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

Septen 1, 1999

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR USING LEAK LIMITING ALLOY 800 SLEEVES (TAC NOS. MA4278 AND MA4279)

Dear Mr. Cruse:

The Commission has issued the enclosed Amendment No. 231 to Facility Operating License No. DPR-53 and Amendment No. 207 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, 1998, as supplemented May 25, 1999.

The amendments revise the appropriate TSs to permit the use of leak-limiting Alloy 800 repair sleeves developed by AAB - Combustion Engineering (ABB-CE) to be used at Calvert Cliffs.

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</u> <u>Register</u> notice.

Sincerely,

ORIGINAL SIGNED BY:

Alexander W. Dromerick, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures: 1. Amendment No. 231 to DPR-53
2. Amendment No. 207 to DPR-69
3. Safety Evaluation

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Baltimore Gas and Electric Company
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September 1, 1999

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

September 1, 1999

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: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR USING LEAK LIMITING ALLOY 800 SLEEVES (TAC NOS. MA4278 AND MA4279)

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Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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UNITED STATES
NUCLEAR REGULATOR'. 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. 231
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, 1999, as supplemented May 25, 1999, 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. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 231 , 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 during the spring 2000 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

M Bajwa

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 1, 1999

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. 207
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, 1998, as supplemented May 25, 1999, 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. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 207 , 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 during the spring 2001 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 1, 1999

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 231 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 207 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
5.0-18	5.0-18
5.0-19	5.0-19

5.5 Programs and Manuals

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:

i. original tube wall	40%
ii. Westinghouse laser welded sleeve wall	40%
iii. ABB-Combustion Engineering leak tight sleeve wall	28%
iv. ABB-Combustion Engineering Alloy 800 leak-limiting sleeve wall	35%

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 5.5.9.c.3 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.

5.5 Programs and Manuals

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

    i. 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.

    ii. ABB-Combusting Engineering Leak Tight Sleeving as described in the proprietary ABB-Combustion Engineering Report CEN-630-P, Revision 01, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," August 1996. A post-weld heat treatment during installation will be performed.

    iii. ABB-Combustion Engineering Alloy 800 leak-limiting sleeving as described in the Proprietary ABB Combustion Engineering Report CEN-633-P, Revision 03-P, "Steam Generator Tube Repair For Combustion Engineering Designed Plants with 3/4-.048 Inch Wall Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves, " October 1998.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 5.5.9.d.9 is required prior to returning previously plugged tubes to service.

e. Surveillance Completion - The Steam Generator Tube Surveillance Program is met after completing the corresponding actions (plug or repair all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 5.5.9-2 and 5.5.9-3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 231 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 207 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, 1998, as supplemented May 25, 1999, Baltimore Gas and Electric Company (the licensee) submitted for staff review a request for an amendment to the Calvert Cliffs Units 1 and 2 Technical Specifications (TS) to allow the installation of Alloy 800 leak-limiting sleeves in defective steam generator tubes as a tube repair method.

ABB-Combustion Engineering (ABB-CE) developed the proposed leak-limiting repair that uses sleeves made of Alloy 800 material. The sleeve is inserted into and expanded inside of the degraded tube to form structural joints with the tube. The joints are installed to be leak tight; however, by design, a limited leakage is allowed. The sleeve design, installation, analysis, and qualification are documented in the ABB-CE Report, "Steam Generator Tube Repair for Combustion Engineering Designed Plants with 3/4" - .048" Wall Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," CEN-633-P, Revision 03, dated October 1998 (proprietary). Two sleeve designs are available for tube repair. An expansion transition zone (ETZ) sleeve is designed to repair the degraded region of a tube in the vicinity of the top of the tubesheet. An egg crate support (ECS) sleeve is designed to repair a degraded tube region that spans an egg crate support elevation or be used on a degraded freespan section of tube. The leak limiting sleeves have been installed in foreign nuclear plants but not in any domestic nuclear plants.

Each of Calvert Cliffs units has two Combustion Engineering model 67 steam generators. The tube material is mill annealed Alloy 600 and the tubes are supported by eggcrate and drilled support plate configurations. The tube has an outside diameter of 0.75 inches and a thickness of 0.048 inches. The primary system pressure is 2250 psi and secondary system pressure is 850 psi. The main steam or feedwater line break pressure is 2560 psi. The May 25, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The sleeve repairs a part of existing reactor coolant pressure boundary and as such, it should be qualified for service in accordance with the specifications in Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which refers back to Section III of the ASME Code. Section III of the ASME Code contains the design requirements for the original steam generator tubes. Because the sleeve repairs the degraded portion of the pressure boundary, specifications in Section III apply. The sleeve should be analyzed by appropriate ASME Code equations considering design, operating, and accident loading conditions. The resulting stresses should satisfy corresponding ASME Code allowables. The sleeve wall thickness needs to satisfy the minimum wall thickness requirement of the ASME Code.

