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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 7 1980

NRC PDR

Docket No. 50-318

The Honorable Gladys Spellman
United States House of Representatives
Washington, D.C. 20515

Dear Congresswoman Spellman:

Your letter of May 15, 1980 to Mr. Carlton Kammerer of the NRC Congressional Affairs Office has been forwarded to me for response. The following information is provided for your use in answering the May 1, 1980 questions of your constituent, Mr. John A. May. The subject of this request is the radioactive gas leakage at the Calvert Cliffs Nuclear Power Plant.

Mr. May's questions referred to the time approximately one week before his letter of May 1, 1980. A radioactive gas release occurred close to this time on March 27, 1980. The cause of that release was a poor gasket seal on the Unit 2 degassifier relief valve. This release activated local area radiation alarms in the Auxiliary Building. Consequently, the building was evacuated in accordance with the Plant Emergency Procedures. The maximum instantaneous release rate was calculated to be about 4% of our Technical Specification Limit. It is estimated that the maximum integrated dose would be approximately 0.0007 mrem at a point south of the site. Further detailed information on this event can be found in Enclosure 1, Nuclear Regulatory Commission Inspection Report 50-317/80-03 and 50-318/80-03.

Activities of this agency and our licensees following the accident at Three Mile Island (TMI) have been directed toward further improvements in commercial nuclear power plant design, construction, and operation. These improvements are directed toward both preventing accidents of this severity and toward mitigating the consequences of such an accident in the event it should occur. Recently Baltimore Gas & Electric issued a press release (Enclosure 2) describing changes they have made following the TMI accident. Our safety evaluation of a number of these actions taken at Calvert Cliffs Unit Nos. 1 and 2 is provided for your information in Enclosure 3.

We trust that this letter will provide you with the information needed to satisfactorily respond to questions from your constituents.

Sincerely,

(Signed) T. A. Rehm

William J. Dircks
Acting Executive Director
for Operations

Enclosures:
1. NRC IE Inspection Report
   50-317/80-03, 50-318/80-03,
   April 25, 1980.
2. BG&E News Release, March 28, 1980.
3. NRC Staff's Evaluation of Category
   "A" Lessons Learned Implementation,
   April 7, 1980.
4. Letter dated May 1, 1980 from
   John A. May.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

ENCLOSURE 1

APR 25 1980

Docket Nos. 50-317 and 50-318

Baltimore Gas and Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President, Supply
P. O. Box 1475
Baltimore, Maryland 21203

Gentlemen:

Subject: Inspection Nos. 50-317/80-03 and 50-318/80-03

This refers to the inspection conducted by Mr R. Architzel of this office on March 10-28, 1980 of activities authorized by NRC License Nos. DPR 53 and DPR 69 at the Calvert Cliffs Nuclear Power Plant, Lusby, Maryland and to the discussions of our findings held by Mr Architzel with Mr L. Russell of your staff at the conclusion of the inspection.

Areas examined during this inspection are described in the Office of Inspection and Enforcement Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of this inspection, no items of noncompliance were observed.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractor) believe to be proprietary, it is necessary that you make a written application within 20 days to this office to withhold such information from public disclosure. Any such application must be accompanied by an affidavit executed by the owner of the information, which identifies the document or part sought to be withheld, and which contains a statement of reasons which addresses with specificity the items which will be considered by the Commission as listed in subparagraph (b) (4) of Section 2.790. The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit.

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If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

No reply to this letter is required; however, if you should have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Eldon J. Brunner, Chief
Reactor Operations and Nuclear
Support Branch

Enclosure: Office of Inspection and Enforcement Inspection Report Numbers 50-317/80-03 and 50-318/80-03

cc w/encl:
R. M. Douglass, Manager, Quality Assurance
L. B. Russell, Chief Engineer
T. Sydnor, General Supervisor, Operations QA
R. C. L. Olson, Senior Engineer
K. H. Sebra, Principal Engineer

bcc w/encl:
IE Mail & Files (For Appropriate Distribution)
Central Files
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Technical Information Center (TIC)
REG:I Reading Room
State of Maryland (2)
Dr. Steven Long, Administrator for Nuclear Evaluations
R. Architzel, RRI
D. Beckman, RRI
C. Cowgill, RRI
R. Conte/D. Haverkam RRI
J. Higgins, RRI
L. Norrholm, RRI
T. Rebelowski, RRI
R. Gallo, RRI
J. Shedlosky, RRI

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

APR 30 1980

Region I

Report No. 50-317/80-03
50-318/80-03

Docket No. 50-317
50-318

License No. DPR-53
DPR-69

Priority --

Category C
C

Licensee: Baltimore Gas and Electric Company
P.O. Box 1475
Baltimore, Maryland 21203

