U. S. NUCLEAR REGULATORY COMMISSION
Region I

Docket/Report: 50-317/84-05 License: DPR-53
50-318/84-05 DPR-69

Licensee: Baltimore Gas and Electric Company

Facility: Calvert Cliffs Nuclear Power Plant, Units 1 & 2

Inspection At: Lusby, Maryland

Dates Conducted: March 1 and 2, 1984

Submitted:

K P Ferlic
K. P. Ferlic, Project Engineer

June 11 1984
date

E. M. Kelly
E. M. Kelly, Project Engineer

June 6, 1984
date

Approved:

DC Trimble for
E. C. Wenzinger, Chief, Project Section 1A

June 22, 1984
date

Summary: Special inspection by two region-based inspectors (30 hours onsite) of licensee actions taken to comply with requirements discussed in NUREG-0737, Item II.B.2, Design Review of Plant Shielding. Calculational assumptions, results and design details were reviewed with cognizant engineers. Physical modifications, along with appropriate emergency procedure changes, were observed and discussed with plant staff.

The development and implementation of actions to comply with Item II.B.2 were found to be thorough, well-documented, and satisfactory with respect to NUREG-07`7 commitments. Of particular note were the calculational techniques utilized to estimate streaming radiation through containment penetrations. No violations were identified, and one open item was generated (Detail 5.c) which addresses the possibility of radiation scatter from a short length of exposed sample tubing at the PASS station on elevation 45'-0" in the Auxiliary Building.

DETAILS

1. Principal Contacted

   Baltimore Gas and Electric Company (BG&E)

   R. Esenwine, Civil Engineer, Electric Engineering Department (EED)
   N. Millis, General Supervisor, Radiation Safety
   * B. Montgomery, Licensing Engineer, EED
   R. Niedzrelsh, Operations
   E. Reimer, Plant Health Physicist
   B. Rudell, Engineer, Technical Support
   * L. Russell, Plant Superintendent
   * L. Salyards, Operational Licensing and Safety
   R. Sprecher, Plant Chemistry
   J. Tiernan, Manager, Nuclear Power Department
   S. Willats, Project Management

   Bechtel Power Corporation

   **R. Stakenborghs, Mechanical Group Engineer

   Nuclear Regulatory Commission (NRC)

   R. Architzel, Senior Resident Inspector
   D. Jaffe, Licensing Project Manager
   * D. Trimble, Resident Inspector

   *denotes those present at the exit meeting on March 2, 1984.
   **indicates telephone communication.

   Other licensee employees were also contacted during the course of this inspection.

2. Plant Shielding Design Review - Summary

   a. Background

      As discussed in Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," licensees were required to perform a radiation and shielding design review of spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas (defined as any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident) in which personnel occupancy may be unduly limited by the radiation fields present during post-accident operation of essential systems. This was to be accomplished by provisions for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

These requirements were discussed in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Requirements"; and were issued to all operating plants by NRC letters dated September 13 and October 30, 1979, and finally incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in guidance were described in an NRC letter to all licensees dated September 5, 1980, and were subsequently described in Item II.B.2 of NUREG-0737. Lastly, NRC Generic Letter 82-05 dated March 17, 1982 requested reconfirmation of licensee schedules for completing Item II.B.2 of NUREG-0737.

The October 30, 1979 NRC letter indicated that plant shielding design reviews were among those TMI items for which post-implementation NRC review is acceptable. Although prior NRC approval was not required, licensees were required to document their implementation of Item II.B.2, in the form of a summary of the shielding design review, by January 1, 1980. The summary was to include:

(1) Specification of source terms used in the evaluation, including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in the analysis to contain high levels of radioactivity in a post-accident situation;

(3) Specification of areas where access is considered necessary for vital system operation after an accident; and,

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

NUREG-0737 did not require licensees to submit this summary to the NRC. Rather, they were to have available for NRC review the final design details, including any modifications resulting from the shielding review. Modifications were to be completed by January 1, 1982.

b. Scope of Inspection

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. This inspection included licensee submittals to the NRC, a sampling verification of the shielding design methodology and representative calculations, a review of selected emergency procedures to determine if the vital areas where personnel must go are safety accessible, and a review of corrective actions taken or planned by the licensee including plant modifications. The findings of this inspection will form the bases, in part, for the safety evaluation to be issued by NRC's Office of Nuclear Reactor Regulation for NUREG-0737, Item II.B.2.

