypass valve and associated position switches around each Main Steam admission valve (1/2-CV-4070 and 4071). The admission valve will begin opening a short time after the actuation signal is initiated. The time delay for each steam admission valve will be accomplished via a safety-related (SR) adjustable time delay relay located in Main Control Room (MCR) Panel C04.

2. Added a flange pair and restriction orifice downstream of the 2" bypass valve.

3. Relocated the check valves from the vertical run of the steam supply line to a horizontal run.

4. Added a locked open manual gate valve upstream of each station's steam admission, new bypass, and existing manual bypass valves.

5. Added a manual bypass valve around both valves in item 4 above.

6. Added a manual, locked open gate valve downstream of each relocated check valves before the individual steam admission lines combine.

7. Replaced the handswitches for the steam admission valves so that both the admission valve and the bypass valve at each valve station are controlled from a single handswitch and provide position indication for each valve individually. The existing AFAS relay is used to actuate both valves.

8. Replaced the handswitches for the Auxiliary Feed pump turbine trip with push button switches.

9. Removed the air supply for Main Steam admission valves CV-4070 and CV-4071 from the AFW accumulators B and A, respectively, and provide a new air source for these valves and their respective bypass valves CV-4070A and CV-4071A. The new air supply will be normally from the NSR Instrument Air header, backed up by the SR SWAC system (which is normally isolated) as well as by dedicated SR accumulators. The accumulators will be charged by the existing AFW air amplifier system, which also includes normally isolated SWAC and nitrogen backup. The accumulators (one for each valve station) will be located in the Service Water (SRW) pump room. Each one will be sized considering system leakage for two hours and then stroking its associated valves two times at the end of the two hours. A low pressure switch will be provided on each accumulator to initiate a low pressure alarm in the MCR.

10. Relocated the existing solenoid valves and associated air regulators for CV-4070 and CV-4071 from the MSIV Room to the East Penetration Room, Elev. 27'-0". The new solenoid valves, gauges, check valves, etc. for CV-4070A and CV-4071A will also be located in the East Penetration Room. Sizing of the solenoid valves has taken into consideration the 5 second opening/closing criteria for these valves. All new and replacement components are SR, including the electrical installation in support of this modification. All solenoid valves will be normally de-energized operation, in accordance with the existing design for the Main Steam Admission Valve.

In addition, FEC 89-26-12 requests a correction to the statement of small pipe high energy line break criteria as stated in FSAR Chapter 10A.

The changes were required to allow the turbine governor to accelerate the Auxiliary Feedwater pump in a more controlled manner. The Safety Evaluation concluded there was no unreviewed safety question or change to the technical specifications.

FCR 89-121

FCR 89-121 was being issued to correct the FSAR description of Section 11.3, "Radiation Safety". The change corrects the administrative quarterly dose limit currently provided in the FSAR to match those which are provided in CCI-800C, "Calvert Cliffs Radiation Safety Manual". This 50.59 Evaluation was being written to evaluate the consequences of changing the administrative quarterly dose limits currently stated in the FSAR to match the actual administrative quarterly dose limits as shown in CCI-800C.

Quality Audits unit Finding Sheet, finding number 89-02-1 finds the "as-found condition of the FSAR" as: "FSAR Section 11.3.3.2, Personnel Monitoring Program, states that "an alert system will be sued to emphasize those individuals who are approaching the administrative quarterly dose limit (1.25 rem)".

But CCI-800C, Calvert Cliffs Radiation Safety Manual, Attachment (1) Section I. C., Administrative Dose Limits, states that "the quarterly whole body dose limit for individuals 19 years of age older, is administratively set at 2.0 rem. Individuals will be restricted at the Alert Point (900 mrem)..."

The QA finding lists the root cause of this error as "inattention to detail". The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 89-0179

This modification replaced existing piston operated dampers 1-PO-5406 and 1-PO-5407 in the discharge of ECCS pump room exhaust fans No. 11 and 12, respectively, with gravity dampers. In addition, solenoid valves 1-SV-5406 and 1-SV-5407, pressure control valves 1-PCV-5406 and 1-PCV-5407, instrument air lines and accumulators were removed. Also, limit switches 1-ZS-5406A & B and 1-ZS-5407A & B, which are installed on the dampers and indicate fully open or fully closed at control panel 1C34, and all associated circuits and raceways were removed or spared. The indicating function was added per FCR 84-1088, Item No. 51 to meet the Reg. Guide 1.97 requirements.

