UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 22, 1996

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M94205) AND UNIT NO. 2 (TAC NO. M94206)

Dear Mr. Cruse:

The Commission has issued the enclosed Amendment No. 213 to Facility Operating License No. DPR-53 and Amendment No. 190 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 30, 1995, as supplemented on March 15, 1996.

The amendments allow the installation of tube sleeves as an alternative to plugging for repairing steam generator (SG) tubes using repair techniques developed by Westinghouse Electric Corporation. In your November 30, 1995, letter, you also requested approval of repair techniques developed by ABB Combustion Engineering, Inc., for repairing SG tubes. We are still reviewing that portion of your request.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Daniel G. McDonald, Jr.

Daniel G. McDonald, Jr., Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosures:
1. Amendment No. 213 to DPR-53
2. Amendment No. 190 to DPR-69
3. Safety Evaluation

cc w/encls: See next page

DATED: March 22, 1996

AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-53-CALVERT CLIFFS UNIT 1
AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-69-CALVERT CLIFFS UNIT 2

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
S. Shankman
S. Little
D. McDonald
OGC
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
H. Conrad
ACRS
PD plant-specific file
C. Cowgill, Region I
J. Strosnider

cc: Plant Service list

?80017

DF01

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

cc:

President
Calvert County Board of
Commissioners
175 Main Street
Prince Frederick, MD 20678

D. A. Brune, Esquire
General Counsel
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203

Jay E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW
Washington, DC 20037

Mr. G. L. Detter, Director, NRM
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

Resident Inspector
c/o U.S. Nuclear Regulatory
Commission
P. O. Box 287
St. Leonard, MD 20685

Mr. Richard I. McLean
Administrator - Radioecology
Department of Natural Resources
580 Taylor Avenue
Tawes State Office Building
B3
Annapolis, MD 21401

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Joseph H. Walter
Engineering Division
Public Service Commission of
Maryland
6 St. Paul Centre
Baltimore, MD 21202-6806

Kristen A. Burger, Esquire
Maryland People's Counsel
6 St. Paul Centre
Suite 2102
Baltimore, MD 21202-1631

Patricia T. Birnie, Esquire
Co-Director
Maryland Safe Energy Coalition
P. O. Box 33111
Baltimore, MD 21218

Mr. Larry Bell
NRC Technical Training Center
5700 Brainerd Road
Chattanooga, TN 37411-4017

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
License No. DPR-53

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated November 30, 1995, as supplemented on March 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. DPR-53 is hereby amended to read as follows:

2. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Susan Frant Shankman

Susan Frant Shankman, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated November 30, 1995, as supplemented on March 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. DPR-69 is hereby amended to read as follows:

2. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Susan Frant Shankman, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 1996

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 213 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 190 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-16	3/4 4-16
B3/4 4-4	B3/4 4-4
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6*
B3/4 4-7	B3/4 4-7*

*Indicates rollover pages.

3/4.4 **REACTOR COOLANT SYSTEM**

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE**.

APPLICABILITY: **MODES** 1, 2, 3 and 4.

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

    1. All nonplugged tubes that previously had detectable wall penetrations (> 20%), and

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a. original tube wall..................................40%
   b. Westinghouse laser welded sleeve wall..............40%

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following process:

    a) Westinghouse Laser Welded Sleeving as described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per Specification 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

    1. Number and extent of tubes inspected.

    2. Location and percent of wall-thickness penetration for each indication of an imperfection.

    3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug or repair defective tubes and inspect 2S tubes in each other SG. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug or repair defective tubes. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon perminute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired. Defective tubes may be repaired by a Westinghouse Laser Welded Sleeve. The technical bases for Westinghouse Laser Welded Sleeve are described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections at or exceeding the plugging or repair limit of 40% of the original tube nominal wall thickness. If a tube contains a Westinghouse Laser Welded Sleeve with imperfection exceeding 40% of nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit is based on Regulatory Guide 1.121 analyses, and is described in the Westinghouse sleeving technical report mentioned above. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion.

Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Calvert Cliffs site, such as **SITE BOUNDARY** location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The **ACTION** statement permitting **POWER OPERATION** to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 μCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 increase the 2 hour thyroid dose at the **SITE BOUNDARY** by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and **STARTUP** and shutdown operation. The various categories of load cycles used for design purposes are provided in Section 4.1.1 of the UFSAR. During **STARTUP** and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the appropriate heatup and cooldown curves assures the integrity of the reactor vessel against fracture induced by combinative thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and ever more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for a peak neutron fluence to the inner surface of the reactor vessel of $\leq 2.61 \times 10^{19} N/cm^2$ (E > 1 MeV). This fluence corresponds to the Pressurized Thermal Shock Screening Criteria defined in 10 CFR 50.61 for weld 2-203 A, B, C.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Section 4.1.5 of the UFSAR. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an increase in the RT_{NDT}. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4-13 and are approved by the NRC prior to implementation in compliance with the requirements of 10 CFR Part 50, Appendix H.

The shift in the material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For a fluence of $2.61 \times 10^{19} N/cm^2$, the adjusted reference temperature (ART) value at the 1/4 T position is 241.4°F. At the 3/4 T position the ART value is 181.0°F.

These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/4 T heatup, and 3/4 T heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for

3/4.4 **REACTOR COOLANT SYSTEM**

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE.**

APPLICABILITY: **MODES** 1, 2, 3 and 4.

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a. original tube wall................................40%
   b. Westinghouse laser welded sleeve wall..............40%

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following process:

    a) Westinghouse Laser Welded Sleeving as described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per Specification 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

    1. Number and extent of tubes inspected.

    2. Location and percent of wall-thickness penetration for each indication of an imperfection.

    3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug or repair defective tubes and inspect 2S tubes in each other SG. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug or repair defective tubes. 24-hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired. Defective tubes may be repaired by a Westinghouse Laser Welded Sleeve. The technical bases for Westinghouse Laser Welded Sleeve are described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections at or exceeding the plugging or repair limit of 40% of the tube original nominal wall thickness. If a tube contains a Westinghouse Laser Welded Sleeve with imperfection exceeding 40% nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit is based on Regulatory Guide 1.121 analyses, and is described in the Westinghouse sleeving technical report mentioned above. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon-per-day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on RCS chemistry ensure that corrosion of the RCS is minimized and reduce the potential for RCS leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the RCS over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Calvert Cliffs site, such as **SITE BOUNDARY** location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The **ACTION** statement permitting **POWER OPERATION** to continue for limited time periods with the primary coolant's specific activity > 1.0 µCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 µCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 increase the 2 hour thyroid dose at the **SITE BOUNDARY** by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and **STARTUP** and shutdown operation. The various categories of load cycles used for design purposes are provided in Section 4.1.1 of the UFSAR. During **STARTUP** and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the appropriate heatup and cooldown curves assures the integrity of the reactor vessel against fracture induced by combinative thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and even more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for up to and including a fluence of $4.0x10^{19}$ n/cm^2 (E > 1 Mev) at the clad/vessel interface, which corresponds to approximately 30 Effective Full Power Years.

The reactor vessel materials have been tested to determine their initial RT_{NDTv}; the results of these tests are shown in Section 4.1.5 of the UFSAR. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4-13 and are approved by the NRC prior to implementation in compliance with the requirements of 10 CFR Part 50, Appendix H.

The shift in the material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For a fluence of $4.0x10^{19}$ n/cm^2, the adjusted reference temperature (ART) value at the 1/4 T position is 177.1°F. At the 3/4 T position the ART value is 146.8°F. These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/2 T heatup, and 3/4 T heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for the heatup event. The composite pressure-temperature limit for the

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated November 30, 1995, as supplemented on March 15, 1996, the Baltimore Gas and Electric Company (BGE/the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (Calvert Cliffs), Technical Specifications (TSs). The requested changes would allow the use of laser welded sleeves designed by Westinghouse Electric Corporation for repairing steam generator (SG) tubes. BGE also requested approval of repair techniques developed by ABB Combustion Engineering, Inc., for repairing SG tubes. The NRC staff is still reviewing that portion of BGE's request. This Safety Evaluation (SE) only relates to the laser welded sleeves designed by Westinghouse. The March 15, 1996, letter specifically identified the approved Westinghouse reports in the TSs which were previously indicated as reports currently approved by the NRC. Identifying the specific approved reports does not change the initial proposed no significant hazards consideration determination.

The technical justification supporting the Westinghouse laser welded sleeve process is given in WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feeding-Type and Westinghouse Preheater Steam Generators," Generic Sleeving Report (Proprietary) and WCAP-14469, "Specific Application of Laser Welded Sleeves for the Calvert Cliffs Power Plant Steam Generators" (Non-Proprietary).