c. Inspection Findings and Conclusion

The requirements of Item II.B.2 were met, in that vital post-accident areas were appropriately identified and analyzed to assure that no individual whole body doses in excess of 5 rem would be incurred, when integrated over the duration of an accident. A shielding design review was performed and documented, to support the above conclusion. Finally, committed modifications which included the addition of a number of shield walls and motor-operated core flush valves, were found to be installed and properly addressed by appropriate emergency procedures and as-built documentation.

3. Licensee Committments and NRC Evaulations

The licensee and NRC letters discussed below were reviewed as part of this inspection to determine those actions completed by BG&E, and the extent of previous NRC inspection or evaluation of the shielding design review for Calvert Cliffs. The licensee's documented committments provided a basis, in part, for an implementation verification of plant modifications necessary to allow access to required vital areas.

a. BG&E Correspondence with NRC

The Calvert Cliffs Nuclear Power Plant shielding design review was summarized in Appendix B to Enclosure 1 of a January 4, 1980 BG&E letter to the NRC. This summary indicated that the shielding study was preliminary, and would be expanded to consider further effects such as streaming from Containment penetrations. Anticipated modifications were also addressed in the January 4, 1980 Summary. The initial study and planned actions were evaluated by the NRC staff, and determined to meet the Category 'A' Lessons Learned requirements for this item (NUREG-0578, Item 2.1.6.b), as discussed in an NRC letter to BG&E dated April 7, 1980 which stated that a detailed evaluation of the study would be performed by NRC at a later date. The licensee subsequently discussed the status of their shielding design review in letters to the NRC dated December 15 and 30, 1980. The completion of plant shielding modifications was addressed in an April 19, 1982 BG&E letter to the NRC.

b. Previous Inspection 82-05

NRC Region I Combined Inspection Report No. 82-05, issued on April 16, 1982, reviewed the modifications committed to be completed by the licensee as part of their plant shielding review. The physical plant modifications were found to be completed, although in some variance with those "considered" in the January 4, 1980, Appendix B Summary. The modifications considered as "necessary" by the licensee were found to be satisfactory, and reflected in as-built drawings, with appropriate procedural changes incorporated and operator training instituted. The finalized modifications included:

-- Added core flush MOV's 1-SI-399, 2-SI-399, 1-CVC-269, 2-CVC-269.

-- New shielding in Emergency Core Cooling System (ECCS), decontamination, and component cooling rooms.

-- New shielding near the elevator on the five foot elevation of the Auxiliary Building.

-- Shielding in the north-south hallway on the five foot elevation of the Auxiliary Building.

The modifications which had been initially considered, but later judged by the licensee to be not required (and therefore not completed) included:

-- New shielding in the Radiation Exhaust Ventilation Equipment rooms, Switchgear room 311, and near the containment airlock

-- Motor operators on MOV SI-400 and CVC-269 (instead, new MOV's were added as bypasses around these valves)

Some final shield wall locations differed from those originally proposed. Also, sampling modifications (containment air and reactor coolant) were not complete at the time of Inspection 82-05, and were to be examined as part of the PASS installation. Inspection 82-05 did identify a violation of 10 CFR Part 50.59 in that no written safety evaluation was performed to address the addition of the new core flush/bypass MOV's and piping. The corrective action associated with that violation was subsequently completed by BG&E and satisfactorily resolved with NRC.

4. Shielding Design Calculations and Dose Estimates

a. Initial BG&E Design Review

The inspector reviewed the licensee's calculational summary documented in Appendix B to Enclosure 1 of their January 4, 1980 letter (Lundvall to Eisenhut) entitled "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations (NUREG-0578, Item 2.1.6.b)".

This study identified systems most likely to be contaminated (outside containment) following a severe design basis accident. Rooms and areas affected by these large sources were identified, and evaluated in light of those operator recovery actions outside control room (including time required and access routes) which would be necessary following an accident. The study was not based on detailed shielding calculations, but rather was representative of "good engineering estimates" (accurate to within one order of magnitude). The study was only based on post-accident recirculation systems, with letdown isolation accounted for and hence no contamination of CVCS (Volume Control Tank), Waste Processing (Waste Gas) or Shutdown Cooling systems.