In addition, this modification replaced the existing flexible connections between the existing piston operated dampers and duct work with continuous molded flexible connections between the new gravity dampers and the existing duct work.

The dampers are classified as safety-related, where as the flexible connections are non-safety-related.

This modification was made for the following reasons:

1. To reduce the probability of a malfunction of equipment, the piston operated dampers require a source of compressed air, solenoid valves, and associated controls; whereas, the only external means necessary to operate the gravity dampers is the static differential pressure across the dampers which is provided by the operation of the exhaust fans.

2. To reduce the load on the salt water air supply, which results in spare capacity for future safety-related use.

3. To maintain a consistent system configuration with the Unit 1 ECCS pump room exhaust system. Engineering for the replacement of the piston operated dampers on the discharge of the Unit ECCS pump room exhaust fans was issued under this FCR.

In addition, this modification replaced the existing flexible connections between the exiting piston operated dampers and duct work with continuous molded flexible connections between the new gravity dampers and the existing duct work. The existing flexible connections develop leaks at the overlap folds over a period of time. The use of continuous molded flexible connection will eliminate this problem. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-19

This safety evaluation was written to determine if an Unreviewed Safety Question exists due to the lack of fire rated dampers in the ventilation ducts where the battery room ventilation system (supply and exhaust) penetrates two barriers. Specifically, dampers are not installed in the barrier between the Unit 1 Cable Spreading Room and Cable Chase 1B and the barrier between Cable Chase 1B and Cable Chase 1A. If no USQ exists, then no dampers will be installed.

It was discovered that a total of four fire rated dampers had not been installed where the 125VDC Battery Room ventilation system supply and exhaust ducts penetrate two barriers. A review of commitments regarding the rating of these barriers revealed that BG&E had indicated that at least 1-1/2 hour rated dampers would be provided in ventilation penetrations. However subsequent guidance provided by the NRC in Generic Letter 86-10 addressed deficiencies in barriers which may eventually be discovered and barriers which must undergo changes as a result of plant modifications. In this Generic Letter, (Section 3.1.2 of Enclosure 2), the NRC directs that an Engineering Evaluation be performed by a fire protection engineer to determine if a barrier which is not completely sealed from floor-to-ceiling will still provide adequate separation.

The Generic Letter provides guidance on Future Changes. This guidance recommends that an evaluation be made in conformance with 10CFR50.59 to determine whether an Unreviewed Safety Question exists primarily in the context of Appendix R compliance. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-0020

This change removed the function of the Iodine Removal Unit (IRU) Dousing System. This change isolates the IRU Dousing System when either the containment spray system or the containment iodine removal system is required to be operable by the Technical Specifications (Modes 1, 2, 3, and 4). Manual valves SI-4949, SI-4950, SI-4951, SI-4958, SI-4959, SI-4960 will provide the isolation. The Main Control Room switches for dousing valves SV-4952, SV-4953, SV-4954, SV-4955, SV-4956, and SV-4957 will not provide any function when the manual valves are closed. The control circuits for SV-4159 and SV-4160 will be classified as NSR. CV-4159 and CV-4160 will retain a safety-related pressure boundary function. In Modes 5 and 6, the manual valves may be open to allow the dousing system to be functional during iodine removal unit maintenance to provide fire protection if required.

The IRU temperature indication and monitoring equipment is reclassified as NSR as the temperature function is not required nor is its function related to any automatic safety actions. Since an analysis shows that IRU overheating is not credible, the temperature instrumentation is not needed as an indication that would trigger operator action.

Coincident with a LOCA or post-LOCA, a single failure of the limiting component (the loss of a diesel generator or an IRU fan failure) will not result in the charcoal ignition or desorption temperature being reached. Such a failure during a LOCA will not cause an increase in offsite does consequences through the removal of the dousing system function.

