5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety.

5.1.2 The Control Room Supervisor (CRS) shall be responsible for the control room command function. During any absence of the CRS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the CRS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Final Safety Analysis Report (UFSAR) or Quality Assurance Policy;

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. The Vice President-Nuclear Energy shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A total of three non-licensed operators shall be assigned to the Units 1 and 2 shift crews.

b. Those licensed operators counted toward minimum shift crew composition required by 10 CFR 50.54(m)(2)(i) shall be licensed for both units.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i), 5.2.2.a, and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. A radiation protection technician shall be onsite when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related-functions (e.g., licensed Senior Reactor Operators, licensed Reactor Operators, health physicists, auxiliary operators, and key maintenance personnel). The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized by the Plant General Manager or the Plant General Manager's designee, in accordance with approved administrative procedures and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

5.2 Organization

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

f. The operations manager shall hold or have held an SRO license at Calvert Cliffs. The General Supervisor-Nuclear Plant Operations shall hold an SRO license.

g. One Shift Technical Advisor (STA) shall be assigned to the shift crew when either unit is in MODE 1, 2, 3, or 4, and shall be filled as follows:

1. By the Shift Supervisor (SS) or an on-shift SRO license holder, provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift; or

2. By an individual with a Bachelors Degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant transient and accidents; or

3. By an SRO license holder previously approved by the Nuclear Regulatory Commission as an exception to the minimum STA education requirements of Specification 5.2.2.g.2, provided the following conditions are met:

   i. With both units in MODE 1, 2, 3, or 4, the STA shall be an SRO license holder in addition to the two SRO license holders required,

   ii. With one unit in MODE 1, 2, 3, or 4, and the other unit in MODE 5 or 6, the STA shall be an SRO license holder other than the SS, and

5.2 Organization

iii. With one unit in MODE 1, 2, 3, or 4, and the other unit defueled, the STA shall be an SRO license holder in addition to the one SRO license holder required.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, for comparable positions, except for the Radiation Protection Manager, who shall meet or exceed the requirements of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor, who shall meet the requirements of Specification 5.2.2.g.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual

a. The Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   i. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   ii. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and 10 CFR Part 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive

5.5 Programs and Manuals

Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, and Chemical and Volume Control. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months. The provisions of SR 3.0.2 are applicable.

5.5.3 Post-Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation, including surveillance tests and setpoint determination, in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR Part 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas to be limited:

   1. During any calendar quarter: Less than or equal to 3 mrems to the total body, and to less than or equal to 10 mrems to any organ; and

   2. During any calendar year: Less than or equal to 6 mrems to the total body, and to less than or equal to 20 mrems to any organ;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year, in accordance with the methodology and

5.5 Programs and Manuals

parameters in the ODCM, at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the Liquid Radwaste Treatment System to ensure that appropriate portions of this system are used to reduce releases of radioactivity when the projected doses to unrestricted areas exceeds 0.36 mrem to the total body, or 1.20 mrem to any organ in a 92-day period;

g. Limitations on the functional capability and use of the Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the calculated doses to unrestricted areas exceeds 1.20 mrad for gamma radiation, and 2.40 mrad for beta radiation in a 92-day period;

h. Limitations on the functional capability and use of the Ventilation Exhaust Treatment System to ensure that appropriate portions of this system are used to reduce releases of radioactivity when the calculated doses due to gaseous releases to unrestricted areas exceeds 1.8 mrem to any organ in a 92-day period;

i. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary, to be limited:

   1. For noble gases: Less than or equal to 500 mrems/yr to the total body, and less than or equal to 3000 mrems/yr to the skin; and

   2. For Iodine-131 and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ;

5.5 Programs and Manuals

j. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary, to be limited to:

   1. During any calendar quarter: Less than or equal to 10 mrads for gamma radiation, and less than or equal to 20 mrads for beta radiation; and

   2. During any calendar year: Less than or equal to 20 mrads for gamma radiation, and less than or equal to 40 mrads for beta radiation;

k. Limitations on the annual and quarterly doses to a member of the public from Iodine-131 and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released from each unit to areas beyond the site boundary, to be limited:

   1. During any calendar quarter: Less than or equal to 15 mrems to any organ:

   2. During any calendar year: Less than or equal to 30 mrems to any organ; and

   3. Less than 0.1% of the limits of 5.5.4.k(1) and (2) as a result of burning-contaminated oil; and

l. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity, and to radiation from uranium fuel cycle sources to be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5 Programs and Manuals

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 4.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operation. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

5.5 Programs and Manuals

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.9 Steam Generator Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program except as specified for individual requirements. This program provides controls for the inservice inspection of steam generator tubes to ensure that structural

5.5 Programs and Manuals

integrity of this portion of the Reactor Coolant System is maintained. The program shall contain the requirements listed below.

a. Steam Generator Sample Selection and Inspection - The minimum number of steam generators to be inspected shall be determined as specified in Table 5.5.9-1.

b. 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 Tables 5.5.9-2 and 5.5.9-3. The inservice inspection of steam generator tubes shall be performed at the Frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. When applying the exceptions of 5.5.9.b.1 through 5.5.9.b.3, 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:

1. 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.

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

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

   ii. Tubes in those areas where experience has indicated potential problems.

5.5 Programs and Manuals

3. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the three categories specified below. In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the percentage calculations.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes, or more than 1% of the inspected tubes are defective.

c. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following Frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months, but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If at least 20 percent of the

5.5 Programs and Manuals

tubes were inspected and the results were in the C-1 Category, or if at least 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in the C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections following service under all-volatile treatment conditions, not including the preservice inspection result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the inservice inspection results of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1; the interval may then be extended to a maximum of once per 30 or 40 months, as applicable.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and

5.5 Programs and Manuals

5.5.9-3 during the shutdown subsequent to any of the following conditions:

i. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13;

ii. A seismic occurrence greater than the Operating Basis Earthquake;

iii. A loss-of-coolant accident requiring actuation of the engineered safeguards; or

iv. A main steam line or feedwater line break.

4. The provisions of Specification SR 3.0.2 do not apply for extending the Frequency for performing inservice inspections as stated in Specifications 5.5.9.c.1 and 5.5.9.c.2.

d. Acceptance Criteria - 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 dimension, 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.

5.5 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.	ABB-Combustion Engineering leak tight sleeve wall (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)	28%
iii.	ABB-Combustion Engineering Alloy 800 leak-limiting sleeve wall(Unit 2 through Cycle 14 only. Not applicable for Unit 1.)	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. 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. (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)

    ii. 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. (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)

    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.

5.5 Programs and Manuals

Table 5.5.9-1
Minimum Number of Steam Generators to be
Inspected During Inservice Inspection

Preservice Inspection	No			Yes		
No. Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3 Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

5.5 Programs and Manuals

Table 5.5.9-2
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 steam generator	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 steam generators	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this steam generator.	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 steam generator, plug or repair defective tubes and inspect 2S tubes in each other steam generator. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	All other steam generators are C-1	None	N/A	N/A
			Same steam generators C-2 but no additional steam generator are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional steam generator is C-3	Inspect all tubes in each steam generator and plug or repair defective tubes. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	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.

5.5 Programs and Manuals

Table 5.5.9-3
Steam Generator Repaired Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A Minimum of 20% of repaired tubes[1][2]	C-1	None	N/A	N/A
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this SG.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this SG, plug defective tubes and inspect 20% of the repaired tubes in the other SG. 24-Hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4	Other SG is C-1	None
			Other SG is C-2	Perform action for C-2 result of first sample
			Other SG is C-3	Inspect all repaired tubes in each SG and plug defective tubes. 24-hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from either SG based on outage plans.

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which are required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of engineered safety feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.11.a and 5.5.11.b shall be performed once per 18 months for ventilation systems other than the Iodine Removal System (IRS) and 24 months for the IRS; after each complete or partial replacement of the high efficiency particulate air (HEPA) filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or

5.5 Programs and Manuals

chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.11.c shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS; after 720 hours of system operation; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.11.d shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System (CREVS)	2,000 cfm
Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)	3,000 cfm
Penetration Room Exhaust Ventilation System (PREVS)	2,000 cfm
Spent Fuel Pool Exhaust Ventilation System (SFPEVS)	32,000 cfm
IRS	20,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52,