Facility Name: Calvert Cliffs Nuclear Power Station, Units 1 and 2

Inspection at: Lusby, Maryland

Inspection conducted: March 10-28, 1980

Inspectors: E. C. McCabe, Jr., for 4/24/80
R. Architzel, Resident Reactor Inspector — date signed

date signed

date signed

Approved by: E. C. McCabe 4/24/80
E. C. McCabe, Jr., Chief, Reactor Projects Section No. 2, RO&NS Branch — date signed

Inspection Summary:

Inspection on March 10-28, 1980 (Combined Report Nos. 50-317/80-03 and 50-318/80-03)

Areas Inspected: Routine, onsite regular and backshift inspection by the resident inspector (16 hours, Unit 1; 16 hours, Unit 2). Areas inspected included the control room and the accessible portions of the auxiliary, turbine, service, and intake buildings; radiation protection; physical security; fire protection; plant operating records; and reporting to the NRC.

Results: No items of noncompliance were identified.

DUPLICATE DOCUMENT

Entire document previously entered into system under:

ANO 8006170816

No. of pages: 12

Region I
(Rev. Apr

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 3

April 7, 1980

Dockets Nos. 50-317
and 50-318

Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

Enclosed is the staff's evaluation of the implementation of Category "A" Lessons Learned requirements (excluding 2.1.7a) at Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on February 19, 1980.

Based on our evaluation, we conclude that the implementation of the Category "A" requirements at Calvert Cliffs, Units Nos. 1 and 2 is acceptable. Certain items, identified in the evaluation, will be verified by the Office of Inspection and Enforcement.

This evaluation does not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the Category "A" review. You should be considering the proposal of such Technical Specifications. We will be in communication with you on this item in the near future.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure: Evaluation

cc w/enclosure: See next page

cc w/enclosure(s):

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Mr. Bernard Fowler
President, Board of County
Commissioners
Prince Frederick, Maryland 20768

Director, Technical Assessment
Division
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(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CALVERT CLIFFS UNITS 1 & 2

EVALUATION OF CATEGORY "A" LESSONS LEARNED

IMPLEMENTATION

Introduction

By letters dated October 17, November 20, December 5 and 14, 1979; January 4, 22, and 30, February 1 and 29, and March 12 and 17, 1980, Baltimore Gas and Electric Company (BG&E or the licensee) submitted documentation of the actions taken at Calvert Cliffs Units Nos. 1 and 2 (the plant) to implement the requirements resulting from TMI-2 Lessons Learned. To facilitate our review of the licensee's actions, members of the staff visited the plant on February 19, 1980.

Evaluation

The NRC's Category "A" Lessons Learned Requirements and acceptance criteria are documented in NUREG-0578 and NRC letters dated September 13 and October 30, 1979. The number designation of each item is consistent with the identifications used in NUREG-0578.

2.1.1 Emergency Power Supplies

Pressurizer Heaters

The pressurizer proportional heaters are supplied from a Class 1E bus. Two banks of pressurizer backup heaters are fed from a 480V engineered safety feature load center. The redundant capacity of 450kw provided by the backup and proportional pressurizer heaters is sufficient to maintain natural circulation. The load center breakers supplying the proportional heaters are tripped on an undervoltage signal after a loss of offsite power. These breakers can be closed manually when required after diesel generator load sequencing is complete.

Power Operated Relief Valves (PORVs) and Block Valves

The motive components of the PORVs are supplied from safety related 480V motor control centers which have a diesel backup. The control components of the PORVs are supplied from safety related 125 VDC battery buses. The motive and control components of the PORV block valves are supplied from safety related 480V motor control centers which have diesel backup. In each case the motive and control power for the block valve is supplied from a power supply train different from that which supplies the associated PORV.

Pressurizer Level Indicator

Two of the pressurizer level instruments for each unit are powered from the vital DC buses and the third is powered from offsite AC power with diesel backup.

The licensee meets the requirements of Category "A" for Item 2.1.1.

2.1.3.a Direct Valve Position Indication

The direct valve position indication of the two power operated relief valves (2 indicators) and two safety valves (1 indicator) have been accomplished with an acoustic monitoring system manufactured by Technology for Energy Corporation (TEC). The TEC 914 Valve Flow Monitor uses an accelerometer as the sensor ahead of a charge converter. The accelerometer converts acceleration to charge, which is then converted to voltage. The processor unit then indicates relative valve flow. The flow indicator located in the control room is a bar indicator made up of 10 light emitting diodes (LEDs) which light up relative to the flow. The system is safety grade and has an alarm in the control room.

