ATTACHMENT 2

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To

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d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 500 gallons per day through any one steam generator except when the tube support plate, voltage-based repair criteria is applied. Primarily to secondary leakage is limited to an average of 150 gallons per day through any one steam generator when the tube support plate voltage-based repair criteria is applied. With tube leakage greater than the above limit, reduce the leakage rate within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable.

Leakage of Reactor Coolant (TS 3.1.d)[18]

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}\,\mu$ Ci/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.5 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve (12) hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

The 150 gpd average leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

[18]USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.3

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of I0 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

TS 3.1.d.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

TS 3.1.d.5

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

   If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.I and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[19]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[20]. Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[21][22]

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

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[19]USAR Section 4.2

[20]USAR Section 9.2

[21]USAR Table 3.2-1

[22]USAR Figure 3.2-8

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[23] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.2 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[24]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

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(23) USAR Figure 3.2-9

(24) USAR Table 3.2-1

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Steam and Power Conversion System.

OBJECTIVE

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

SPECIFICATION

a. Steam Generators

1. The reactor shall not be heated > 350°F unless the following conditions are satisfied.

   A. Two steam generators are OPERABLE.

      I. System piping and valves directly associated with providing auxiliary feedwater flow to the steam generators are OPERABLE.

      2. Five main steam safety valves per OPERABLE steam generator are OPERABLE, except during required surveillance tests or during in-service testing of these valves and steam generators in accordance with 10 CFR 50.55a, provided that at least two main steam safety valves associated with the steam generator under test are OPERABLE.

   B. A minimum of 39,000 gallons of water is available in the condensate storage tanks and the Service Water System is capable of delivering an unlimited supply from Lake Michigan.

   C. The DOSE EQUIVALENT I-131 on the secondary side of the steam generators does not exceed 0.1 μCi/cc.

2. If, when the reactor is > 350°F, any one of the conditions of TS 3.4.a.1 cannot be met within 48 hours, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System < 350°F within an additional 12 hours

b. Auxiliary Feedwater Pumps

1. The reactor shall not be heated > 350°F unless the following conditions are met:

   A. Both motor-driven auxiliary feedwater pumps shall be operable.

   B. The turbine-driven auxiliary feedwater pump shall be operable, or if not demonstrated operable prior to > 350°F, it shall be declared inoperable when 350°F is exceeded.

2. If, when the reactor is > 350°F, any one of the following conditions of inoperability may exist during the time interval specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System < 350°F within an additional 12 hours

   A. One auxiliary feedwater pump may be inoperable for 72 hours.

   B. Two auxiliary feedwater pumps may be inoperable for 4 hours.

3. If, when the reactor is > 350°F, three auxiliary feedwater pumps are discovered to be inoperable, all LIMITING CONDITIONS FOR OPERATION requiring MODE changes shall be suspended until at least one auxiliary feedwater pump is restored to OPERABLE status. Upon discovery, action shall be initiated immediately to restore at least one auxiliary feedwater pump to OPERABLE status.

c. Turbine Overspeed Protection System

1. Reactor power shall not exceed 50% of rated power unless two of the three turbine overspeed protection systems are OPERABLE, except as provided by TS 3.4.c.2.

2. If two or more of the turbine overspeed protection systems are inoperable, then maintain power < 50% of rated power. When only two systems are OPERABLE, an individual system may be blocked for no longer than 4 hours to allow for testing.

BASIS

Steam Generators (TS 3.4.a)

Two steam generators are required to be OPERABLE when the average reactor coolant temperature is > 350°F to ensure that sufficient heat removal capability exists for power operation and decay heat removal. Although one steam generator would provide sufficient decay heat removal capability, two steam generators are required in order to provide the necessary redundancy to meet the single failure criterion. An OPERABLE steam generator is defined by TS 3.4.a.

The ten main steam safety valves (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr at 1181 lbs. pressure. The maximum full-power steam flow at 1721 MWTH is 7,449,000 lbs./hr; therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary. The requirement that five main steam safety valves per OPERABLE steam generator are available will assure sufficient steam relief capability.

Testing of the main steam system while the plant is in HOT SHUTDOWN conditions is permitted provided that at least two main steam safety valves associated with the steam generator under test are available to provide sufficient relief capacity to protect the system during the test.

The specified minimum water supply in the condensate storage tanks is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. When AC power is available, unlimited replenishment of the condensate storage supply is available from Lake Michigan through the Service Water System.

An evaluation was performed to determine the maximum permissible steam generator primary-to-secondary leak rate during a steam line break event. The evaluation considered both a preaccident and accident initiated iodine spike. The results of the evaluation show that the accident initiated spike yields the limiting leak rate. This evaluation was based on a 30 REM thyroid dose at the site boundary and initial primary and secondary coolant iodine activity levels of 1.0 μCi/gm and 0.1 μCi/gm DOSE EQUIVALENT I-131 respectively. A leak rate of 34.0 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the steam generator faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm, the standard operating leakage limit applied for the tube support plate voltage-based plugging criteria specified in TS 4.2.b.5.

Auxiliary Feedwater Pumps (TS 3.4.b)

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned to both steam generators; the discharge of the turbine-driven pump, which starts automatically, is aligned to backup both motor-driven pumps. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

It is acceptable to exceed 350°F with an inoperable turbine-driven auxiliary feedwater pump. However, operability of the pump must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

With no auxiliary feedwater pumps OPERABLE, action shall be taken to restore a pump as soon as possible. The action with three pumps inoperable is to maintain the plant in an operating condition in which the auxiliary feedwater system is not needed for heat removal. When one pump is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no auxiliary feedwater pumps available, there would be no feedwater to the steam generator to cool the plant to 350°F when the Residual Heat Removal System could be placed in operation.

Turbine Overspeed Protection System (TS 3.4.c)

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the redundant overspeed trip system (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two-out-of-three channels to trip.

REFERENCES

USAR Section 10
USAR Section 14.1

1. Steam Generator Sample Selection and Inspection

The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection

The tubes selected for each in-service inspection shall:

a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 3% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted below and in TS 4.2.b.2.b.

   Tubes left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

   Implementation of the steam generator tube support plate voltage-based plugging criteria requires a 100% 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 tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

d. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories, and actions taken as described in Table 4.2-2.

<u>Category</u>	<u>Inspection Results</u>
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.

NOTE: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. <u>Inspection Frequencies</u>

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service 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.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.2-2 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 a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.a.I.C or

   2. A seismic occurrence greater than the Operating Basis Earthquake, or

   3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

   4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded I00°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

4. Plugging Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied.

a. Any tube which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Tube repair shall be in accordance with the methods described in WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)" or CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves."

b. Any Westinghouse mechanical sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged prior to returning the steam generator to service. Figure TS 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

5. Tube Support Plate Voltage-Based Plugging Criteria

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage $\leq$ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but $\leq$ 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > 5.6 volts will be plugged or repaired.

d. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of TS 4.2.b.5.c. If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

Where:

V	=	measured voltage
V_{BOC}	=	voltage at BOC
Δt	=	time period of operation to unscheduled outage
CL	=	cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL}	=	9.6 volt for 7/8 inch tubes

6. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degradation.

   3. Identification of tubes plugged.

   4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with Specification 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.

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

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

   4. If the calculated conditional burst probability exceeds the threshold value, notify the NRC and provide an assessment of the safety significance of the occurrence.

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.9I and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

Technical Specification 4.2.b.5

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

Technical Specification 4.2.b.6

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under IO CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).