A detailed analysis was performed (NUCON Report No. 6BG021/01, dated 1/19/90 and Supplement 1, dated 7/25/90) to assess the maximum charcoal filter temperature possible during a maximum hypothetical accident (LOCA). System performance parameters (spray iodine removal, charcoal efficiency data) were obtained from FSAR Chapter 6. Source

term and isotope distribution data were obtained from Chapter 14. Other inputs came from design drawings, manufacturers' data, and use of the computer code along with the input and assumptions used. The report concluded that under postulated accident conditions iodine desorption or charcoal filter fires are not credible.

BG&E did not commit to monitoring IRU charcoal bed temperature as part of the Regulatory Guide 1.97 submittal, nor has BG&E committed to having the dousing system perform a safety-related function. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-110

The activity associated with FCR 90-110 made permanent eleven CCI-117 Temporary Modifications of the Reactc. Vessel Level Monitoring System (RVLMS, a Heated Junction Thermocouple System - HJTCS). A secondary purpose of this activity is to provide the basis for future changes to the RVLMS which are identical to the types of wiring modifications described in FCR 90-110 and evaluated herein. Specifically, the following types of wiring modifications to the RVLMS's electronics are evaluated:

1. Revise the reference thermocouple used for some sensors which have a failed reference junction (open or shorted Unheated thermocouple lead or Common thermocouple lead), to restore operability of those sensors;

2. Jumper failed heated thermocouples (either due to a failed heater or a failed thermocouple to adjust thermocouples (if possible, to the operable sensor immediately above it) so that the alarms can be cleared to restore the channel to operability and o that the RVLMS indicators conservatively alert the operator to level changes; and

3. Replace failed heaters with a dropping resistor to provide proper power to other heaters in its series string and allow the remaining sensor string to operate properly.

The above changes are within the design intent of the system and document the current "As-Built" condition of the plant. If a new HJTCS probe is ever installed, these changes would have to be re-evaluated. All the above changes are performed by lifting leads and providing jumpers and/or replacement heater dropping resistors in the back of the HJTCS electronics cabinets.

This activity makes the existing CCI-117 temporary wiring modifications to the HJTS's electronics permanent. The closes out eleven CCI-117's on Units 1 and 2 combined. The activity also documented the CCI-117 wiring modifications on the design documentation for Units 1 and 2. In addition, the activity provided for future changes to the RVLMS which are identical to the types of wiring modifications in FCR 90-110. The changes to the RVLMS restore some disabled sensors and clear alarms from inoperable sensors to allow operable sensors to alarm if required. This FCR allowed an RVLMS sensor with a disabled reference thermocouple to be operable for the purposes of meeting the operability requirements of the newly approved Technical Specifications No. 3/4.3.3.6 (Unit 1 Amendment 147; Unit 2 Amendment 128). The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 90-117

Revised the description of the subcooled margin monitor in UFSAR 7.5.9.1.

Revised the first line of 7.5.9 to change the acronym for the subcooled margin monitor from SMM to SCMM. Revised the second to last line of 7.5.9.2 for the same reason.

The stated range and accuracy were not correct, and there is only one pressure input per channel. This change also provides clarification of the lack of electrical separation of the signal inputs within each channel.

Additionally, both the subcooled margin monitor and the shutdown margin monitor were referred to as SMM. The subcooled margin monitor has been changed to SCMM. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 91-246

This modification provided a new Corrosion Products Sampler (CPS) in the Turbine Plant Sample System (TPSS). The CPS is a passive device that allows a continuous sample stream to flow through a filter, where corrosion products are trapped for laboratory analysis.

FCR 91-246, Supplement 0 provided engineering for installing a Radiological & Chemical Technology, Inc. (RCT) CPS, at each TPSS auxiliary panel (1/2-T21A), connected to the main feedwater header, so that corrosion products sampling is permanently available. Corrosion product sampling only existed via a portable test rig. By augmenting the existing TPSS, the CPS will permanently enable enhanced corrosion protection monitoring for the main feedwater system, thereby providing added assurance that the feedwater chemistry and secondary side corrosion rate remains within acceptable limits. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FCR 91-208

This evaluation is being performed in response to NCR 8474, to ensure an unreviewed safety question does not exist when the following statement in UFSAR Section 9.8.2.3 is deleted: "The relative humidity in the auxiliary building isn't likely to exceed 50 percent, consequently, the maximum iodine removal efficiency should be realized" and the following statement is added: "Periodic testing is conducted to ensure maximum filter efficiency of 90% or greater is maintained."