Based on the above we find the licensee has met requirement 2.1.3.a.

2.1.3.b Instrumentation for Inadequate Core Cooling

Procedures have been upgraded to aid operators in detection of inadequate core cooling and to assure appropriate actions are taken. Additional procedures to be used by the operator to recognize inadequate core cooling are being developed based on analysis and guidelines required by Item 2.1.9, Transient and Accident Analysis.

The CE Owners Group provided report CEN-117 in response to Item 2.1.9 and Item 2.1.3.b which addresses the instrumentation for detection of inadequate core cooling. CEN-117 concludes that present instrument response during various postulated means of approaching inadequate core cooling yield several significant patterns of indication available to the operator. From these patterns he can detect the approach to inadequate core cooling. In addition, the licensee has submitted a design description of a reactor vessel water level measurement system. This system, as well as CEN-117, will be reviewed at a later date.

Subcooling Meter

BG&E has installed a subcooled margin monitor (SMM). The SMM is a microcomputer based instrument which continuously displays the subcooled margin to saturation. It is designed for use as a post-accident monitoring instrument. The SMM is a Safety Class 1, Seismic Class 1, Quality Class 1 instrument and is designed to meet IEEE Stds. 344-1975 and 323-1974.

The SMM provides the operator with continuous digital display of either the pressure or temperature margin to saturation. An alarm is provided as part of the SMM. Temperature inputs are from two hot legs and two cold legs per meter. The range is 212°-705°F. The RTD input sensors are seismic qualified and are part of the existing Reactor Protection System (RPS). There are two pressure inputs per meter with the range of 15-2300 psia. Pressure sensors are seismic qualified and are part of the existing RPS.

Installation of the new dual transmitters has had compatibility problems with the existing equipment. One channel per meter is now operational. Operation of the second channel will be delayed four weeks until receipt of compatible equipment for the remaining channel.

At present the plant computer calculates the margin to saturation. The computer uses incore thermocouples.

The licensee meets the requirements of Category "A" for Item 2.1.3.b. Operation of the second channel of the subcooling meter will satisfy the redundancy requirement of Category "A" Item 2.1.3.b. The redundancy at the present is provided by the plant computer. Our Office of Inspection and Enforcement (IE) will verify modifications to the SMMs have been completed.

2.1.4 Containment Isolation

The NRC requirements are that the licensee is to: (a) carefully reconsider their determination of which systems should be considered essential or non-essential for safety; (b) modify systems as necessary to isolate all non-essential systems by automatic, diverse, safety grade isolation signals; and (c) modify systems as necessary to assure that the resetting of the containment isolation signal does not cause the inadvertent reopening of containment isolation valves.

The licensee's submittals of November 20, 1979 and February 29, 1980 identified the essential and non-essential systems and provided the bases for the essential system classification.

The containment isolation system has been modified so that containment penetrations associated with non-essential systems are either locked closed or are automatically isolated on diverse parameters including the safety injection actuation signal (SIAS) and the containment isolation signal. SIAS is initiated on either high containment pressure or low pressurizer pressure.

The isolation valve control circuits have been modified to prevent inadvertent opening after resetting the isolation signal. This has been accomplished by wiring the valves' control switches so as to form a reset permissive, i.e., resetting will only be accomplished if all the isolation valves' handswitches in a given circuit are in the isolation position. With this design, deliberate operator action is required to reopen each isolation valve after the isolation signal is reset. In addition, each circuit can be bypassed by a bypass switch located in a locked closed cabinet in the cable spreading area. This bypass provides a backup reset capability in the event of a failure in the reset circuitry.

The above modifications have been made for all the isolation valves, except the oxygen and reactor coolant system sample valves. Physical constraints require that this reset design objective be accomplished using additional lock-in relays that will interrupt the power to the valves and fail them closed once an isolation signal is received; the circuit will keep the valves closed until the operator repositions the valve control switch to the open position. These modifications will be made within 30 days of receipt of the required equipment and prior to June 1, 1980. Our conclusion is that with the modification of the sample valves the licensee's containment isolation design meets the NUREG-0578 Section 2.1.4 containment isolation requirements and is therefore acceptable. Our Office of IE will verify that modifications to the sample valves control circuits have been completed.

2.1.5.a <u>Dedicated Penetrations for External Recombiners or Post-Accident External Purge System</u>

The NRC's position is that dedicated containment isolation systems should be used for the external recombiners or purge systems that meet redundancy and single failure requirements. This requirement does not apply to the licensee since recombiners located wholly within the containment are used.

