ATTACHMENT 2

TO

Letter from C. R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

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PROPOSED TS AMENDMENT NO. 119

Affected TS Sections

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TECHNICAL SPECIFICATIONS AND BASES

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability, that must be enforced to ensure safe operation of the facility.

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e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0; and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The nonautomatic Containment System isolation valves and blind flanges are closed as required.

2. The Reactor Containment Vessel and Shield Building equipment hatches are properly closed.

3. At least ONE door in both the personnel and the emergency airlocks is properly closed.

4. The required automatic Containment System isolation valves are OPERABLE or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed.

5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.

6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.b.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

5. FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals in Table TS 1.0-1.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq$ -5%	$\leq$ 140	~0
COLD SHUTDOWN	$\leq$ -1%	$\leq$ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	$\geq$ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	$\geq$ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to Figure TS 3.10-1			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. GASEOUS RADWASTE TREATMENT SYSTEM

   A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting off-gases from the primary coolant system and providing for delay or holdup for the purpose of reducing the total radioactivity released to the environment.

2. MEMBER(S) OF THE PUBLIC

   MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

3. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

   The ODCM shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

4. PROCESS CONTROL PROGRAM (PCP)

   The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, Federal and State regulations and other requirements governing the disposal of the radioactive waste.

5. PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other OPERATING condition, in such a manner that replacement air or gas is required to purify the confinement.

6. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

7. SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

8. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

9. VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature atmospheric cleanup systems (i.e., Auxiliary Building special ventilation, Shield Building ventilation, spent fuel pool ventilation) are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

10. VENTING

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other OPERATING conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, as used in system names, does not imply a VENTING process.

11. RADIOLOGICAL ENVIRONMENTAL MONITORING MANUAL (REMM)

The REMM shall contain the current methodology and parameters used in the conduct of the radiological environmental monitoring program.

p. STANDARD SHUTDOWN SEQUENCE

When a LIMITING CONDITION FOR OPERATION is not met, and a plant shutdown is required except as provided in the associated action requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 36 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the specified time limits as measured from the time of determination of the failure to meet the LIMITING CONDITION FOR OPERATION. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable when the plant is in COLD or REFUELING SHUTDOWN.

TABLE TS 1.0-1

FREQUENCY NOTATIONS

NOTATION	FREQUENCY
Shift	At least once per 12 hours
Daily	At least once per 24 hours
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly	At least once per 92 days
Semiannual	At least once per 184 days
Each Refueling Outage	Each refueling cycle not to exceed 18 months
N.A.	Not applicable

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

APPLICABILITY

Applies to the operational status of the Chemical and Volume Control System.

OBJECTIVE

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

SPECIFICATIONS

a. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

b. The reactor shall not be made critical unless the following conditions are satisfied, except as provided in TS 3.2.c.

   1. A minimum of TWO charging pumps shall be OPERABLE.

   2. BOTH boric acid transfer pumps shall be OPERABLE.

   3. At least ONE boric acid tank shall contain a minimum of 2,000 gallons of 11.5% to 13% by weight boric acid (19,700 to 23,000 ppm boron) solution at a temperature of at least 145°F.

   4. System piping and valves shall be OPERABLE to the extent of establishing flow paths from the boric acid tank(s) and the RWST Refueling Water Storage Tank to the Reactor Coolant System.

   5. TWO trains of heat tracing shall be OPERABLE for the above flow paths for concentrated boric acid.

c. Any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the HOT SHUTDOWN condition if OPERABILITY is not restored within the time specified, and it shall be placed in the COLD SHUTDOWN condition if OPERABILITY is not restored within an additional 48 hours.

   1. ONE of the OPERABLE charging pumps may be removed from service provided two pumps are again OPERABLE within 24 hours.

   2. ONE boric acid transfer pump may be out of service provided both pumps are OPERABLE within 24 hours.

   3. ONE train of heat tracing may be out of service provided it is restored to OPERABLE status within 48 hours.

BASIS - Chemical and Volume Control System (TS 3.2)

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with any one of the two boric acid transfer pumps. An alternate method of boration will be use of the charging pumps directly from the Refueling Water Storage Tank. A third method will be to use the Safety Injection pumps. There are two sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the suction of the charging pumps.

(2) The charging pumps can take suction directly from the Refueling Water Storage Tank containing a concentration of 1950 ppm boron solution. Reference is made to TS 3.3.b.1.A.

(3) The Safety Injection pumps can take their suctions from either the boric acid tanks or the Refueling Water Storage Tank.

The quantity of boric acid stored in either the boric acid tanks or the Refueling Water Storage Tank is sufficient to achieve COLD SHUTDOWN at any time during core life.

Approximately 1800 gallons of boric acid of at least 11.5% concentration (19700 ppm boron) is required to ensure COLD SHUTDOWN. A minimum of 2000 gallons in the boric acid tank is therefore specified. A minimum temperature of 145°F is required to ensure solution solubility. Two trains of heat tracing are installed on lines normally containing concentrated boric acid solution.

The capacity of each charging pump is 60 gpm. This is sufficient to provide make-up water requirements for the Reactor Coolant System in the event of an allowable leak which permits continued safe plant operation. Any two of the three installed charging pumps can be used to comply with TS 3.2.b.1.

There are two trains of boric acid heat tracing with each train powered from a separate safeguard power supply, and each train being made up of several individual circuits. An individual circuit can be removed from service indefinitely, provided that the temperature of the fluid in that circuit can be maintained greater than 145°F without reliance on the redundant heat trace circuit.

3.3 ENGINEERED SAFETY FEATURES AND AUXILIARY SYSTEMS

APPLICABILITY

Applies to the OPERATING status of Engineered Safety Features and Auxiliary Systems.

OBJECTIVE

To define those LIMITING CONDITIONS FOR OPERATION that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, and (2) to remove heat from containment in normal OPERATING and emergency situations.