No documentation has been identified to confirm the value of 50% relative humidity in the auxiliary buidling. This value is not used in design bases in the accident scenarios of the UFSAR or in HVAC design base calculations for the auxiliary buidling. Further, the EQ Design Manual lists the maximum relative humidity for normal and LOCA situations as 70%. For main steam line breaks (MSLB) and high energy line breaks (HELB) the auxiliary building relative humidity is listed as 100%. The charcoal filters are not required to be in operation during HELB and MSLB incidents. The Safety Evaluation concluded there is no unreviewed safety questions or change in the Technical Specifications.

FCR 91-248

FSAR Section 5A.3.2.2 states that ASME Code Case N-411 will be used when performing new analyses or reconciliation on Seismic Class 1 piping systems. This activity changes the requirement to allow optional use of ASME Code Case N-411 when not using Regulatory Guide 1.61, in order to take advantage of increased piping system damping values.

BG&E's original intent was to allow ASME Code Case N-411 to be used as an optional method of piping analysis. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 83-47-03

This activity revised the description of the boronometer in the UFSAR Chapter 9 to correctly describe the range and accuracy as modified by FCR 83-0047.

FCR 83-0047 installed new electronics and associated hardware to upgrade the boronometer to a temperature compensated, digital/analog display with an overall range of 0 to 5000 ppm boron concentration. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 89-01-476

This minor modification added an isolation valve, pipe nipples, and pipe cap to valve 2CC-118 sight glass isolation valve on the #21 component cooling chemical addition tank. The addition of this valve and piping made the design consistent with that of the Unit 1 chemical addition tank.

This evaluation also includes the revision of FSAR Figure 9-6 to reflect the Unit 1 as-built design. The modification identified above is the current as-built design for Unit 1. Figure 9-6 is being revised to reflect that current as-built design. Since the design described above currently exists for Unit 1, all discussions concerning the modification for Unit 2 would also apply to Unit 1.

There was no method which facilitated draining the Unit 2 chemical addition tank to allow the addition of chemicals. The addition of the isolation valve and pipe allows venting of the tank by removing the pipe cap and opening the isolation valve. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

FEC 90-01-955

This activity removed pressure indicators 1/2-PI-294, 1/2-PI-295, 1/2-PI-290, 1/2-PI-291, and 1/2-PI-292 from the suction side of the LPSI and HPSI pumps and caps the tubing where the pressure indicators tie into the system. In addition, P&ID M-74 (FSAR figure 6-1) and P&ID M-462 (FSAR figure 6-10) were also modified to reflect as-built conditions identified during a system walkdown. These P&ID drawing discrepancies with as-built conditions are the following:

1. Locations of 1-PP-302W and 1-PI-295 are reversed.

2. Delete 1-PP-301V and 1-PP-301W.

3. Show 1-PP-301Y connected to flanged spool piece.

4. Show capped tube connected to flanged spool piece.

5. Locations of 2-PI-295 and 2-PS-302Y are reversed.

Pressure indicators on the suction side of the LPSI and HPSI pumps (both units) are used only during testing as required by existing STPs. Installed PIs were becoming a metrology problem because of their tendency to drift out of calibration. Using high accuracy PIs when needed as opposed to the permanently installed PIs will alleviate this situation.