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," provides guidance for determining appropriate plugging limits. RG 1.121 recommends criteria for determining minimum wall thickness, beyond which the degraded tube should be plugged (i.e., plugging limits). RG 1.121 also recommends that the margin of safety against tube rupture under normal operating conditions should not be less than 3 at any tube location where defects have been detected. The margin of safety against tube failure under postulated accidents, such as loss-of-coolant accident (LOCA), steam line break, or feedwater line break concurrent with the safe shutdown earthquake, should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Code. Because the sleeve is a part of tube, it should follow RG 1.121.

The NRC staff reviews the following areas of the sleeve repair method: design, installation, material selection, qualification tests, structural and leakage analyses, and nondestructive examination. In terms of sleeve implementation in the Technical Specifications, the staff reviews the proposed inservice inspection program, sleeve plugging limits, operating primary-to-secondary leakage limit, and corresponding proposed TS changes. These topics are discussed below:

The Alloy 800 sleeve is designed to fit inside of the parent tube. The lengths of the ETZ and ECS sleeves are sized to encompass the degraded regions of the tube. The ETZ sleeve has several hydraulic expansion joints in the upper region of the sleeve and a hard roll joint in the lower region of the sleeve that is located at about the middle elevation of the tubesheet. The ECS sleeve has several hydraulic expansion joints in the upper and lower regions of the sleeve. Each of the expansion joints is about 0.3-inch long and spaced about 0.75-inch apart.

Prior to sleeve installation, the inside surface of the candidate tubes is cleaned and a visual inspection (VT) is performed to verify the quality of the contact surface. After VT, the sleeves are inserted into the cleaned tubes and positioned at the desired location. An expansion device is inserted into the sleeve to expand the sleeve to make hydraulic expansion joints with the tube. The expansion device is controlled and monitored to ensure consistent diametral expansion. The hydraulic expansion joints support the required structural and leakage integrity while minimizing residual stresses in the parent tube. The hard roll in the lower end of ETZ sleeve is performed by a roll expander. The torque of the roll expander is also monitored and controlled. After the installation, all sleeve-tube joints will undergo an initial acceptance and baseline inspection by the eddy current technique. ABB-CE indicated that when the cleaning process is demonstrated to assure cleaning efficiency, a sampling VT of cleaned tubes may be used in the future.

The sleeve material, Alloy 800, is a nickel-iron-chromium alloy. ABB-CE selected Alloy 800 for its favorable mechanical properties and corrosion resistance in the PWR water chemistry. It is procured to the requirements of the ASME Code, Section II, Part B, SB-163, NiFeCr Alloy, Unified Numbering System (UNS) N08800, and Section III, Subsection NB-2000. Alloy 800 is incorporated in ASME Code Case N-20 and is approved for use by RG 1.85, "Materials Code Case Acceptability ASME Section III, Division 1," Revision 24, dated July 1986. ABB-CE requires additional restriction on contents of various chemical elements in Alloy 800.
To qualify the leak limiting sleeve for the field service, ABB-CE performed mechanical load tests, leakage tests, and corrosion tests. The mechanical tests consist of axial load tests, pressure tests, collapse tests, and load cycling tests. The tests were performed on sleeve-tube mock-ups that were constructed to the same dimensions as the installed sleeves in the field.

ABB-CE performed the axial load tests to determine the sleeve-tube integrity under differential thermal expansion of an Alloy 800 sleeve and Alloy 600 tube. The test loads cover the full range loading under startup, transient, normal power, shutdown, and accident conditions. The axial load tests showed that the Alloy 800 sleeve can support differential thermal conditions and accident loads even if the parent tube is severed.

ABB-CE performed the pressure tests to determine the sleeve-tube integrity under primary-to-secondary pressure differentials during normal operating, transient and postulated accident conditions. The pressure tests showed that the sleeve-tube joints maintain a margin of 3 with respect to the normal operating differential pressure load.

ABB-CE performed the collapse tests to show that the sleeve would not collapse if water is trapped in the annulus region between the inside surface of the parent tube and the outside surface of the sleeve. The trapped water may be pressurized during operation and potentially cause the sleeve to collapse. The collapse tests showed that the sleeve would not collapse under the maximum secondary side pressure.