2.1.5.c <u>Recombiner Procedures</u>

The NRC's position is that the procedures for use of the recombiners be reviewed considering shielding requirements and personnel exposure limitations.

The plant utilizes recombiners located inside the containment. Controls for operating the recombiners are located inside the control room. During the site visit we discussed the licensee's review of the recombiner operating procedures and agreed that no modifications are required.

We have concluded that the licensee has met the NUREG-0578 requirements for review of the recombiner procedures, Section 2.1.5.c.

2.1.6.a <u>Systems Integrity</u>

The licensee has provided a list of those systems which he has determined may contain radioactivity following an accident. These systems are the safety injection, containment spray, shutdown cooling, containment sump recirculation, and reactor coolant and containment atmosphere sampling system. He has also provided a description of the immediate leak reduction program which included walk down inspections to identify leakage, cleanup and repair of these systems. The licensee has also measured and reported the final system leak rates to the NRC.

The licensee has established a surveillance test program for the systems which may contain activity following an accident which includes testing once per refueling cycle.

In order to assure that radioactivity will be restrained to those systems specified and to allow for operation of the reactor coolant pumps, the licensee has committed to incorporate a procedure for dumping the reactor coolant drain tank to the containment sump. IE will assure that this procedure is in place.

Our October 30, 1979 clarification letter requests the licensee to include a review of potential release paths due to design and operator deficiencies as discussed in the October 17, 1979 letter regarding North Anna. The licensee has analyzed their plant with regard to the North Anna Incident and concluded no corrective action is necessary.

Based on the above information, we conclude that the licensee has met the Category "A" requirements for this item.

2.1.6.b Plant Shielding Review

The licensee's January 4, 1980 submittal includes a design review of plant shielding and environmental qualification of equipment. The licensee has performed the design review assuming the systems identified in Item 2.1.6.a contain radioactivity. The licensee has used the source term as specified in the October 30 letter for his review. The licensee has determined high radiation areas and identified components which may be affected. The licensee also discussed possible modifications in the affected areas. The licensee has stated that components which may be adversely affected will be identified and corrective actions completed by January 1, 1981, if equipment is available. They have also identified areas where access may be required. For these areas, corrective actions will be taken to assure that the necessary functions can be performed. A detailed evaluation of the submittal will be performed at a later date. We conclude that the licensee has met the Category "A" requirements for this item.

2.1.7.b Auxiliary Feed Flow Indication

Indication of auxiliary feed flow provided to each steam generator is safety grade and is powered from vital power supplies. As a backup to the safety grade auxiliary feed flow indication each steam generator has four safety grade level channels with control room readout.

We find this satisfies the Category "A" requirements for this item.

2.1.8.a Post-Accident Sampling

The licensee's January 4, 1980 submittal contains a design review of the plant sampling capability for primary coolant and containment air samples assuming a source as specified in NUREG-0578.

The licensee has incorporated interim procedures for obtaining and analyzing a reactor coolant sample following an accident. They also incorporated interim procedures for obtaining and analyzing a containment air sample with the existing system. IE will assure that the procedure is in place.

The licensee has provided a preliminary design of the proposed plant modifications necessary to meet the Category "B" requirements for reactor coolant and containment atmosphere sampling.

Based on the above, we conclude the licensee has met the Category "A" requirements for this item.

2.1.8.b High Range Radiation Monitors

The licensee has provided equipment and implementing procedures to quantify noble gas release rated from the plant vent, condensor air ejector and steam safety release and dump valves if the existing instrumentation goes offscale.

IE will assure that the procedures are in place.

The licensee has provided a description of his system to be used to determine radioiodine and particulate effluents. They have also modified existing procedures for obtaining effluent samples to allow for potential high dose rate levels following an accident. IE will assure that the appropriate procedures have been modified.

Based on the above information, we conclude that the licensee has met the Category "A" requirements for this item.

2.1.8.c Improved Iodine Instrumentation

The licensee has committed to use charcoal cartridges to collect air samples in occupied areas. The sample cartridges will be counted using one of the plant GeLi systems, which has been dedicated for analyzing air samples. The samples can be counted in a short enough time to allow for operator protection.

This system meets the requirements of NUREG-0578. The licensee has also provided assurance that all areas occupied by essential personnel will be monitored. Therefore, we conclude that the licensee meets the requirements of NUREG-0578, Item 2.1.8.c.

2.2.1.a Shift Supervisor Responsibilities

The NRC requirement for this item is to revise, as necessary, the responsibilities of the Shift Supervisor such that he can provide command oversight of operations and perform management review of ongoing operations that are important to safety.