SPECIFICATIONS

a. Accumulators

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by TS 3.3.a.2.

   A. Each accumulator is pressurized to at least 700 psig and contains 1250 $ft^3 \pm 25$ ft^3 of water with a boron concentration of at least 1900 ppm, and is not isolated.

   B. Accumulator isolation valves SI-20A and SI-20B shall be opened with their power breakers locked out at or before the Reactor Coolant System pressure exceeds 1000 psig.

2. During power operation or recovery from an inadvertent trip, ONE accumulator may be inoperable for a period of 1 hour. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within the next 6 hours.
   - Achieve HOT SHUTDOWN within the following 6 hours.
   - Achieve COLD SHUTDOWN within an additional 36 hours.

b. Safety Injection/Residual Heat Removal Systems

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by TS 3.3.b.2.

   A. The Refueling Water Storage Tank contains not less than 272,500 gallons of water with a boron concentration of at least 1950 ppm.

   B. TWO SI/RHR trains are OPERABLE with each train comprised of:

      1. ONE OPERABLE safety injection pump.

      2. ONE OPERABLE residual heat removal pump.

      3. ONE OPERABLE residual heat removal heat exchanger.

      4. An OPERABLE flow path consisting of all valves, piping and interlocks associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the selected boric acid tank and the Refueling Water Storage Tank upon a Safety Injection signal and after manual transfer taking suction from the containment sump.

   C. Isolation valves SI-9A, SI-11A and SI-11B in the discharge of the high head SIS and block valve SI-3 are in the open position with their power breaker locked out.

   D. During the Quarterly Valve Operation Surveillance Testing of the Safety Injection System it is permissible to close the hand operated valve isolating the Boric Acid Storage Tanks from the Safety Injection Pumps Suction. During this short test period an operator shall stand by the valve to open it if Safety Injection is required. He will have headset communication with the Control Room.

2. During power operation or recovery from an inadvertent trip, ONE SI/RHR train may be inoperable for a period of 72 hours.

   A. If the inoperability is due to a component in the Safety Injection System and OPERABILITY is not restored within 72 hours, then within 1 hour action shall be initiated to:

      - Achieve HOT STANDBY within the next 6 hours.
      - Achieve HOT SHUTDOWN within the following 6 hours.
      - Achieve COLD SHUTDOWN within an additional 36 hours.

   B. If the inoperability is due to a component in the Residual Heat Removal System and OPERABILITY is not restored within 72 hours, then within 1 hour action shall be initiated to:

      - Achieve HOT STANDBY within the next 6 hours.
      - Achieve HOT SHUTDOWN within the following 6 hours.
      - Achieve and maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods within an additional 36 hours.

c. Containment Cooling Systems

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by TS 3.3.c.2.

   A. A minimum of 300 gallons of not less than 30% by weight of NaOH solution is available as a containment spray additive.

   B. Two containment spray trains are OPERABLE with each train comprised of:

      1. ONE containment spray pump.

      2. An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the Refueling Water Storage Tank and the spray additive tank upon a Hi-Hi containment pressure signal and after manual transfer being supplied from the containment sump.

   C. TWO trains of containment fancoil units are OPERABLE with two fancoil units in each train.

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2. During power operation or recovery from inadvertent trip, any one of the following conditions of inoperability may exist during the time intervals specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within the next 6 hours.
- Achieve HOT SHUTDOWN within the following 6 hours.
- Achieve COLD SHUTDOWN within an additional 36 hours.

A. The quantity of NaOH solution available as a containment spray additive may be less than that specified in TS 3.3.c.1.A for a period of 48 hours.

B. One containment fancoil unit train may be out of service for 7 days provided the opposite containment fancoil unit train remains OPERABLE.

C. One containment spray train may be out of service for 72 hours provided the opposite containment spray train remains OPERABLE.

D. Both containment fancoil unit trains may be out of service for 72 hours provided both containment spray trains remain OPERABLE.

E. The same containment fancoil unit and containment spray trains may be out of service for 72 hours provided their opposite containment fancoil unit and containment spray trains remain OPERABLE.

d. Component Cooling System

1. The reactor shall not be made or maintained critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by TS 3.3.d.2.

   A. TWO component cooling water trains are OPERABLE with each train consisting of:

      1. ONE component cooling water pump
      2. ONE component cooling water heat exchanger
      3. An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions.

2. During power operation or recovery from an inadvertent trip, ONE component cooling water train may be inoperable for a period of 72 hours. If OPERABILITY is not restored within 72 hours, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within the next 6 hours.
   - Achieve HOT SHUTDOWN within the following 6 hours.
   - Achieve and maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods within an additional 36 hours.

e. Service Water System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by TS 3.3.e.2.

   A. TWO service water trains are OPERABLE with each train consisting of:

      1. TWO service water pumps

      2. An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking a suction from the forebay and supplying water to the redundant safeguards headers.

   B. The Forebay Water Level Trip System is OPERABLE.

2. During power operation or recovery from an inadvertent trip, ONE service water train may be inoperable for a period of 72 hours. If OPERABILITY is not restored within 72 hours, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within the next 6 hours.
   - Achieve HOT SHUTDOWN within the following 6 hours.
   - Achieve and maintain Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods within an additional 36 hours.

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BASIS - Engineered Safety Features and Auxiliary Systems (TS 3.3)

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near OPERATING temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore, to be conservative, most engineered safety features components and auxiliary cooling systems shall be fully OPERABLE.

The OPERABLE status of the various systems and components is to be demonstrated by periodic tests, defined by TS 4.5. These periodic tests ensure, with a high reliability, that the various systems will function properly if required to do so. A large fraction of these tests will be performed while the reactor is OPERATING in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full OPERABILITY within a relatively short time. LIMITING CONDITIONS OF OPERATION permit temporary outages of redundant components and are specified for specific time intervals that are consistent with minor maintenance. These permissible conditions and time intervals are specified in such a manner as to apply identically during sustained power operation and during recovery from an inadvertent trip. The transient condition of restart in the latter case in no way alters the types of safety features equipment nor the extent of redundancy that must be available.