The generic sleeving report, WCAP-13698, presents the technical bases developed to support the licensing of the laser welded sleeve installation process for use in 3/4 inch diameter tubes in feeding-type SGs such as those at Calvert Cliffs. The Combustion Engineering (CE) SGs at Calvert Cliffs are slightly different than the one modeled in WCAP-13698. The major difference is that the tubesheet is thicker in the Calvert Cliffs SG, which reduces deflections and causes the calculations in the generic report to be conservative with respect to the Calvert Cliffs Nuclear Power Plant SGs. All of the sleeve design, mechanical testing, stress corrosion testing, installation processes, and non-destructive examination procedures discussed in WCAP-13698 apply to the Calvert Cliff SGs. In WCAP-14469, BGE provided a review and evaluation of the applicability of the generic analysis to the plant specific operating parameters and to the plant design-basis parameters.

The NRC staff found that BGE's technical justification for using laser welded sleeves is within the Calvert Cliffs licensing basis. The Calvert Cliffs licensing basis, as it relates to sleeving repairs of defective SG tubes, is provided in the Attachment to this SE.

2.0 DISCUSSION

2.1 Background

Tubes in an operating pressurized water reactor (PWR) SG can be degraded by mechanisms such as primary water stress corrosion cracking (PWSCC), outer diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), pitting, or by other phenomena such as denting and vibration induced wear. Tubes that become excessively degraded reduce the integrity of the primary-to-secondary pressure boundary and must be removed from service or repaired. Degradation of SG tubes has typically been monitored using eddy current testing (ECT) techniques. Plant TSs have historically required that SG tubes be plugged at both the inlet and outlet ends of the tubes, when tubes are determined to have degraded below a calculated minimum wall thickness value (termed the "plugging limit"). However, installing plugs in a SG tube reduces the heat transfer surface area available for reactor core cooling. For this reason, design restrictions limit the total number of SG tubes which may be plugged in any one SG during the lifetime of a plant.

Alternatively, SG tubes experiencing localized degradation can be fitted with sleeves over the degraded area to reestablish the integrity of the reactor coolant pressure boundary (RCPB). The sleeves are expanded and sealed inside the tubes to provide an acceptable leak resistant load-carrying path. The reductions in heat transfer area and primary flow resulting from sleeving are slight in comparison to that resulting from tube plugging. ECT methods have been developed for monitoring any degradation in the sleeve and the underlying parent tube. However, because the sleeve becomes part of the RCPB, licensees are required to submit amendments to its TSs for review and approval before the NRC will authorize a given sleeving technique as an acceptable SG tube repair method. Licensees typically implement the TS amendments by referencing the generic or plant specific sleeving topical report(s) in the appropriate SG Limiting Condition for Operation (LCO) or Surveillance Requirement (SR).

In this case, where a licensee has amended its TS to allow for sleeving of SG tubes, the licensee may use either plugging or sleeving as a SG tube repair method. Should subsequent ECT measurements of installed sleeves indicate that a sleeved tube has degraded beyond the plugging limit for its design, the TSs would then require that the defective SG tube be plugged and removed from service. The tube and sleeve plugging limits conservatively account for the uncertainties in ECT measurements and contain additional margins for expected or postulated degradation which may occur between inspections.

2.2 Sleeve Design

Sleeves are of three basic designs: full-length tubesheet sleeve (FLTS), elevated tubesheet sleeve (ETS) and tube support sleeve (TSS). The FLTS spans from the end of the tube, at the bottom surface of the tubesheet, to a point above the secondary-side surface of the tubesheet. The ETS spans from an intermediate location within the tubesheet, up from the bottom surface of the tubesheet, to a point above the secondary-side surface outside of the tubesheet. The TSS may be installed centered approximately on a tube support intersection or in a free span section of the tube.

Tubesheet sleeves are secured by first hydraulically expanding the upper and lower portions of the sleeve. The hydraulic expansion brings the sleeve into contact with the parent tube to optimize weld performance and minimize tube deformation. A continuous circumferential laser weld is applied in the area of the hydraulically expanded region of the upper joint and stress relieved with a post weld heat treatment (PWHT). This weld structurally supports the sleeve and at the same time forms a seal.