5.5 Programs and Manuals

Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREVS	2,000 cfm
ECCS PREFS	3,000 cfm
PREVS	2,000 cfm
SFP Ventilation System	32,000 cfm
IRS	20,000 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetrations	RH
CREVS	5%	70%
ECCS PREFS	50%	95%
PREVS	35%	95%
SFP Ventilation System	15%	95%
IRS	35%	95%

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52,

5.5 Programs and Manuals

Revision 2, and ANSI N510-1975 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREVS	4 inwg	2,000 cfm
ECCS PREFS	4 inwg	3,000 cfm
PREVS	6 inwg	2,000 cfm
SFP Ventilation System	4 inwg	32,000 cfm
IRS	6 inwg	20,000 cfm

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System and the quantity of radioactivity contained in gas storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in the ODCM.

The program shall include:

a. The limits for concentrations of oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 58,500 curies noble gases (considered as Xe-133).

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance Frequencies.

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A Diesel Fuel Oil Testing Program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An American Petroleum Institute gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. Water and sediment ≤ 0.05%.

b. Within 31 days following addition of new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil, when determined by gravimetric analysis based on ASTM D2276-1989, is ≤ 10 mg/l when tested every 92 days.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Frequencies.

5.5.14 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

5.5 Programs and Manuals

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the Technical Specifications incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into Limiting Condition for Operation (LCO) 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

5.5 Programs and Manuals

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:

a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15,

5.5 Programs and Manuals

1992 Type A test shall be performed no later than June 14, 2007.

b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 psig. The containment design pressure is 50 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.20 percent of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.0002\ L_a$ when pressurized to ≥ 15 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----------------------------NOTE-----------------------------
A single submittal may be made for both units, but shall not include the occupational radiation exposure from the Independent Spent Fuel Storage Installation. The submittal should combine sections common to both units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, electronic personal dosimeter, or thermoluminescent dosimeter. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----------------------------NOTE-----------------------------
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results

5.6 Reporting Requirements

of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10 CFR Part 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

------------------------------NOTE------------------------------
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a, as modified by approved exemptions. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM, Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	SHUTDOWN MARGIN
3.1.3	Moderator Temperature Coefficient
3.1.4	CEA Alignment
3.1.6	Regulating Control Element Assembly Insertion Limit
3.2.1	Linear Heat Rate
3.2.2	Total Planar Radial Peaking Factor
3.2.3	Total Integrated Radial Peaking Factor
3.2.5	AXIAL SHAPE INDEX
3.3.1	RPS Instrumentation - Operating
3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CENPD-199-P, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems"

2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II"

5.6 Reporting Requirements

3. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2"

4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2"

5. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2"

6. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

7. CEN-348(B)-P, "Extended Statistical Combination of Uncertainties"

8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

9. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core"

10. CENPD-162-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

11. CENPD-207-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

12. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods"

5.6 Reporting Requirements

13. CENPD-225-P-A, "Fuel and Poison Rod Bowing"

14. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design"

15. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

16. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers"

17. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report"

18. CEN-161-(B)-P-A, "Improvements to Fuel Evaluation Model"

19. CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model"

20. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

21. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure"

22. Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and <u>W</u> Designed NSSS"

24. CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis"

5.6 Reporting Requirements

25. CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core"

26. Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"

27. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program"

28. Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

29. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

30. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident"

31. Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

32. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup"

33. Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

5.6 Reporting Requirements

34. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

35. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

36. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

37. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

38. Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

39. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

40. CENPD-188-A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

41. The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:

    i. CENPD-153-P, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

5.6 Reporting Requirements

ii. CEN-119(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2"

iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)

43. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel"

44. CENPD-199-P, Supplement 2-P-A, Appendix A, "CE Setpoint Methodology"

45. CENPD-404-P-A, Latest Approved Revision, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs"

46. CENPD-132, Supplement 4-P-A, Latest Approved Revision, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model"

47. CENPD-137, Supplement 2-P-A, Latest Approved Revision, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model"

5.6 Reporting Requirements

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Not Used

5.6.7 Post-Accident Monitoring Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.9 Steam Generator Tube Inspection Report

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 NRC within 15 days.

5.6 Reporting Requirements

b. The complete results of the steam generator tube inservice inspection during the report period shall be submitted to the NRC prior to March 1 of each year. 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; and

   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.