(3) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) FENOC, pursuant to the Act and 10 CFR Parts 30, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Auxiliary River Water System

(Deleted by Amendment No. 8)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

3. Reactor Coolant System LEAKAGE through a steam generator to the secondary system (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 is calculated with the following equation:

$$C_{I-131_{D.E.}} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

Where "C" is the concentration, in microcuries/gram of the iodine isotopes. This equation is based on dose conversion factors derived from ICRP-30.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTION:

- - - - - - - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - -

Separate action statement entry is allowed for each SG tube.

- -

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program:

   1. Verify within 7 days that tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.

   2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering MODE 4 following the next refueling outage or SG tube inspection.

b. With Action a not being completed within the specified completion time or if SG tube integrity is not being maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering MODE 4 following a SG tube inspection.

(next page is 3/4 4-11)

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 150 gallons per day primary to secondary LEAKAGE through any one steam generator, and

d. 10 gpm identified LEAKAGE.

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

ACTION:

a. With any Reactor Coolant System operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce the LEAKAGE to within limits within 4 hours.

b. With the required action and associated completion time of Action a not met, or with pressure boundary LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System operational LEAKAGES shall be demonstrated to be within each of the above limits by:

a. Monitoring the following leakage detection instrumentation at least once per 12 hours:(1)

1. Containment atmosphere gaseous radioactivity monitor.

(1) Only on leakage detection instrumentation required by LCO 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

2. Containment atmosphere particulate radioactivity monitor.

3. Containment sump discharge flow monitor.

4. Containment sump narrow range level monitor.

b. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.[2][3]

c. Verifying primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator at least once per 72 hours.[2]

(2) Not required to be performed until 12 hours after establishment of steady state operation.

(3) Not applicable to primary to secondary LEAKAGE.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Continued)

The methodology listed in WCAP-14040-NP-A was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.9.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.19, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

Containment Leakage Rate Testing Program (Continued)

b. Air Lock testing acceptance criteria and required action are as stated in Specification 3.6.1.3 titled "Containment Air Locks."

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.18 Technical Specifications (TS) 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 TS shall be made under appropriate administrative controls and reviews.

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 TS incorporated in the license; or

   2. a change to the updated FSAR 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 FSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b.1 & 2 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).

6.19 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for Condition Monitoring Assessments

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by

Steam Generator Program (Continued)

other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Provisions for Performance Criteria for SG Tube Integrity

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is also not to exceed 1 gpm per SG, except during a SG tube rupture.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2.

c. Provisions for SG Tube Repair Criteria

Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

Steam Generator Program (Continued)

d. Provisions for SG Tube Inspections

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. During each period inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three intervals between refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one interval between refueling outages (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE transactions shall have no effect on the license for the BVPS Unit 2 facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 158, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary LEAKAGE, or

3. Reactor Coolant System LEAKAGE through a steam generator to the secondary system (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

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

ACTION:

- - - - - - - - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - -

Separate action statement entry is allowed for each SG tube.

- -

a. With one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program:

   1. Verify within 7 days that tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.

   2. Plug or repair the affected tube(s) in accordance with the Steam Generator Program prior to entering MODE 4 following the next refueling outage or SG tube inspection.

b. With Action a not being completed within the specified completion time or if SG tube integrity is not being maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program prior to entering MODE 4 following a SG tube inspection.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 150 gallons per day primary to secondary LEAKAGE through any one steam generator, and

d. 10 gpm identified LEAKAGE.

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

ACTION:

a. With any Reactor Coolant System operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce the LEAKAGE to within limits within 4 hours.

b. With the required action and associated completion time of Action a not met, or with pressure boundary LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System operational LEAKAGES shall be demonstrated to be within each of the above limits by:

a. Monitoring the following leakage detection instrumentation at least once per 12 hours:(1)

1. Containment atmosphere gaseous radioactivity monitor.

---

(1) Only on leakage detection instrumentation required by LCO 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

2. Containment atmosphere particulate radioactivity monitor.

3. Containment sump discharge flow monitor.

4. Containment sump narrow range level monitor.

b. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.(2)(3)

c. Verifying primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator at least once per 72 hours.(2)

(2) Not required to be performed until 12 hours after establishment of steady state operation.

(3) Not applicable to primary to secondary LEAKAGE.

PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.9.7 STEAM GENERATOR TUBE INSPECTION REPORT

1. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.19, Steam Generator (SG) Program. The report shall include:

   a. The scope of inspections performed on each SG,

   b. Active degradation mechanisms found,

   c. Nondestructive examination techniques utilized for each degradation mechanism,

   d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

   e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

   f. Total number and percentage of tubes plugged or repaired to date,

   g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

   h. The effective plugging percentage for all plugging and tube repairs in each SG, and

   i. Repair method utilized and the number of tubes repaired by each repair method.

2. A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.19, Steam Generator Program, when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

STEAM GENERATOR TUBE INSPECTION REPORT (continued)

3. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

   a. If circumferential crack-like indications are detected at the tube support plate intersections.

   b. If indications are identified that extend beyond the confines of the tube support plate.

   c. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiological Work Permit[1]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

---

(1) Radiation protection personnel, or personnel escorted by radiation protection personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM (Continued)

2. a change to the updated FSAR 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 FSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b.1 & 2 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).

6.19 STEAM GENERATOR (SG) PROGRAM

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for Condition Monitoring Assessments

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Provisions for Performance Criteria for SG Tube Integrity

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and, except for flaws addressed

STEAM GENERATOR PROGRAM (Continued)

through application of the alternate repair criteria discussed in Specification 6.19.c.4, a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

When alternate repair criteria discussed in Specification 6.19.c.4 are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG will burst under postulated main steam line break conditions shall be less than $1x10^{-2}$.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Except during a steam generator tube rupture, leakage from all sources excluding the leakage attributed to the degradation described in TS Section 6.19.c.4 is also not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2.

c. Provisions for SG Tube Repair Criteria

1. Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in Specification 6.19.c.4.

STEAM GENERATOR PROGRAM (Continued)

2. Tubes with sleeves found by inservice inspection to contain flaws that are not in the sleeve to tube joint, with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness, shall be plugged:

ABB Combustion Engineering TIG welded sleeves	27%
Westinghouse laser welded sleeves	25%

3. Tubes with a flaw in a sleeve to tube joint shall be plugged.

4. The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria of Technical Specification 6.19.c.1:

Tube Support Plate Voltage-Based Repair Criteria

Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is described below:

a) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

b) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 6.19.c.4.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation.

STEAM GENERATOR PROGRAM (Continued)

d) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) will be plugged or repaired.

e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits specified in 6.19.c.4.a, 6.19.c.4.b, 6.19.c.4.c and 6.19.c.4.d.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC). The NDE is the value provided by the NRC in GL 95-05 as supplemented.

Implementation of these mid-cycle repair limits should follow the same approach as in Specifications 6.19.c.4.a through 6.19.c.4.d.

STEAM GENERATOR PROGRAM (Continued)

d. Provisions for SG Tube Inspections

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one interval between refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one interval between refueling outages (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

STEAM GENERATOR PROGRAM (Continued)

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (6.19.c.4) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

e. Provisions for monitoring operational primary to secondary LEAKAGE

f. Provisions for SG Tube Repair Methods

   Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

   1. ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

   2. Westinghouse laser welded sleeves, WCAP-13483, Revision 2.