This activity involved the removal of pressure indicators that are not used during normal operations. The system pressure boundary is maintained since existing isolation valves will remain in their normal operating (shut) condition. These isolation valves are identified as the safety-related boundary. The modification left at least one Pressure Point (PP) on each pump that can be utilized for pump testing. This activity also modified P&IDs M-74 and M-462 to reflect as-built conditions identified during a system walkdown. The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

<u>FEC 90-01-115</u>

FEC 90-01-115 (DCR 89-1695 and DCR 90-1377) were initiated to revise P&IDs M-64 SH 1, Auxiliary Building Ventilation System. (DCR 90-0181 revises the Johnson Controls Tubing/Actuator drawings to reflect the changes made on M-64 Sh. 1)

1. Each of the Fuel Handling Area HVAC units' dampers, O-PO-5414 and O-PO-5415, are shown on P&ID M-64 SH 1 as having a single pneumatic piston actuator. This activity proposes that each damper be shown as having two pneumatic piston actuators. This is the as-built condition. There is no vendor documentation specifying the number of piston actuators required; however, each of the two dampers is supplied with two blade actuating rods.

2. Each of the Fuel Pool Exhaust Filter's dampers, O-PO-5417 and O-PO-5418, are shown on P&ID M-64 SH 1 as having a single pneumatic piston actuator. This activity proposes that each damper be shown as having four pneumatic piston actuators. This is the as-built condition and is supported by the filter vendor's design drawing.

The dampers described above are safety-related. The proposed activities do not affect the function or operation of any safety-related system, structure, or component. A 50.59 Evaluation is necessary since the changes create revisions to P&ID, M-64 SH 1 which is FSAR Figure 9-21. The Safety Evaluation concluded there was no unreviewed safety question or a change in the technical specifications.

<u>FEC 90-01-1054</u>

FEC 90-01-1054 authorized the addition of reinforcement bars to the Fuel Transfer Tube Blind Flange. The modification was intended to address minimum wall thickness concerns with the blind flange.

A SE was done because FSAR figure 9-14 is being changed as a result of FEC 90-01-1054.

Prior to commencing the U2 ILRT, the U2 Fuel Transfer Tube Blind Flange had to be installed. During the reinstallation process, questions arose concerning the minimum wall thickness of the blind flange. HOLTEC INTERNATIONAL was contracted to calculate the minimum wall thickness using finite element analysis. The finite element analysis was scheduled to take several weeks and previous "hand" calculations showed there was a high probability the existing blind flange was inadequately sized. Therefore, BG&E decided to make a conservative modification to the blind flange immediately. The modification would add cross members to the blind flange compensating for the potential inadequate thickness and allow the ILRT to begin without the delay of waiting for the finite element analysis to be completed (Modification details were obtained from HOLTEC INTERNATIONAL and installed by way of a Provisional Modification). The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

MASE 90-4

Since the Service Water System 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 proposed limits provide flexibility, establishing a range of allowable contamination levels for normally non-contaminated systems wherein continued operation is acceptable.

IE Bulletin 80-10 states the regulatory requirements for operating non-radioactive systems which become contaminated. If continued operation of the system as contaminated is necessary, an evaluation is required to determine whether continued operation is acceptable. The evaluation considers the level of contamination, potential releases to the environment, the relationship of such releases to the radioactive effluent limits of 10CFR20 and the facility's Technical Specifications, and the environmental radiation dose limits of 40CFR190. The evaluation sets forth the basis and criteria on which the determination was made.

This evaluation uses the methodology and parameters used in the ODCM to calculate off-site doses due to both accident and normal releases from the system.

Chemistry Department procedures already require periodic sampling of the Service Water System to provide early identification of cross-contamination. Criteria established in this evaluation assure that the regulatory limits of 10CFR20, 10CFR50, and 40CFR190 are met. The allowable contamination levels are set such that either accident or normal releases from the Service Water System will contribute less than the Technical Specification limits for off-site doses. Furthermore, the limits assure that releases to the environment will not contain radioactive material in concentrations greater than the MPC. This low level of contamination still allows operational flexibility while ensuring that the incremental increase in off-site doses remains low enough that the Technical Specification limits will not be exceeded. The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

MASE 90-5

Since the Demineralized Water System 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 proposed limits provide flexibility,

establishing a range of allowable contamination levels for normally non-contaminated systems wherein continued operation is acceptable.