Finally, the conclusions of the study indicated that further refined analyses were to be continued to consider both units (i.e. an accident in one unit and shut down of the unaffected unit), as well as the contribution of sources inside containment.

The study identified the following potential problem areas and proposed solutions or "modifications considered":

-- Core Flush (install motor operators on valves SI-400 and CVC-269)

-- Containment Air Sampling (install new sampling station)

-- Reactor Coolant Liquid Sample (install PASS)

-- ECCS Pump Rooms (install additional 8-inch block shielding at doorways)

-- Component Cooling Pump Rooms;
Radiation Exhaust Ventilation Rooms;
Decontamination Room;
Passages 202 and 212; (install additional 8-inch block shields)

-- Containment Purge Air
Discharge Chase in Switchgear
Room 311 (install additional shielding)

-- Streaming from Containment Personnel Airlock (install 8-inch block shield at doorway)

-- Streaming from Pipeway Adjacent to Control Room (install lead shield in control room or a platform shield at lower elevation)

-- Stair No. AB-2 (install shielding at el. 5' and 27')

The study also identified the following vital areas, where actions outside control room would be required as part of the recovery from a severe accident:

-- Auxiliary Building Room 203, [illegible] Area (el. 5'-0"); core flush valve SI-400, Containmen[illegible] sa[illegible]ling

-- Auxiliary Building Room [illegible]3 (el 4[illegible]'-0"); reactor coolant sampling

-- Auxiliary Building Rooms 522 and 519 (el 69'-0"); hot laboratory and counting facility for coolant analysis

-- Control Room and Technical Support Center

b. Bechtel Calculations

The inspector reviewed the assumptions, methodology and results of detailed calculation packages which were compiled by the licensee's architect-engineer, Bechtel Power Corporation. The calculations were performed by the Bechtel Mechanical Group, and included the development of source terms, the estimation of dose rates using various computer codes, and the verification of the shielding computer codes which were utilized.

The computer codes used to estimate gamma shielding dose rates included the Bechtel-developed code PIPEND, and the standard industry code QAD-CG. PIPEND uses the Rockwell (Reactor Shielding Design Manual) solutions for dose rates from cylindrical volumetric sources with a capability for modelling of nine energy bins and multi-layered buildup using Broder's formula (with the Capo coefficients), and was used to model contaminated piping systems and equipment. QAD-CG is a three dimensional point-kernel code which was used to model the containment atmosphere as a source. Both codes were found to be well-documented and appropriately utilized.

Source term development was documented in Mechanical Group Calculation M-81-25. A Combustion Engineering ORIGEN program run for total core inventory at the end of an 18 month cycle at 2700 MWt was used as input to the Bechtel-developed code PROCESS which, in turn, was used to generate time-dependent source terms for the Containment atmosphere and recirculation fluid systems. Three distinct dilution volumes ("sources A, B and C") were considered, to distinguish between containment air and reactor coolant samples and ECCS recirculation fluids. The source term development was clearly-documented, used the proper combinations of noble gas/halogen/solid release fractions, and was accomplished by a previously verified computer program.

Of particular note was the treatment of gamma streaming through containment penetrations from airborne concentrations inside Containment. Bechtel Calculation M-80-23 documented the use of a program entitled BARF*PENN which numerically solved an integral equation resulting from the "ray-analysis" technique (The Engineering Compendium on Radiation Shielding, page 514) of handling gamma streaming through straight cylindrical duct. The QAD-CG results for internal Containment radiation levels were assumed as infinite-planar sources at one end of a penetration, and the BARF*PENN code calculated the streaming contribution at the outboard penetration location (with no credit taken for the 1/4-inch thick steel caps on either end of the penetration). In this fashion, the dose rates at rooms near or affected by penetrations (including personnel and equipment airlocks) were conservatively estimated and uniquely detemined. These calculations were critical since their contibution to projected radiation levels throughout the plant was, in many cases, the controlling dose rate. For example, even at 10 hours post-accident, dose rates immediately outside of penetrations (such as penetration rooms, the emergency personnel airlock, etc) were estimated to be on the order of 10,000 to 50,000 R/hr. By comparison, the direct (through-wall) dose rate outside of the Containment wall at time zero was estimated on the order of 10 R/hr. This illustrated the importance of the streaming contribution in some vital areas, even though these dose rates are reduced significantly (by inverse square) with distance away from the penetration.