ABB-CE performed the load cycling tests to show that the sleeve-tube structural and leakage integrity will be maintained under cyclical differential thermal expansion and internal pressure in normal operating and transient conditions. The load cycling tests included fatigue tests, thermal cycling tests, and mechanical load cycling tests. The load applied in the cycling tests was greater than three times the maximum differential pressure. These tests showed that under various temperatures, the sleeve-tube joint is not degraded by cyclic loads. The cycling tests confirm that slip during the initial heat-up is small, and the sleeve repositions itself inside of the parent tube to accommodate the thermal expansion without subsequent slip. As a part of the load cycling tests, the specimens were also tested for leakage integrity. The leak tests showed that the seal in the hydraulically expanded joints improved after load cycling. The leakage measured after load cycling was consistently lower than the leakage before the load cycling.

ABB-CE performed leak rate tests on sleeve-tube assembly at room and operating temperatures with primary-to-secondary pressure differential under normal operating and main steam line break conditions. The test results showed that the leakage from a single sleeve is extremely small relative to the primary-to-secondary leakage limits of 100 gallon per day per steam generator in the current Calvert Cliffs Units 1 and 2 TS. It would take thousands of leaking sleeves to reach the leakage limit.

In general, sleeve installation increases the residual stresses in the parent tube which, in turn, may increase susceptibility to stress corrosion cracking. The Alloy 800 sleeve is designed to impart minimal residual stresses in the parent tube to avoid potential corrosion in the hydraulic expansion joints. ABB-CE performed various corrosion tests and assessments of Alloy 800 sleeves. ABB-CE sponsored corrosion tests for the Alloy 800 sleeves installed in European nuclear power plants. ABB-CE also performed an accelerated corrosion tests with full length sleeved-tube mock-ups. Sleeve-tube assemblies were pressurized with highly concentrated sodium hydroxide. In all corrosion tests, Alloy 800 sleeve did not develop any cracking in either the primary and secondary side tests. The parent tube did develop cracks. The Alloy 800 sleeve has demonstrated a higher corrosion resistence than the Alloy 600 parent tube.

ABB-CE stated that besides Alloy 800 sleeves, Alloy 800 tubing has been used in pressurized-water reactor (PWR) conditions in foreign power plants without experiencing primary or secondary side stress corrosion cracking. This is based on experience of over two hundred thousand tubes in service for up to 19 years.

Besides testing, ABB-CE performed structural analyses in accordance with 10 CFR Part 50 Appendix B and followed the methods in Section III of the ASME Code. The structural analyses included pressure, relative displacement, fatigue, axial load, seismic, and thermal radial differential loads under normal and accident loading conditions. To determine the axial loading on the sleeve, ABB-CE assumed two bounding tube configurations: 1) the tube is intact, and 2) the tube is severed at the flaw location. The two bounding tube support configurations are 1) the tube is free to move past the supports and 2) the tube is locked in the first support and is prevented from axial motion. The analyses showed that stresses and fatigue factor in the worst sleeve-tube configuration satisfy the allowables in Section III of the ASME Code.

The structural analysis also included calculations for minimum required sleeve thickness and sleeve plugging limit. The minimum required sleeve thickness is calculated based on ASME Code Section III, Paragraph NB-3324.1. The actual sleeve wall thickness is greater than the required thickness and is, therefore, acceptable.

With respect to non-destructive examination, ABB-CE specified that the plus point coil will be used in the inservice inspection of Alloy 800 sleeved tubes. The eddy current technique has been qualified and used to inspect Alloy 800 sleeves in international nuclear plants. The eddy current inspection method has been qualified in accordance with Appendix H of Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines, Revision 5, dated September 1997. The qualification test program was performed in accordance with 10 CFR Part 50, Appendix B. ABB-CE recommends plug on detection for those tubes that contains indications in sleeve-tube pressure boundary.

3.0 EVALUATION

The NRC staff has reviewed the licensee's structural analyses of the Alloy 800 sleeve and found the stresses in the sleeve satisfy the allowables in Section III of the ASME Code.

With respect to leakage integrity, the licensee has demonstrated leak resistance of the sleeve through laboratory tests. Bounding calculations and laboratory tests have verified that, should leakage develop in the sleeved tubes, it would not exceed 1 gallon per minute (gpm) and, thus, the 10 CFR Part 100 guidelines for radiological release would not be affected, even under the most severe postulated conditions. In addition, the licensee has limited primary-to-secondary leakage to 100 gallons per day through any one steam generator in the Calvert Cliffs TS. The staff finds that the leakage tests have demonstrated the adequacy of limiting leakage capacity of the sleeve.