IE Bulletin 80-10 states the regulatory requirements for operating non-radioactive systems which become contaminated. If continued operation of the system as contaminated is necessary, an evaluation is required to determine whether continued operation is acceptable. The evaluation considers the level of contamination, potential releases to the environment, the relationship of such releases to the radioactive effluent limits of 10CFR20 and the facility's Technical Specifications, and the environmental radiation dose limits of 40CFR190. The evaluation sets forth the basis and criteria on which the determination was made.

This evaluation used the methodology and parameters used in the ODCM to calculate off-site doses due to both accident and normal releases from the system.

Chemistry Department procedures already require periodic sampling of the Demineralized Water System to provide early identification of cross-contamination. Criteria established in this evaluation assure that the regulatory limits of 10CFR20, 10CFR50, and 40CFR190 are met. The allowable contamination levels are set such that either accident or normal releases from the Demineralized Water System will contribute less than the Technical Specification limits for off-site doses. Furthermore, the limits assure that releases to the environment will not contain radioactive material in concentrations greater than the MPC. This low level of contamination still allows operational flexibility while ensuring that the incremental increase in off-site doses remains low enough that the Technical Specification limits will not be exceeded. The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

MASE 90-6

Since the Plant Heating System 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 proposed limits provide flexibility, establishing a range of allowable contamination levels for normally non-contaminated systems wherein continued operation is acceptable.

IE Bulletin 80-10 states the regulatory requirements for operating non-radioactive systems which become contaminated. If continued operation of the system as contaminated is necessary, an evaluation is required to determine whether continued operation is acceptable. The evaluation considers the level of contamination, potential releases to the environment, the relationship of such releases to the radioactive effluent limits of 10CFR20 and the facility's Technical Specifications, and the environmental radiation dose limits of 40CFR190. The evaluation sets forth the basis and criteria on which the determination was made.

This evaluation uses the methodology and parameters used in the ODCM to calculate off-site doses due to both accident and normal releases from the systems.

Chemistry Department procedures already require periodic sampling of the Plant Heating System to provide early identification of cross-contamination. Criteria established in this evaluation assure that the regulatory limits of 10CFR20, 10CFR50 and 40CFR190 are met. The allowable contamination levels are set such that either accident or normal releases from the Plant Heating System will contribute less than the Technical Specification limits for off-site doses. Furthermore, the limits assure that releases to the environment will not contain radioactive material in concentrations greater than the MPC.

This low level of contamination still allows operational flexibility while ensuring that the incremental increase in off-site doses remains low enough that the Technical Specification limits will not be exceeded. The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

MASE 90-7

Since the Nitrogen System 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 proposed limits provide flexibility, establishing a range of allowable contamination levels for normally non-contaminated systems wherein continued operation is acceptable.

IE Bulletin 80-10 states the regulatory requirements for operating non-radioactive systems which become contaminated. If continued operation of the system as contaminated is necessary, an evaluation is required to determine whether continued operation is acceptable. The evaluation considers the level of contamination, potential releases to the environment, the relationship of such releases to the radioactive effluent limits of 10CFR20 and the facility's Technical Specifications, and the environmental radiation dose limits of 40CFR190. The evaluation sets forth the basis and criteria on which the determination was made.

This evaluation uses the methodology and parameters used in the ODCM to calculate off-site doses due to both accident and normal releases from the system.

Chemistry Department procedures already require periodic sampling of the Nitrogen System to provide early identification of cross-contamination. Criteria established in this evaluation assure that the regulatory limits of 10CFR20, 10CFR50, and 40CFR190 are met. The allowable contamination levels are set such that either accident or normal releases from the Nitrogen System will contribute less than the Technical Specification limits for off-site doses. Furthermore, the limits assure that releases to the environment will not contain radioactive material in concentrations greater than the MPC. This low level of contamination still allows operational flexibility while ensuring that the incremental increase in off-site doses remains low enough that the Technical Specification limits will not be exceeded. The Safety Evaluation concluded there was no unreviewed safety question or change in the technical specifications.

MM 91-012-015-0

Minor Mod 91-012-015-0 installed high accuracy transmitters to monitor the pressure drop across the tube side (saltwater) of the service water heat exchangers (SRWHX).