Finally, the methods used to generate post-accident dose rate zone maps, so as to specify shield locations and thicknesses for operator access throughout the plant, were reviewed in Mechanical Group Calculation M-80-14. Source 'C' (recirculation fluids) was utilized principally, and dose rate "factors" calculated for various piping diameters and standard concrete shield thicknesses at times 0, 10 and 100 hours post-accident. Individual rooms (e.g. Component Cooling Rooms 228 and 201, ECCS Pump Rooms 101 and 102) were analyzed in detail, using actual line sizes and piping lengths, to come up with room and corridor dose rates which would then dictate the placement of required shield walls. The dose rates generated were founded on sound and consistent assumptions, and were judged to be both reasonable and conservative.

The consideration of not only obvious sources (such as recirculation piping), but the not-so-obvious sources (such as penetration streaming or doses to operators performing actions on the unaffected unit) was indicative of the thorough approach evident in the calculational sample inspected.

c. Additional Discussions

Certain other topics were discussed which were not addressed by specific calculation, to ascertain any additional areas in the plant (or onsite) where operators may have to go to or pass through during the course of an accident, and in which a significant dose could be encountered. These included areas in the vicinity of the Security Building, diesel fuel oil tanks, plant access road, Turbine Hall, and various non-vital areas necessary for restoration of equipment assumed to be lost or unavailable (e.g. non-vital electrical loads) following a design basis accident.

All of these additional areas were reasoned to be either non-essential, beyond the design basis, or radiologically insignificant relative to the NUREG-0737 criterion of 5 rem whole body intergrated dose (10 CFR Part 50, Appendix A, GDC19). The inspectors were therefore satisfied that all vital areas had been appropriately identified, and that operator access had been considered and adequately analyzed to support the conclusion that no whole body individual doses in excess of 5 rem would be incurred.

5. Plant Modifications

Plant modifications necessary to meet NUREG-0737, Item II.B.2, involved both (a) the addition and shielding in the Auxiliary Building for post-accident access, and (b) modifications to ECCS core flush piping/valves for remote operation (to obviate the need for such access). All modifications were found to be installed in satisfactory compliance with Item II.B.2.

a. Facility Change Request 80-1009

Engineering package FCR 80-1009, "Access to Vital Areas-Design Changes and Shielding", documented the design of various shield walls and motor-operated valves associated with the licensee's shielding design review. This package included safety evaluations (as required by 10 CFR Part 50.59) for the shield walls, but as described in detail 3.b of this inspection, did not originally contain a written safety evaluation for the addition of MOV's SI-399 and CVC-269. Safety evaluations were subsequently completed prior to this inspection, for these valve/bypass piping additions.

The valves were provided with remote operation capability to facilitate core flush operations which would be required at approximately 8 to 11 hours after an accident involving a reactor coolant cold leg break wherein increased reactor boron concentrations would necessitate the flush. Motor-operated valves were installed, rather than adding new motor operators to the existing manual valves. Their corresponding effect on pipe stress due to seismic and other loads was taken into account by Bechtel analysts, and found to be within Code allowables given the existing support configuration. A new MOV SI-399 was added to existing cross-connection piping between Low Pressure Safety Injection and Shutdown Cooling systems on both units. New MOV CVC-269, with associated piping, was added as a bypass path around existing manual CVC valves in the Chemical and Volume Control systems on both units.

Shield walls were added at a number of locations to allow for Auxiliary Building access, principally for operations associated with the unaffected unit following an accident. Free-standing concrete masonry unit (8-inch block), and in some cases 4-inch lead brick walls, were emplaced at the following locations:

-- ECCS Pump Rooms

-- Component Cooling Pump Rooms

-- Passageways 202 and 212

-- Decontamination Room 210

-- Stairwell No. AB-2

Shield wall loadings on Auxiliary Building structural floor slabs were considered by comparing stresses created by the new walls with existing capacities, as well as with seismic wall responses. No unreviewed safety questions were found by Bechtel and BG&E reviewers during this comparison.