The time for the initiation of corrosion in sleeve-tube assemblies is difficult to quantify accurately. Although vendors traditionally conduct accelerated corrosion tests of sleeve-tube assemblies to predict service life, the staff finds this method unreliable for deterministic predictions. However, the staff does consider that the corrosion tests give a viable indicator of potential performance. With regard to ABB-CE's corrosion assessment, it is evident that Alloy 800 material has had good operating experience in foreign nuclear plants. However, there has not been any Alloy 800 sleeves used in domestic nuclear plants. Presently, the staff can only assume a limited life expectancy for Alloy 800 sleeves. Considering the uncertainties in sleeve life expectancy, the licensee will inspect a sample of sleeves at each refueling outage in accordance with Table 5.5.9-3 of the Calvert Cliffs TSs to ensure that any degradation in the sleeve assembly is detected and addressed early.

For inservice inspection of sleeved tubes, the licensee has implemented an initial, minimum inspection of 20 percent sleeves at each refueling outage as required in the current Calvert Cliffs TS. In addition to the 20-percent initial sample, the results from inspections would be classified and, depending on the classification of sleeve degradation, additional sleeves may be inspected. This inspection sampling is consistent with the staff position and with the current industry guidance for steam generator sleeve examinations as specified in EPRI report, "Steam Generator Examination Guideline," Revision 5.

The sleeve plugging limit is defined in the TS as the imperfection depth in the sleeve at or beyond which the sleeved tube shall be removed from service. RG 1.121 provides guidance on calculating sleeve plugging limit on the basis of structural consideration. In addition to structural consideration, RG 1.121 also suggests that an allowance for inspection uncertainty and postulated growth of degradation be accounted for in the sleeve degradation calculations. The licensee assumes a 10 percent allowance for inspection uncertainty and a 10 percent allowance for degradation growth per cycle. The licensee calculated a minimum acceptable wall thickness of 54.67 percent through wall due to structural consideration. After deducting a total allowance of 20 percent through wall, the licensee specified a sleeve plugging limit of 35 percent of sleeve wall thickness. On the basis of the calculation presented, the staff finds that a plugging limit of 35 percent of sleeve wall thickness is acceptable.

Under severe accident conditions in which primary system temperature may reach to 1200 to 1500 degree F, the material properties of Alloy 800 behave not significantly different from that of Alloy 600 at the severe accident temperature. The staff does not believe additional risk affects structural and leakage integrity of the pressure boundary from Alloy 800 sleeve as compared to Alloy 600 parent tubing.

The licensee has proposed the following changes to TS 5.5.9, "Steam Generator Tube Surveillance Program:"

1. The sleeve plugging limit is included in TS 5.5.9.d.7 as follows:

    iv. ABB-Combustion Engineering Alloy 800 leak-limiting sleeve wall — 35%

2. CEN-633 is referenced in TS 5.5.9.d.10 as follows:

   "iii. ABB-Combustion Engineering Alloy 800 leak-limiting sleeving as described in the proprietary ABB Combustion Engineering Report CEN-633-P, Revision 03-P, "Steam Generator Tube Repair for Combustion Engineering Designed Plants with 3/4-.048 Inch Wall Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," October 1998."

The staff finds that both of these changes are acceptable because they satisfy the technical basis of licensee's proposed changes.

The licensee has investigated the structural and leakage integrity of the sleeve design analytically and experimentally. The licensee performed structural analyses and tests for a variety of thermal and pressure loadings that enveloped plant-specific design, operating, transient and accident loads. The analyses, testing, and foreign plant operating experience demonstrate that the sleeved tube assembly is capable of restoring steam generator tube integrity. The staff finds that the licensee has demonstrated the acceptability of the sleeve repair in accordance with Appendix B to 10 CFR Part 50, RG 1.121, and the ASME Code. The staff concludes that the licensee may incorporate the proposed changes into the Calvert Cliffs Units 1 and 2 TS.

4.0 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.

5.0 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. 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 (64 FR 2244). 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.

6.0 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: J. Tsao

Date: September 1, 1999

DATED: September 1, 1999

AMENDMENT NO. 231 TO FACILITY OPERATING LICENSE NO. DPR-53-CALVERT CLIFFS UNIT 1
AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-69-CALVERT CLIFFS UNIT 2

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