At the lower hydraulically expanded joint, an additional mechanical roll expansion is performed on the FLTS and ETS. The lower joint is known as a hybrid expansion joint (HEJ) and it provides structural integrity under all plant conditions. Because it is a mechanical seal, the HEJ has not historically been considered leak-tight, although the test data indicate that the joint will be essentially leak-tight at operating and accident temperatures and pressures. According to BGE, the results of laboratory mock-up leak tests which were used to qualify the process are applicable to the 3/4-inch tubes used in Calvert Cliffs. The Generic Sleeving Report qualifies an optional continuous circumferential laser seal weld without PWHT that can be applied to the HEJ to provide additional leak-tightness.

2.3 Tube Lock-up (FIXITY)

BGE has discussed the consequences of the assumption that tubes might be locked to the support plate structure of the SG. An estimate was derived based on a model developed from laboratory far field stress tests of laser welded sleeves under conditions of full lock up. The model included effects of span length and stress relief temperature. A reduction of residual stress results from the current practice of hard rolling the sleeve into the tubesheet after the stress relief operation. Based on laboratory accelerated corrosion tests and experience in plants known to have some degree of lock up, resistance to stress corrosion cracking is expected for Calvert Cliffs when tube fixity is assumed. This issue was the subject of additional testing and analysis related to the use of laser welded sleeves at the Main Yankee facility recently. BGE will have the benefit of this experience and the staff recommends that the issue of tube lock-up and bowing be examined for plant specific concerns at Calvert Cliffs.

2.4 Service Experience

About 28,000 laser welded sleeves have been installed in foreign and domestic plants since 1988 including sleeves installed in both 7/8 inch and 3/4 inch tubes. It reports that at one plant there was no indication of corrosion, either at the laser weld or the hydraulic expansion of the upper joint, or at the hydraulically expanded and rolled joint in the tubesheet, after five cycles of operation.

The sleeves are manufactured from thermally treated, ASME SB-163, Alloy 690. This material, also known as Alloy 690 TT (thermally treated), has been demonstrated to be highly resistant to intergranular stress corrosion cracking (IGSCC) under SG conditions. The resistance of the laser welded sleeve joint to in-service cracking depends primarily on the resistance of the parent Alloy 600 tubing to IGSCC. As mentioned previously, stresses in the tubing, either service operating stresses or residual stresses, can potentially cause cracking. Two sources of residual stresses are related to hydraulic expansion during sleeve placement and to stresses introduced as a result of welding.

A testing program was conducted under conditions which accelerate corrosion in SG materials to simulate long term SG service. Each test contained a rolled tube transition, which served as a control sample. The stress levels in the control sample were representative of the residual stress conditions in the hard rolled transitions found at the top of the tubesheets in operating SGs and are considered a bounding stress level.

The accelerated corrosion testing was performed in high temperature-high pressure autoclaves. A doped steam environment was utilized to accelerate crack initiation and propagation. Results of the accelerated corrosion tests indicate that laser welded sleeve joints with post-weld heat treatment have a corrosion resistance greater than that of the as-welded joint.

2.5 Licensee Commitments

In its November 30, 1995, submittals, BGE has committed to the use of a post weld stress relief heat treatment for the laser welded sleeving method in accordance with staff positions.

BGE has also committed to use qualified processes for periodic inservice inspection and to evaluate improved inspection techniques as they are developed and qualified for use.

By letters dated June 27, 1995, and October 5, 1995, BGE responded to Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," in which they made the commitment to perform their SG inspections in conformance with EPRI recommendations.

With respect to the staff position regarding primary-to-secondary leakage limits to account for the installation of the sleeves into the SGs, BGE has already implemented a change to TS 3.4.6.2 adopting a 100 gpd per SG leakage limit (TS Change dated July 13, 1992).

3.0 EVALUATION

3.1 Sleeve Design and Analysis and Testing

The laser welded sleeves (both tubesheet and tube support plate sleeves) have been analyzed to ensure they will maintain SG tube structural and leak integrity during all plant conditions. The sleeve joint designs have been qualified through laboratory testing and analysis. Analytical verification has been performed using design and operating transient parameters which have been determined to apply to Calvert Cliffs conditions.