Inoperability of a single component does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the plant design and thereby limits the ability to tolerate additional equipment failures. However, the equipment out-of-service times specified in the LIMITING CONDITIONS FOR OPERATION are a temporary relaxation of the single failure criterion, which, consistent with overall system reliability considerations, provides a limited time to restore equipment to the OPERABLE condition. If the inoperable component is not repaired within the specified allowable time period or a second component in the same or related system is found to be inoperable and cannot be repaired within the specified time, the reactor will initially be put in HOT STANDBY and subsequently in the HOT SHUTDOWN condition to reduce the stored energy in the Reactor Coolant System and to provide for the reduction of the decay heat from the fuel. These actions result in a reduction of the cooling requirements after a postulated loss-of-coolant accident. If the malfunction(s) are not corrected after the specified time in a HOT SHUTDOWN condition, the reactor will be placed in the COLD SHUTDOWN condition, utilizing normal shutdown and cooldown procedures. In the COLD SHUTDOWN condition there is no possibility of an accident that would release fission products or damage the fuel elements.

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[1]USAR Section 3.2

When the inoperable component is part of the Residual Heat Removal (RHR), Component Cooling Water (CCW) or Service Water (SW) Systems, the average Reactor Coolant System temperature (T_{avg}) will be maintained below 350°F through an alternate heat removal method. The various alternate heat removal methods include the redundant RHR train and the steam generators.

Assuming the reactor has been OPERATING at full-rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating HOT SHUTDOWN.

Time After Shutdown	Decay Heat, % of Rated Power
1 minute	4.5
30 minutes	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the HOT SHUTDOWN condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the HOT SHUTDOWN condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety features in order to effect repairs. Failure to complete repairs after placing the reactor in the HOT SHUTDOWN condition may be indicative of need for major maintenance, and in such cases the reactor should therefore be placed in the COLD SHUTDOWN condition.

The accumulator and Refueling Water Storage Tank conditions specified are consistent with those assumed in the LOCA analysis.[2]

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[2] USAR Section 14.3

The containment cooling function is provided by two systems: containment fancoil units and containment spray systems. The containment fancoil units and containment spray system protect containment integrity by limiting the temperature and pressure that could be experienced following a Design Basis Accident. The Limiting Design Basis accidents relative to containment integrity are the loss-of-coolant accident and steam line break. During normal operation, the fancoil units are required to remove heat lost from equipment and piping within the containment.[3] In the event of the Design Basis Accident, any one of the following combinations will provide sufficient cooling to limit containment pressure to less than design values: four fancoil units, two containment spray pumps, or two fancoil units plus one containment spray pump.[4]

In addition to heat removal, the containment spray system is also effective in scrubbing fission products from the containment atmosphere. Therefore, a minimum of one train of containment spray is required to remain OPERABLE in order to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water.[5][6]

Sodium Hydroxide (NaOH) is added to the spray solution for pH adjustment. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump.

The alkaline pH of the containment sump water inhibits the volatility of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the sump fluid. Test data has shown that no significant stress corrosion cracking will occur provided the pH is adjusted within 2 days following the Design Basis Accident.[7][8]

A minimum of 300 gallons of not less than 30% by weight of NaOH solution is sufficient to adjust the pH of the spray solution adequately. The additive will still be considered available whether it is contained in the spray additive tank or the Containment Spray System piping due to an inadvertent opening of the spray additive valves (CI-1001A and CI-1001B).

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[3] USAR Section 6.3

[4] USAR Section 6.4

[5] USAR Section 6.4.3

[6] USAR Section 14.3.5

[7] USAR Section 6.4

[8] Westinghouse Chemistry Manual SIP 5-1, Rev. 2, dated 3/77, Section 4.

One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load either following a loss-of-coolant accident, or during normal plant shutdown. If, during the post-accident phase, the component cooling water supply were lost, core and containment cooling could be maintained until repairs were effected.[9]

A total of four service water pumps are installed, and a minimum of two are required to operate during the postulated loss-of-coolant accident.[10] The service water valves in the redundant safeguards headers have to be OPERABLE in order for the components that they supply to be considered OPERABLE.

The various trains of equipment referred to in the specifications are separated by their power supplies (i.e.: SI Pump 1A, RHR Pump 1A, Valves SI-2A and SI-4A, etc.). Shared piping and valves are considered to be common to both trains of the systems (i.e.: SI-3, etc.).

The closure of the hand operated valve for a brief period of time during the surveillance testing of the automatic valves in the Safety Injection System will prevent dilution of the concentrated boric acid or loss of concentrated boric acid to the Refueling Water Storage Tank.

[9] USAR Section 9.3

[10] USAR Section 9.6

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the HOT SHUTDOWN margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon or boron.

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits:

   (i) Westinghouse Electric Corporation Fuel

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.14)/P \times K(Z)$ for $P > .5$

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.28) \times K(Z)$ for $P \leq .5$

   (ii) Siemens Nuclear Power Corporation

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z)$ for $P > .5$

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z)$ for $P \leq .5$

where:

P is the fraction of full power at which the core is OPERATING

K(Z) is the function given in Figure TS 3.10-2

Z is the core height location for the F_Q of interest

B. $F_{\Delta H}^N$ Limits

(i) For Siemens Nuclear Power Corporation and Westinghouse Electric Corporation fuel with burnup less than 24,000 MWD/MTU

$$F_{\Delta H}^N \times 1.04 \leq 1.55 [1 + 0.2(1-P)]$$

(ii) For Westinghouse Electric Corporation fuel with burnup exceeding 24,000 MWD/MTU.

$$F_{\Delta H}^N \times 1.04 \leq 1.52 [1 + 0.2(1-P)]$$

where:

P is the fraction of full power at which the core is OPERATING

2. If, for any measured hot channel factor, the relationships specified in TS 3.10.b.1 are not true, reactor power shall be reduced by a fractional amount of the design power to a value for which the relationships are true, and the high neutron flux trip setpoint shall be reduced by the same fractional amount. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied.

4. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core:

   A. Westinghouse Electric Corporation Fuel

   $F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.14)/P \times K(Z)$

   B. Siemens Nuclear Power Corporation

   $F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.28)/P \times K(Z)$

   where:

   P is the fraction of full power at which the core is OPERATING

   V(Z) is defined in Figure TS 3.10-6

   $F_Q^{EQ}(Z)$ is a measured F_Q distribution obtained during the target flux determination

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of TS 3.10.b.4 according to the following schedules with allowances for a 25% grace period:

   A. During the target flux difference determination or once per effective full-power monthly interval, whichever occurs first.

   B. Upon achieving equilibrium conditions after reaching a thermal power level ≥ 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.5.A.

   C. If a power distribution map indicates an increase in peak pin power, $F_{\Delta H}^{N}$, of 2% or more, due to exposure, when compared to the last power distribution map, either of the following actions shall be taken:

      i. $F_Q^{EQ}(Z)$ shall be increased by an additional 2% for comparison to the relationship specified in TS 3.10.b.4, OR

      ii. $F_Q^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every 7 effective full-power days until a power distribution map indicates that the peak pin power, $F_{\Delta H}^{N}$, is not increasing with exposure when compared to the last power distribution map.

6. If, for a measured F_Q^{EQ}, the relationships of TS 3.10.b.4 are not satisfied and the relationships of TS 3.10.b.1 are satisfied, within 12 hours take one of the following actions:

   A. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships specified in TS 3.10.b.4 are satisfied, OR

   B. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand sides of the relationships specified in TS 3.10.b.4 exceed the limits specified in the right hand sides. Reactor power may subsequently be increased provided that a power distribution map verifies that the relationships of TS 3.10.b.4 are satisfied with at least 1% of margin for each percent of power level to be increased.

7. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full-power month.

8. The indicated axial flux difference shall be considered outside of the limits of TS 3.10.b.9 through TS 3.10.b.12 when more than one of the OPERABLE excore channels are indicating the axial flux difference to be outside a limit.

9. Except during physics tests, during excore detector calibration and except as modified by TS 3.10.b.10 through TS 3.10.b.12, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference.

10. At a power level ≥ 90% of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band within 15 minutes or reactor power shall be reduced to a level no greater than 90% of rated power.

11. At power levels > 50% and ≤ 90% of rated power:

A. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of 1 hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by -10% and +10% from the target axial flux difference at 90% rated power and increasing by -1% and +1% from the target axial flux difference for each 2.7% decrease in rated power < 90% and > 50%. If the cumulative time exceeds 1 hour, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

If the indicated axial flux difference exceeds the outer envelope defined above, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

B. A power increase to a level > 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

12. At a power level no greater than 50% of rated power:

A. The indicated axial flux difference may deviate from its target band.

B. A power increase to a level > 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than 2 hours (cumulative) of the preceding 24-hour period.

One half of the time the indicated axial flux difference is out of its target band, up to 50% of rated power is to be counted as contributing to the 1 hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

13. Alarms shall normally be used to indicate nonconformance with the flux difference requirement of TS 3.10.b.10 or the flux difference time requirement of TS 3.10.b.11.A. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within 2 hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

   A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

   B. If the tilt condition is not eliminated within 12 hours, the reactor shall be brought to a minimum load condition (≤ 30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, the reactor shall immediately be brought to a no load condition (≤ 5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limit is shown in Figure TS 3.10-3.

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure TS 3.10-1 must be maintained except for the Low Power Physics Test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In TS 3.10.e.1 and TS 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing.

1. When reactor power is ≥ 85% of rating, the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is ≥ 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 85% of rating.

2. When reactor power is < 85% but ≥ 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is < 85% but ≥ 50%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If the peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 50% of rating.

3. And, in addition to TS 3.10.e.1 and TS 3.10.e.2, if the misaligned rod cluster control assembly is not realigned within 8 hours, the rod shall be declared inoperable.

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:

   A. For operation between 50% and 100% of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) at least once per 8 hours, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

   B. During operation < 50% of rating, no special monitoring is required.

2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from TS 3.10.f.1.A, then TS 3.10.e will be applied.

g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under TS 3.10.e or TS 3.10.h.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made OPERABLE earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At OPERATING temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged at least once per 8 hours after a load change ≥ 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change ≥ 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. During steady-state 100% power operation, T_{inlet} shall be maintained $\leq$ 536.5°F.

l. During steady-state 100% power operation, Reactor Coolant System pressure shall be maintained $\geq$ 2200 psig.

m. During steady-state power operation, reactor coolant flow rate shall be $\geq$ 92,560 gallons per minute average per loop; or the $F_{\Delta H}^{N}$ hot channel factor limit for fuel of $\geq$ 15,000 MWD/MTU shall be reduced 1% for every 1.8% of reactor coolant loop design flow below 92,560 gallons per minute. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow after each REFUELING.

BASIS

Shutdown Reactivity (TS 3.10.a)

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits of Figure TS 3.10-3 must be observed. In addition, for HOT SHUTDOWN conditions, the shutdown margin of Figure TS 3.10-1 must be provided for protection against the steam line break accident which requires more shutdown reactivity at end of core life (due to a more negative moderator temperature coefficient at end-of-life boron concentrations).

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity or excess beyond needs decreases with decreasing boron concentration, because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.