During the staff's site visit we reviewed the licensee's management directives and revisions to their administrative procedures, QAP-25, Plant Operations. We have determined that these directives and procedures satisfy the requirements of NUREG-0578, Item 2.2.1.a, for delineation of Shift Supervisor responsibilities.

2.2.1.b Shift Technical Advisor (STA)

The NRC requirement is for the licensee to provide an on-shift advisor to the Shift Supervisor to serve the two functions of accident assessment and operating experience assessment. As a supplement to the operating staff, the STA must be available to the control room to assist in diagnosing an off-normal event.

To satisfy the staff's requirements, the licensee has implemented a program, described in the December 14, 1979 submittal, wherein the staff for the two plants is increased to include one additional (SOL) holder or degreed engineer. This individual would satisfy the required STA accident assessment function. The operating experience assessment function of the STA is satisfied by a standing committee staffed by onsite engineers and augmented as necessary by engineers from the Engineering Department.

We have reviewed the licensee's submittal describing their STA programs. In addition, during the site visit we discussed the program with the licensee and determined that a satisfactory STA program is in operation. We find that their STA program is in agreement with the staff's requirements described in Section 2.2.1.c of NUREG-0578 and is therefore acceptable.

2.2.1.c Shift and Relief Turnover Procedures

The NRC requirement is for the licensee to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the offgoing and oncoming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the oncoming shift.

The licensee's submittal indicated that checklists and logs have been developed which satisfy our acceptance criteria. Further, he has established a system to evaluate the effectiveness of the shift turnover procedure.

During the site visit our check of the revised shift turnover procedure checklists and logs confirmed that the licensee has addressed this position.

We conclude that the licensee has satisfied the requirements of Item 2.2.1.c related to shift turnover procedures. Adequacy of the checklists and logs will be performed by the Office of IE and will be documented by appropriate Inspection Reports.

2.2.2.a Control Room Access

Procedure QAP-25 establishes the authority of the person in charge of the control room to limit access to the control room. The licensee has implemented procedures in the Site Emergency Plan defining lines of communication and authority between the control room and the different emergency centers. Procedures also exist which establish the line of succession for personnel in charge of the control room and limit the personnel in charge to those holding a SROL.

We find the licensee has met the Category "A" requirements for this item.

2.2.2.b Technical Support Center (TSC)

A TSC has been established on the 55' Elevation of the Log and Test Instrument Room. This area is adjacent to the control room and is included in the Control Room Ventilation System. This room is directly accessible from the turbine building or the control room. The dedicated communications to the control room and Emergency Operations Center will be provided by dedicated channels of the plant paging system. The licensee has installed an additional head set in the TSC to permit simultaneous communication between the TSC and the control room of the TSC and the Emergency Operations Center. Plans and procedures for engineering/management support of the TSC have been established and are contained in the Site Emergency Plan. The licensee has located the "Record Retrieval Center" in the TSC. This will contain all plant physical data, drawings and parameters. Selected plant parameters can be read out in the TSC via temporary recorders connected to a portion of the startup/physics test panel. The licensee has provided details for the long term TSC. We conclude that this satisfies the Category "A" requirements for this item.

2.2.2.c Onsite Operational Support Center (OSC)

The licensee has established an OSC from which there is the capability to communicate with the control room. The OSC is located in the service building, across the turbine deck from the control room. The Site Emergency Plan has been modified to establish lines of communication and management of the OSC.

We consider this to meet the requirements for this item.

NRR Reactor Coolant System Venting

The licensee has proposed a design for venting of the reactor coolant system in fulfillment of the Short-Term Lessons Learned Requirement.

Conclusion

Based on the above, subject to our Office of IE verification as noted, we find that implementation of the Category "A" Lessons Learned Requirements at Calvert Cliffs Units Nos. 1 and 2, is acceptable.

Dated: April 7, 1980

ENCLOSURE 4

1980

6707 Gateway Blvd.
District Heights, Md. 20028

May 1, 1980

Mrs. Gladys Spellman
U.S. Representative
Cannon Bldg - Dist. Office
6525 Belcrest Rd.
Hyattsville, Md.

Dear Mrs. Spellman:

I am writing this letter in reference to the nuclear power plant which is located in Calvert Cliffs, Maryland.

A week ago, I read in the papers that there was a malfunction in the plant. The paper said that a small amount of radioactive gas leaked out of the nuclear power plant because of a malfunction inside of the building. Why was this caused? I would like to know what exactly happened in the plant and what measures are being taken to protect the nearby community from a possible incident that happened at Three Mile Island.

The paper never really explained what happened at the Calvert Cliffs plant, so I would like to have an answer to the accident at the plant.

Very truly yours,

John A. May

John A. May