The function of the sleeve is to restore the integrity of the RCPB in the region between the sleeve joints to a level which is consistent with the original tube. The sleeve has been designed according to Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Fatigue and stress analyses of sleeved tube assemblies have been completed in accordance with the requirements of the ASME Code, Section III. The analyses include a primary stress intensity evaluation, primary plus secondary stress intensity range evaluation, and a fatigue evaluation for mechanical and thermal conditions. WCAP-14469 compared the current set of transient and operating parameters for Calvert Cliffs to those used in the generic analysis used in WCAP-13698 to confirm that the generic analysis provides a valid analysis for Calvert Cliffs. It was concluded that all applicable code and regulatory requirements were met.

3.2 Sleeve Plugging Limits

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide (RG) 1.121, the ASME Code Section III allowable stress values, and the pressure stress equation in NB-3324.1 of Section III of the ASME Code. According to RG 1.121 criteria, an allowance for non-destructive evaluation (NDE) uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% through-wall per cycle have been assumed for the purpose of determining the sleeve plugging limit. The sleeve plugging limit, which was calculated based on the most limiting of normal, upset or faulted conditions for Calvert Cliffs was determined to be 40% of the sleeve nominal wall thickness. Removal of tubes and/or sleeves from service when degradation reaches the plugging limit provides assurance that the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

3.3 Leakage

While a laser weld should be inherently leak-tight, in practice, the lower joint of a tubesheet sleeve may be installed with or without a seal weld; therefore, the leakage characteristics must be considered. BGE has analyzed the effects of an abnormal tubesheet sleeve lower joint seal. The analysis shows that even under extreme postulated conditions, it will have satisfactory leakage integrity.

3.4 Non-Destructive Examination

The welding parameters are computer controlled. The essential variables, in accordance with the ASME Code, are monitored and documented to produce repeatability of the weld process. In addition, the NDE of the laser welded sleeves utilizes two techniques. Ultrasonic testing (UT) is performed after welding to confirm that the laser welds are consistent with critical process dimensions and are of acceptable weld quality. BGE has presented data on a UT system that demonstrates post weld examinations of the sleeve-tube assembly will be adequate. Standards, which simulated a leakage path along the weld metal, were used in the qualification of the UT technique. The results of the qualification tests demonstrate that the system can confirm that there is a metallurgical bond between the sleeve and the tube and determine if a leak path exists across the weld.

Eddy current testing (ET) is used to establish baseline inspection data for every installed sleeve/tube to be utilized during subsequent inservice inspections. Furthermore, in accordance with the criteria in RG 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," BGE has presented data, which demonstrates that structural integrity of sleeved tubes can be monitored by performing periodic eddy current examinations. However, for future inservice inspections, BGE has stated in its November 30, 1995, submittal, that it will evaluate improved inspection techniques as they are developed and qualified for use. A number of proprietary advanced inspection techniques that are currently under development were described.

3.5 Sleeving of Previously Plugged Tubes

In the event that previously plugged tubes are unplugged and returned to service by using the sleeving process, the plant TSs would require that the sleeving requirements be applied to the tubes designated for sleeving. This includes provisions to ensure that the new sleeve joints are located a minimum acceptable distance apart from the degraded tube area. Following installation, a new "baseline" inspection of the tube and sleeve would then be required for any sleeved tube placed back in service.

4.0 SUMMARY

Based on the preceding analysis, the NRC staff concludes that the repair of SG tubes using laser welded sleeves designed by Westinghouse is acceptable, as supplemented by additional BGE commitments, as discussed above, to 1) using enhanced and improved eddy current testing inspection techniques as they are developed and verified for use, 2) performing post-weld heat treatment of installed laser welded sleeves, 3) maintaining requirements to provide for appropriate inservice inspection for any steam generator tubes containing sleeves and 4) continuing the existing TS requirements for a primary-to-secondary leakage limit of 100 gpd maximum per steam generator. Thus, the proposed changes to TSs 3/4.4.5, "STEAM GENERATORS," and their supporting Bases is acceptable for the laser welded sleeves designed by Westinghouse.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 176). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: H. Conrad

Date: March 22, 1996

DISCUSSION OF LICENSING BASIS

1. 10 CFR 50.55a

10 CFR 50.55a, "Codes and Standards," requires that components, which are a part of the reactor coolant pressure boundary to be built to the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) 10 CFR 50.55a also requires that throughout the service life of the plant, that licensees meet the inservice inspection requirements of the ASME Code Section XI for ASME Code Class 1, 2 or 3 components.