The exception to the rod insertion limits in TS 3.10.d.3 is to allow the measurement of the worth of all rods less the worth of the worst case of an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest, such as end-of-life cooldown or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Power Distribution Control (TS 3.10.b)

Criteria

Criteria have been chosen for Condition I and II events as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed the value assumed in the accident analysis.[1][2] Second, the minimum DNBR in the core must not be < 1.30 in normal operation or in short-term transients.[3]

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss-of-coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F.

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local neutron flux in the core at core elevation Z divided by the core averaged neutron flux assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured F_Q^N distribution obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for F_Q^N defined by TS 3.10.b.1 has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures remain < the 2200°F limit.

The $F_Q^N(Z)$ limits of TS 3.10.b.1.A include consideration of enhanced fission gas release at high burnup, off-gassing (release of absorbed gases), and other effects in fuel supplied by Siemens Nuclear Power Corporation. The result of these analyses show that no additional burnup dependent penalty need be applied for Siemens Nuclear Power Corporation fuel[4].

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[1] USAR Section 4.3

[2] USAR Section 14

[3] USAR Section 4.4

[4] M.S. Stricker, "Kewaunee High Burnup Safety Analysis: Limiting Break LOCA and Radiological Consequences," ZN-NF-84-31 Rev. 1, Exxon Nuclear Company, October 1984.

When a F_Q^N measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent (5%) is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

In TS 3.10.b.1 and TS 3.10.b.4 F_Q^N is arbitrarily limited for $P \leq 0.5$ (except for Low Power Physics Tests).

<u>$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor</u>

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

In the specified limit of $F_{\Delta H}^N$, there is an 8% allowance for design protection uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4% is the appropriate allowance, as specified in TS 3.10.b.1. The logic behind the larger design uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q^N; (b) the operator has a direct influence on F_Q^N through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$; and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5 is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects need be included in TS 3.10.b.1 for Siemens Nuclear Power Corporation fuel rod burnups to 49,000 MWD/MTU.[5] Westinghouse Electric Company fuel requires a burnup dependent penalty be incorporated through a decrease in the $F_{\Delta H}^{N}$ limit of 2% for 0-15,000 MWD/MTU fuel burnup, 4% for 15,000-24,000 MWD/MTU fuel burnup, and 6% for greater than 24,000 MWD/MTU fuel burnup. These penalties are counter-balanced by credits for increased Reactor Coolant flow and lower core inlet temperature. The Reactor Coolant System flow has been determined to exceed design flow by ≥ 8%. Since the flow channel protective trips are set on a percentage of full flow, significant margin to DNB is provided. One half of the additional flow is taken as a DNB credit to offset 2% of the $F_{\Delta H}^{N}$ penalty. The existence of 4% additional reactor coolant flow will be verified after each refueling at power prior to exceeding 95% power. If the reactor coolant flow measured per loop averages ≤ 92,560 gpm, the $F_{\Delta H}^{N}$ limit shall be reduced at the rate of 1% for every 1.8% of reactor coolant design flow (89,000 gpm design flow rate) for fuel with greater than 15,000 MWD/MTU burnup. Uncertainties in reactor coolant flow have already been accounted for in flow channel protective trips for design flow. The assumed T_{inlet} for DNB analysis was 540°F while the normal T_{inlet} at 100% power is approximately 532°F. The reduction of maximum allowed T_{inlet} at 100% power to 536°F as addressed in TS 3.10.k provides an additional 2% credit to offset the rod bow penalty. The combination of the penalties and offsets results in a required 2% reduction of allowed $F_{\Delta H}^{N}$ for high burnup fuel (assembly burnups > 24,000 MWD/MTU). The permitted relaxation of $F_{\Delta H}^{N}$ allows radial power shape changes with rod insertion to the insertion limits.

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

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[5] N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥ 85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-3.

3. The control bank insertion limits are not violated.

4. Axial power distribution control specifications which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.[6]

Conformance with TS 3.10.b.9 through TS 3.10.b.12 ensures the F_Q^N upper bound envelope is not exceeded and xenon distributions will not develop which at a later time would cause greater local power peaking.

At the beginning of cycle, power escalation may proceed without the constraints of TS 3.10.b.5 since the startup test program provides adequate surveillance to ensure peaking factor limits. Target flux difference surveillance is initiated after achieving equilibrium conditions for sustained operation.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is determined from the nuclear instrumentation. This value, divided by the fraction of full power at which the core was OPERATING is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations of ± 5% flux difference are permitted from the indicated reference value. Figure TS 3.10-5 shows a typical construction of target flux difference band at BOL and Figure TS 3.10-4 shows the typical variation of the full power value with burnup.

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[6] XN-NF-77-57 Exxon Nuclear Power Distribution Control for Pressurized Water Reactor, Phase II, January 1978.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of 1 hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference in the range +10% to -10% from the target flux increasing by ± 1% from the target axial flux difference for each 2.7% decrease in rated power ≤ 90% and ≥ 50%. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the ± 5% band for as long a period as 1 hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full-power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Power Tilt Limits (TS 3.10.c)

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The 2% tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by TS 4.1.

The 2 hour time interval in TS 3.10.c is considered ample to identify a dropped or misaligned rod. If the tilt condition cannot be eliminated within the 2 hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map using the movable detector system. For a tilt ratio > 1.02 but ≤ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of 2% for each 1% of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. If a tilt ratio of > 1.02 but ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to ≤ 50%.

If a misaligned rod has caused a tilt ratio > 1.09, the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0. If after 8 hours the rod has not been realigned, the rod shall be declared inoperable in accordance with TS 3.10.e, and action shall be taken in accordance with TS 3.10.g. If the tilt condition cannot be eliminated after 12 hours, the reactor shall be brought to a minimum load condition; i.e., electric power ≤ 30 MW. If the cause of the tilt condition has been identified and is in the process of being corrected, the generator may remain connected to the grid.