The condition of the SRWHXs ultimately affects the ability of the plant to reject heat to the Chesapeake Bay. As surface of the heat exchangers are progressively fouled, differential pressure increases and heat transfer is impeded. The gauges that were installed on the saltwater inlet and outlet of the SRWHXs, 1/2-PI-5209, 5210, 5211, and 5212, did not have the resolution or accuracy required for performing this function. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

MM 91-019-012-0 (Unit 2)
MM 91-019-013-0 (Unit 1)

This activity updates P&IDs M-53 Sh. 2 (Minor Mod 91-019-013-0), M 479 Sh. 1 (Minor Mod 91-019-012-0) and other impacted documents to reflect the "as-built" condition of the Compressed Air System (Instrument Air and Plant Air). The P&IDs are required to be updated by incorporating the following:

On P&ID M-479 Sh. 1 (FSAR Figure 9-23A), show 3/4" diameter Plant Air supply line to Chemistry Lab at EL. 69'-0" with isolation valve 2-PA-282 in the normally open position.

On Unit 1 P&ID M-53 Sh. 2 (FSAR Figure 9-23-2), show 1" x 1/2" reducing bushing downstream of valve 1-IA-219 in lieu of hose connection. Further, change flow direction from 2-way flow to 1-way flow on Instrument Air piping header valves 1-IA-218 and 1-IA-220. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

MM 91-043-005-0

This activity added permanent loop seals to condenser air discharge header drain lines containing valves 1-CAR-156 and 1-CAR-157 for Unit 1 and 2-Car-156 for Unit 2 in the Condenser Air Removal System. The permanent loop seals consist of piping, valves and caps.

In order to prevent air from escaping or entering the condenser air discharge header through the drain lines while the moisture from the air and other noncondensible gases are removed/drained during normal operation, permanent loop seals are added to the existing drain lines containing valves 1-CAR-157 and 2-CAR-156. A 50.59 Safety Evaluation is required because FSAR Figures 10-7 (Unit 1 P&ID M-50) and 10-12 (Unit 2 P&ID M-451) are revised to show the drain lines with loop seals. The Safety Evaluation concluded there was no unreviewed safety question or change in the Technical Specifications.

MM 91-045-006-0

This activity relocated the non-safety related Instrument Air supply tubing run for the Steam Generator Feed Pump Turbine (SGFPT) No. 22 Hydraulic Assembly, Panel 2T30. The modification taps into an Instrument Air source, which is close (within one foot) to the Hydraulic Assembly, and routes new supply tubing to the panel. The revised air supply includes a filter/regulator, a shutoff valve and a maintenance connection for a temporary air supply and is consistent with the original air supply configuration.

The revised tubing run has less pressure drop and provides an improved air supply to the SGFPT speed changer valves.

The air supply configuration is as follows:

The air supply is 20 psig, regulated by a new airset 2-PCV-8040B, located just outside 2T30;

the supply air tubing will be 3/8" O.D. and 1/4" O.D. by 6" long stainless steel with a wall thickness of .065" and a total linear length of 4 feet.

Minor Modification 91-045-006-0 was written due to degraded performance of the SGFPT No. 22 speed changer valves 2-CV-8040 and 8041 located in the Turbine Building, SGFPT Hydraulic Assembly Panel 2T30, on Elev. 12'-)". The change is to increase the size of the air supply tubing to the 22 SGFP speed control valves.

The previous configuration for the air supply to the subject valves was as follows:

- The supply air is 20 psig, and is regulated by airset 2-PCV-8040 located in the Pneumatic Assembly Panel 2T29;
- the supply air is routed from 2T29 to the Hydraulic Assembly Panel 2T30 (where the valves are located);
- the supply air tubing is 1/4" O.D. stainless steel with a wall thickness of .065" and a total linear length of approximately 55 feet.

This configuration imposed a large pressure drop on the air supply and adversely affects the response and operation of the speed changer valves. A temporary test configuration for the air supply, using 3/8" plastic tubing approximately 20 feet in leng was routed from 2T29 to 2T30. The performance of the speed changer valves greatly improved during the test, and proved that the cause of the problem was, in fact, the restrictive and overly lengthy supply air tubing run. The Safety Evaluation concluded there was no unreviewed safety question or a change in the technical specifications.