The shield walls were added to eliminate potential radiation streaming paths through existing doorways and passages to cooridors and other access areas. The walls were intended to attenuate radiation levels to at least an equivalent amount as that afforded by existing building walls. In one case, a platform shield was added at the ceiling of a CCW pump room to prevent a local control room hot spot created by streaming from a pipe chase containing recirculation system piping.

b. Dose Rate Zone Maps

Post-accident radiation dose rates and shield locations were depicted on plan drawings A-180 through 184 for the five elevations of the Auxiliary Building. These drawings were intended as aids in determining access routes and post-accident operational decisions, and were not used for construction. The drawings were included in the January 4, 1980 BG&E letter to NRC, and document final shield emplacements and potentially high radiation areas and streaming paths following an accident. Review of the drawings identified two shield walls in passage 202 on elevation 5'-0" which were incorrectly depicted as concrete block construction (rather than the installed lead brick). This discrepancy was corrected by the licensee in a formal drawing (A-181) revision during conduct of the inspection. The drawings were useful in the inspectors' evaluation of the extent and adequacy of the Calvert Cliffs shield design review.

c. Sampling and Analysis

The Post Accident Sampling Station (PASS), located on Auxiliary Building elevation 45'-0", was observed to be in an area with predicted radiation levels on the order 1.0 R/hr during the first four hours following an accident. Although the personnel exposure while taking a post accident sample would not be expected to exceed the 5 rem guideline of NUREG-0737 Item II.B.2, a small section of sample tubing was identified as causing a potential scatter path (from either the nearby wall or ceiling) to an operator at either the hydrogen monitoring or PASS control panels. The licensee committed to evaluate the effects of this scatter (UNR 84-05-01). The implementation of the PASS system, itself the subject of a separate NUREG-0737 item, will be reviewed by NRC in a separate inspection.

The Health Physics, Counting Room and Hot Lab areas on Auxiliary Building elevation 69'-0" were reviewed and discussed with relation to methods necessary to analyze post-accident samples. Initial dose rates were predicted on the order of 10 R/hr in this area during the first hour, primarily due to streaming contributions from the nearby

Personnel Air lock. These levels drop by an order of magnitude during the following 24 hours such that intermittent occupancy for periods of from 15-30 minutes would still result in individual whole body exposures within the 5 rem guideline. Credit for existing structural walls and the intervening CCW head tank eliminated the need for erection of a new shield wall between the airlock and this area.

d. Control Room and Technical Support Center

With the exception of the platform shield referred to in detail 5.b, no modifications were necessary in these areas, and dose rate levels in the control room and TSC were determined to be within the NUREG-0737 guidelines.

6. Vital Area Accessibility-Procedure Review

The inspectors reviewed five emergency procedures which may be implemented by the licensee in the event of a loss of coolant accident. The review included a walk-through of the procedures with control room operators to determine the capability to perform procedural steps and the accessibility of manual valves or breakers that may require local operation. The review also considered an assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included:

--	EOP-1	Reactor Trip, Rev. 14
--	EOP-5	Loss of Reactor Coolant, Rev. 17
--	EOP-12	Loss of Flow/Natural Circulation, Rev. 8
--	EOP-15	Loss of AC Power, Rev. 6
--	EOP-5	Long Term Cooling Core Flusn, Rev. 6

In addition, a brief discussion of operations expected for the unaffected unit (after a design basis accident) was performed. Options available to plant operators, such as when to go on shutdown cooling, were discussed. This activity would require an estimated 4-5 man effort for no more than 45 minutes, and could occur as early as 20 hours after shutdown of the unaffected unit. While this activity would involve access to the ECCS pump and CCW rooms in the Auxiliary Building, the plant does have the option of remaining in a Hot Standby condition indefinitely.

Based on the above walk-throughs, the inspectors determined that the procedures could be performed from the control room, with appropriate provisions to assure controlled access to vital areas.

7. Exit Interview

The inspectors met with licensee representatives on March 2, 1984 to discuss the findings and conclusions of this inspection.