If the tilt ratio is ≥ 1.09, and it is not due to a misaligned rod, the reactor shall be brought to a no load condition (i.e., reactor power ≤ 5%) for investigation by flux mapping. Although the reactor may be maintained critical for flux mapping, the generator must be disconnected from the grid since the cause of the tilt condition is not known, or it cannot be readily corrected.

Rod Misalignment Limitations (TS 3.10.e)

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of TS 3.10.e. A relaxation in those limits for power levels ≤ 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while OPERATING at lower power levels. This increased flexibility is desirable to account for the nonlinearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators, the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

The plant process computer converts the output voltage signal from each IRPI conditioning module to an equivalent position (in steps) through a curve fitting process, which may include the latest actual voltage-to-position rod calibration curve.

The rod position as determined by any of these methods can then be compared to the bank demand position which is indicated on the group step counters to determine the existence and magnitude of a rod misalignment. This comparison is performed automatically by the plant process computer. The rod deviation monitor on the annunciator panel is activated (or reactivated) if the two position signals for any rod as detected by the process computer deviate by more than a predetermined value. The value of this setpoint is set to warn the operator when the Technical Specification limits are exceeded.

The rod position indicator system is calibrated once per REFUELING cycle and forms the basis of the correlation of rod position vs. voltage. This calibration is typically performed at HOT SHUTDOWN conditions prior to initial operations for that cycle. Upon reaching full-power conditions and verifying that the rods are aligned with their respective banks, the rod position indication may be adjusted to compensate for the effects of the power ascension. After this adjustment is performed, the calibration of the rod position indicator channel is checked at an intermediate and low level to confirm that the calibration is not adversely affected by the adjustment.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

DNB Parameters (TS 3.10.n)

The DNB related accident analysis assumed as initial conditions that the T_{inlet} was 4°F above nominal design or T_{avg} was 4°F above nominal design. The Reactor Coolant System pressure was assumed to be 30 psi below nominal design.

4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

APPLICABILITY

Applies to in-service structural surveillance of the ASME Code Class components and supports and functional testing of pumps and valves.

OBJECTIVE

To assure the continued integrity and operational readiness of ASME Code Class 1, 2 and 3 components.

SPECIFICATION

a. ASME Code Class 1, 2 and 3 Components and Supports

1. In-service inspection of ASME Code Class 1, Class 2 and Class 3 components and supports shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i). The testing and surveillance of shock suppressors (snubbers) is detailed in TS 3.14 and TS 4.14.

2. In-service testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

3. Surveillance testing of pressure isolation valves:

   a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the OPERATING mode after every time the plant is placed in the COLD SHUTDOWN condition for refueling, after each time the plant is placed in a COLD SHUTDOWN condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

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[1]To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

b. Steam Generator Tubes

Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defined to clarify the requirements of the inspection program.

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

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

% Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

Degraded Tube means a tube contains an imperfection $\geq$20% of the nominal wall thickness caused by degradation.

Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

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

Tube is the Reactor Coolant System pressure boundary past the hot leg side of the tubesheet and before the cold leg side of the tubesheet.

Plugged Tube is a tube intentionally removed from service by plugging in the hot and cold legs because it is defective, or because its continued integrity could not be assured.

Repaired Tube is a tube that has been modified to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube.

1. <u>Steam Generator Sample Selection and Inspection</u>

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. <u>Steam Generator Tube Sample Selection and Inspection</u>

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 in 4.2.b.2.b.

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.

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.

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.

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

3. Inspection Frequencies

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.1.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 100°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:

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. 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).

BASIS

The plant was not specifically designed to meet the requirements of Section XI of the ASME Code; therefore, 100% compliance may not be feasible or practical. However, access for in-service inspection was considered during the design and modifications have been made where practical to make provisions for maximum access within the limits of the current plant design. Where practical, the inspection of ASME Code Class 1, Class 2 and Class 3 components is performed in accordance with Section XI of the ASME Code. If a code required inspection is impractical, a request for a deviation from the requirement is submitted to the Commission for approval.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table TS 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b

These Technical Specifications provide the inspection and repair/plugging requirements for the steam generator tubes at the Kewaunee Nuclear Power Plant. Fulfilling these specifications will assure the KNPP steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria in 10 CFR Part 50, Appendix A.

General Design Criterion (GDC) 14 "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires that the Reactor Coolant System and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Furthermore, GDC 32 "Inspection of Reactor Coolant System Pressure Boundary," requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of critical areas to assess their structural and leak tight integrity.

The NRC has developed guidance for steam generator tube inspections and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted prior to any revisions. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines the minimum wall thickness in a steam generator tube, and may be applied to tube sleeves in determining their minimum wall thickness.

Technical Specification 4.2.b.1

If the steam generators are shown to be performing in a like manner, it is appropriate to limit the inspection to one steam generator on a rotating schedule. Economic savings as well as reductions in personnel exposure and outage duration can be realized.

Technical Specification 4.2.b.2

Periodic inspection of the steam generator tubes allows evaluation of their service condition. As operational experience has become available it is evident that certain types of steam generators are susceptible to generic degradation mechanisms. Site specific steam generator tube degradation has also occurred throughout the industry. The inspection program at Kewaunee is designed to identify both generic and site specific tube degradation mechanisms.

Steam generator tube surveillance at Kewaunee is generally performed using eddy current techniques. Various methods of eddy current (EC) testing are used to inspect steam generator tubes for wall degradation. EC methods have improved considerably since Kewaunee began commercial operation in 1974. Single frequency EC testing with a single probe and X-Y plotter have evolved into multifrequency techniques with assorted probe types and sophisticated software to allow more accurate volumetric tube examinations. Profilometery techniques are also being developed which detect imperfections in a tube's original geometry. WPSC is committed to utilize advancing EC testing technology, as appropriate, to assure accurate determination of the steam generator tubes' service condition.