Section 5.2.1.1 of the Standard Review Plan, entitled "Compliance with the Codes and Standards Rule, 10 CFR 50.55a," provides an outline of the standards used for evaluation by the NRC staff. Any modification, repair or replacement of these components must also meet the requirements of the ASME Code to assure that the basis on which the unit was originally evaluated is unchanged.

2. ASME Code Requirements

The design of the sleeves is predicated on the requirements of the ASME Code Section III, Subarticles NB-3200, "Analysis" and NB-3300, "Wall Thickness." The ASME Code provides criteria for evaluation of the stress levels in the tubes for design, normal operating, and postulated accident conditions. The margin of safety is provided, in part, by the inherent safety factors in the criteria and requirements of the ASME Code.

Section IX of the ASME Code, Subsection QW, and Section III, including Code Case N-395, define the applicable essential variables for the welding procedure specification and welding procedure qualification test.

Section XI, IWB-4334 of the ASME Code defines the extent of examination requirements for installation of laser welded sleeves.

3. Regulatory Guide 1.121

Regulatory Guide (RG) 1.121, issued for comment, entitled "Bases for Plugging Degraded PWR Steam Generator Tubes," addresses tubes with defects. The criteria of RG 1.121 are extended to the laser welded sleeve in order to determine the level of degradation, which will require removal of the sleeve from service by plugging. ASME Code allowable strength values were used for this evaluation. By utilizing the requirements for sleeve design according to the ASME Code and RG 1.121 to define acceptance criteria, the sleeve meets the requirements of General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

4. Regulatory Guide 1.83

Regulatory Guide 1.83, "Inservice Inspection of Pressurizer Water Reactor Steam Generator Tubes" (and the Calvert Cliffs Technical Specifications) is used as the basis to determine the inservice inspection requirements for the sleeve.

5. 10 CFR Part 100

Total plant allowable primary-to-secondary leakage rates, derived from the requirements of 10 CFR Part 100, are determined on a plant specific basis. Offsite doses during either a main steam line break or tube rupture event are not to exceed a small fraction of the 10 CFR Part 100 limits per the Bases to the Calvert Cliffs Technical Specifications.

March 22, 1996

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M94205) AND UNIT NO. 2 (TAC NO. M94206)

Dear Mr. Cruse:

The Commission has issued the enclosed Amendment No. 213 to Facility Operating License No. DPR-53 and Amendment No. 190 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 30, 1995, as supplemented on March 15, 1996.

The amendments allow the installation of tube sleeves as an alternative to plugging for repairing steam generator (SG) tubes using repair techniques developed by Westinghouse Electric Corporation. In your November 30, 1995, letter, you also requested approval of repair techniques developed by ABB Combustion Engineering, Inc., for repairing SG tubes. We are still reviewing that portion of your request.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Daniel G. McDonald, Jr., Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosures:
1. Amendment No. 213 to DPR-53
2. Amendment No. 190 to DPR-69
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: See attached page *See Previous Concurrence

DOCUMENT NAME: G:\CC1-2\CC94205.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	LA:PDI-1		PM:PDI-1	E	BC:EMCB*		OGC*		D:PDI-1	
NAME	SLittle		DMcDonald:smm		JStrosnider		EHoller		SShankman	
DATE	03/22/96		03/22/96		03/14/96		03/14/96		03/22/96	

OFFICIAL RECORD COPY

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M94205) AND UNIT NO. 2 (TAC NO. M94206)

Dear Mr. Cruse:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-53 and Amendment No. to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 30, 1995.

The amendments allow the installation of tube sleeves as an alternative to plugging for repairing steam generator (SG) tubes using repair techniques developed by Westinghouse Electric Corporation. In your November 30, 1995, letter, you also requested approval of repair techniques developed by ABB Combustion Engineering, Inc., for repairing SG tubes. We are still reviewing that portion of your request.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly <u>Federal Register</u> notice.

Sincerely,

Daniel G. McDonald, Jr., Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosures: 1. Amendment No. to DPR-53
2. Amendment No. to DPR-69
3. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION</u>: See attached page

DOCUMENT NAME: G:\CC1-2\CC94205.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	LA:PDI-1	PM:PDI-1	BC:EMCB	OGC	D:PDI-1
NAME	SLittle	DMcDonald:smm	JStrosnider	EHolcox	LMarsh
DATE	02/20/96	02/ /96	02/14/96	02/14/96	02/ /96

OFFICIAL RECORD COPY
OPH w/comments