MM 91-059-005-0

Minor Modification 91-059-005-0 (Unit No. 1) and 91-059-005-1 (Unit No. 2) were initiated to evaluate/repair the Containment Pressure instrument tubing systems.

The pressure sensing line installation were analyzed using the ME-101 stress program and building differential movements which were reduced using a more current methodology which differs from the design basis analysis method provided in the FSAR. As a result of the analyses, the activity removed supports from the instrument tubing systems.

When the original Calvert Cliffs seismic analysis was performed in 1970 for the main building structures, such as the Containment and Auxiliary Buildings, a simplistic mathematical model was constructed for input to the original computer analysis.

The model for the building structures used excessively conservative soil damping values.

The conservatism inherent in the original analysis can be quantified by evaluating the original results using more current define soil damping.

These evaluations yield building displacement values which are considerably less than those originally calculated. These reduced displacement values have been used as input to the tubing stress analysis.

This approach, which utilizes more reasonable damping value, has been previously used on Calvert Cliffs to formulate to response to NRC IE Bulletin 80-11 with regard to the seismic qualifications of masonry block walls.

The 50.59 ws performed to document and support the approach based on the revised building differential movement and is applicable to the Unit No. 1 and Unit No. 2 Containment Pressure instrument tubing systems.

A 50.59 Evaluation is necessary since the tubing system analysis is based on seismic movement values which were reduced using the more current methodology which differs from the design basis analysis method provided in the FSAR. The Safety Evaluation concluded there was no unreviewed safety question or a change in the technical specifications.

MM 91-068-001-0

This activity is to omit reference to 250 #JIB crane in FSAR Section 9.7.2.4.

MM# 91-068-001-0 is removing the 250 #JIB crane under the fuel pool restoration project. The Safety Evaluation concluded there was no unreviewed safety question or a change in the technical specifications.

PROCEDURE HE-50

Procedure HE-50 has been initiated to place a temporary cover over the top of the Containment Sump Screen in Modes 4, 5, 6, and when Defueled, whenever maintenance activities which could compromise the cleanliness of the sump screen box are to occur in containment. This temporary cover will be in the form of scaffolding, boards, and a thermal barrier cloth (Mech. No. 55373) cover securely fastened to the scaffolding. The cover is to be raised a minimum of one foot above the sump screen so that flow through the top of the sump screen will not be impeded by the addition of this cover. Prior to entering Mode 3 the scaffolding is to be disassembled, and any debris found lying on the cover is to be removed from containment.

This 50.59 Safety Evaluation was written because of the impact this procedure would have on a structure described in the FSAR. The Containment Sump Screen Box construction and the effective flow area through the screen are described in the FSAR. Also , the sump screen is depicted in a drawing contained in FSAR. Since this cover can be considered to be part of the overall structure of the sump screen the description in the FSAR is being altered. Finally, to properly determine whether a 50.59 was required or not it was necessary in this case to do all the research required for a 50.59. In the final analysis it was still undetermined whether a 50.59 was required; however, because questions could arise concerning the safety significance of this procedure it was deemed prudent to write a 50.59 so that these concerns could be identified and then formally addressed.

During maintenance activities in Modes 4, 5, 6, and when Defueled, debris may fall onto the sump screen. The debris may be of such a size (long and narrow) that it is able to pass through the sump cage wire mesh, and enter the containment recirculation suction lines.

This debris may either block these suction lines, or be transported to the ECCS pumps (HPSI, LPSI, and Containment Spray) where upon it may cause the malfunction of one or more of these pumps when these pumps are required to operate by taking suction from the containment sump. The main purpose of this activity is to prevent debris from collecting in ECCS suction piping (the suction piping used during containment recirculation) by placing a cover over the sump cage. This will prevent debris from entering the containment sump recirculation lines from the direction in which it is likely to enter. Debris entering the Containment Sump Cage by falling through the side is not considered a likely event. The Safety Evaluation concluded there was no unreviewed safety question or a change in the Technical Specifications.