Technical Specification 4.2.b.3

Steam generator tube inspections are generally scheduled during refueling outages at the Kewaunee Nuclear Power Plant. The tubes scheduled for a given inspection are based upon their service condition determined during previous inspections, and operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tube conditions results in augmentation of the inspection effort as well as increasing the frequency of subsequent inspections. In this manner, steam generator tube surveillance is consistent with service conditions.

There are several operational occurrences or transients that will require subsequent steam generator tube inspections. These inspections are required as a result of excessive primary-to-secondary leakage or transients imposing large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

Steam generator tubes found with less than the minimum wall thickness criteria determined by analysis, as described in WCAP-7832[1][2],, must either be repaired to be kept in service or removed from service by plugging.

Steam generator tube plugging is a common method of preventing primary-to-secondary steam generator tube leakage and has been utilized since the inception of PWR nuclear reactor plants. This method is relatively uncomplicated from a structural/mechanical standpoint as flow is cut off from the affected tube by plugging it in the hot and cold leg faces of the tubesheet.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976).

For the Westinghouse mechanical sleeves, the sleeve plugging limit of 31% is applied to the sleeve as shown on Figure TS 4.2-1. For the Combustion Engineering leak tight sleeves, a plugging limit of 40% is applied to the sleeve and weld region. The sleeve plugging limits allow for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

Repair by sleeving, or other methods, has been recognized as a viable alternative for isolating unacceptable tube degradation and preventing tube leakage. Sleeving isolates unacceptable degradation and extends the service life of the tube, and the steam generator. Tube repair, by sleeving in accordance with WCAP-11643[3] and CEN-413-P[4] has been evaluated and analyzed as acceptable. The Westinghouse mechanical sleeve spans the degraded area of the parent tube in the tubesheet region. The sleeves are either 36", 30" or 27" to allow access permitted by channel head bowl geometry. The sleeve is hydraulically expanded and hard rolled into the parent tubing.

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[1]WCAP 7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions."

[2]E. W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

[3]WCAP 11643, Kewaunee Steam Generator Sleeving Report, Revision 1, November 1988 (Proprietary).

[4]CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves," January 1992 (Proprietary).

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 50.91 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

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

6.5 REVIEW AND AUDIT

a. Plant Operations Review Committee (PORC)

1. Function

   The PORC shall function to advise the Manager - Kewaunee Plant on matters related to nuclear safety.

2. Composition

   The PORC shall be composed of, but not necessarily limited to:

Chairman:	Manager - Kewaunee Plant
Required Members:	Assistant Manager - Plant Operations Assistant Manager - Plant Maintenance Superintendent - Plant Operations Superintendent - Plant Instrument and Control Plant Reactor Supervisor Superintendent - Plant Quality Programs Superintendent - Plant Radiation Protection

3. Alternates

   Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates for required members shall participate in PORC meetings at any one time.

4. Meeting Frequency

   The PORC shall meet at least once per calendar month and as convened by the Chairman.

5. Quorum

   A quorum of the PORC shall consist of the chairman (or his designated alternate as stated in TS 6.1) and a majority of the required members including temporary alternates.

6. Responsibilities

   The PORC shall be responsible for:

   A. Review of operating, maintenance and other procedures including emergency operating procedures which affect nuclear safety as determined by the Manager - Kewaunee Plant. Changes to those procedures are made in accordance with the provisions of TS 6.8.a.

B. Review of all proposed tests and experiments that affect nuclear safety.

C. Review of all proposed changes to the Technical Specifications.

D. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

E. Review of all proposed changes to the Security Plan, Emergency Plan, Fire Plan, and their respective implementing procedures.

F. Review all reports covering the investigation of all violations of the Technical Specifications and the recommendations to prevent recurrence.

G. Review plant operations to detect potential safety hazards.

H. Performance of special reviews and investigations and prepare reports thereon as requested by the Chairman of the Nuclear Safety Review and Audit Committee.

I. Review of all REPORTABLE EVENTS

J. Review of changes to the PROCESS CONTROL PROGRAM, the OFF-SITE DOSE CALCULATION MANUAL, and the RADIOLOGICAL ENVIRONMENTAL MONITORING MANUAL.

7. Authority

The PORC shall:

A. Recommend to the Manager - Kewaunee Plant approval or disapproval of items considered under TS 6.5.a.6.A through TS 6.5.a.6.E.

B. Make determinations with regard to whether or not each item considered under TS 6.5.a.6 constitutes an unreviewed safety question.

C. Provide immediate notification in the form of draft meeting minutes to the Senior Vice President - Nuclear Power and the Chairman-Nuclear Safety Review and Audit Committee of disagreement between the PORC and the Manager - Kewaunee Plant. The Manager - Kewaunee Plant shall have responsibility for resolution of such disagreements.

8. Records

Minutes shall be kept of all meetings of the PORC and copies shall be sent to the Senior Vice President - Nuclear Power and the Chairman - Nuclear Safety Review and Audit Committee.

b. Corporate Support Staff (CSS)

1. Function

The CSS shall function to provide engineering, technical and quality assurance activities in support of the Kewaunee Plant Staff.

2. Organization

The CSS consists of the following groups:

A. Nuclear Licensing and Systems
B. Nuclear Projects
C. Corporate Health Physics
D. Nuclear Project Management (Design Change)
E. Engineering Support
F. Engineering Control
G. Emergency Preparedness
H. Power Plant Design and Construction
I. Fuel Services
J. Administrative Staff
K. Quality Assurance
L. Substation and Transmission
M. Safety System Engineering

3. Activities

A. Review and report all violations of the Technical Specifications, codes, regulations, and statutes.

B. Review all activities associated with nuclear safety for technical adequacy and compliance with internal procedures or instructions.

C. Review and report significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

D. Review and report all events which are required by regulations or Technical Specifications to be reported to the NRC (Plant personnel will provide the initial reporting to the NRC of those events requiring 24 hour notification).

E. Investigate any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems or components.

F. Review and/or prepare safety evaluations of all plant design changes.

G. Audits as required by the Quality Assurance Program and as outlined in TS 6.5.c.8.

c. Nuclear Safety Review and Audit Committee (NSRAC) Function

1. Function

The NSRAC shall function to provide independent review and audit of designated activities in the areas of:

A. Nuclear Power Plant Operations
B. Nuclear Engineering
C. Chemistry and Radio-Chemistry
D. Metallurgy
E. Instrumentation
F. Radiological Safety
G. Mechanical and Electrical Engineering
H. Quality Assurance Practices
I. Other appropriate fields as determined by the Committee, to be associated with the unique characteristics of the nuclear power plant.

2. Composition

The NSRAC shall be composed of, but not necessarily limited to:

A. At least three technically qualified persons who are not members of the plant staff.

B. One member from the supervisory staff of the plant.

C. At least two qualified non-company affiliated technical consultants.

D. In-house staff management advisors as required.

The Committee membership and its Chairman and Vice Chairman shall be appointed by the Senior Company Officer to whom the NSRAC reports. Each member of the NSRAC shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum shall be in one or more areas given in TS 6.5.c.1.

3. Alternates

Alternate members shall be appointed by the NSRAC Chairman, upon approval by the Senior Vice President - Nuclear Power, to serve on a temporary basis; however, no more than two alternates shall participate in NSRAC activities at any one time.

4. Consultants

Consultants may be utilized as determined by the Chairman - NSRAC to provide expert advice to the NSRAC.

5. Meeting Frequency

The NSRAC shall meet at least once every six months.

6. Quorum

A quorum of the NSRAC shall consist of the Chairman or Vice Chairman and four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the plant.

7. Review

The NSRAC shall review:

A. Safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

B. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

C. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

D. Proposed changes in Technical Specifications or licenses.

E. Reports covering violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

F. Reports covering significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

G. Reports covering all REPORTABLE EVENTS.

H. Reports covering any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

I. Reports and meeting minutes of the PORC.

8. Audits

Audits of plant activities shall be performed under the cognizance of the NSRAC. These audits shall include:

A. Conformance of plant operation to the provisions contained within the Technical Specifications and applicable license conditions.

B. Performance, training, and qualifications of the entire plant staff.

C. Results of all actions taken to correct deficiencies occurring in plant equipment, structures, systems, or method of operation that affect nuclear safety.

D. Performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR Part 50.

E. The Plant Fire Protection Program, implementing procedures and the independent fire protection and loss prevention program.

F. Any other area of plant operation considered appropriate by the NSRAC or the Senior Company Officer to whom the NSRAC reports.

G. The Radiological Environmental Monitoring Program and the results thereof.

H. The OFF-SITE DOSE CALCULATION MANUAL and implementing procedures.

I. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes.

9. Authority

The NSRAC shall report to a Senior Company Officer and shall advise the Officer on those areas of responsibility specified in TS 6.5.c.7 and TS 6.5.c.8.

10. Records

Records of NSRAC activities shall be prepared, approved and distributed as follows:

A. Minutes of each NSRAC meeting forwarded to the Senior Company Officer to whom the NSRAC reports within 14 days following each meeting.

B. Reports of reviews required by TS 6.5.c.7.E through TS 6.5.c.7.H, forwarded to the Senior Company Officer to whom the NSRAC reports within 14 days following completion of the review.

C. Reports of audits performed by NSRAC shall be forwarded to the Senior Company Officer to whom the NSRAC reports and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

a. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the USAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Reporting Requirements

Routine operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to March 1 of each year. Items reported in this category include:

A. Report of facility changes, tests or experiments required pursuant to 10 CFR 50.59(b).

B. 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,[1] e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and REFUELING. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. 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 shall be assigned to specific major work functions.

C. Challenges to and failures of the pressurizer power operated relief valves and safety valves.[2]

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

(1) The Annual Radiological Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by TS 7.7.2.

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[1]This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

[2]Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (U.S. NRC) dated January 5, 1981.

(2) The Annual Radiological Environmental Monitoring Reports shall include the results of analysis of radiological environmental samples and of environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the RADIOLOGICAL ENVIRONMENTAL MONITORING MANUAL, 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 as soon as possible in a supplementary report when applicable.

(3) The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; legible maps covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by TS 7.7.3; discussion of all deviations from the sampling schedule of Table 7.3; and discussion of all analyses in which the LLD required by Table 8.5 was not achievable.

2. Radioactive Effluent Release Report

A. Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year.

(1) Radioactive Effluent

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit following the format of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974.

(2) Radiation Dose Assessment

The Radioactive Effluent Release Report shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.[3] This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. The assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed based on the calculational guidance, as presented in the OFF-SITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall also include an assessment of radiation doses to the likely most exposed MEMBER(S) OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

(3) Solid Waste Shipped

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped off-site during the report period:

a) Container volume,

b) Total curie quantity (specify whether determined by measurement or estimate),

c) Principal radionuclides (specify whether determined by measurement or estimate),

d) Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

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[3] In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

e) Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f) SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

(4) Unplanned Release

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

(5) PCP and ODCM Changes

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFF-SITE DOSE CALCULATION MANUAL (ODCM).

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

6.11 RADIATION PROTECTION PROGRAM

a. 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.

b. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne inplant iodine concentrations under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.17 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   A. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information;

   B. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   C. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

2. Shall become effective upon review and acceptance by the PORC.

6.18 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

a. The ODCM shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   A. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   B. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   C. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

2. Shall become effective upon review and acceptance by the PORC.

6.19 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS[1]

Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   2. Sufficient information to support the reason for the change without benefit of additional or supplemental information;

   3. A description of the equipment, components and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change, which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   7. An estimate of the exposure to plant operating personnel as a result of the change; and

   8. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

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[1]Licensees may choose to submit the information called for in this TS as part of the annual USAR update.