C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 172 are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	≤ -5%	≤ 140	~0
COLD SHUTDOWN	≤ -1%	≤ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	≥ 540	~0
HOT STANDBY	< 0.25%	$\sim T_{oper}$	< 2
OPERATING	< 0.25%	$\sim T_{oper}$	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

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 1,772 MWt.

n. REPORTABLE EVENT

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation.

c. The peak fuel centerline temperature shall be maintained < 5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

Amendment No. 172

BASIS - Safety Limits-Reactor Core (TS 2.1)

The reactor core safety limits shall not be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of the reactor core safety limits prevent overheating of the fuel and cladding as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and Condition I and II transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report show the loci of points of thermal power, reactor coolant system average temperature, and reactor coolant system pressure for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy at the exit of the core is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within limits defined by the DNBR correlation. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below the safety limit curves.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the increase in peaking factor is more than offset by the decrease in power level.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f_1(\Delta I) \right]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average temperature, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f_1(\Delta I)$	=	A function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f_1(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

Amendment No. 172

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f_2 (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' ≤ [*]°F

K_4 ≤ [*]

K_5 ≥ [*] for increasing T; [*] for decreasing T

K_6 ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f_2(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level ≥ 5% of narrow range instrument span.

Amendment No. 172

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.[4]

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5]

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System.[6]

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

---

[4] WCAP-8092
[5] USAR Section 14.1.8
[6] USAR Section 14.1.10

c. Containment Cooling Systems

1. Containment Spray and Containment Fancoil Units

A. 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.1.A.3.

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

(i) ONE containment spray pump.

(ii) 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 from the containment sump.

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

3. 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.

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

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

(iii) 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.

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.[(2)] 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 or two fancoil units plus one containment spray pump.[(3)]

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.[(4)(5)]

Sodium Hydroxide (NaOH) is added to the spray solution for pH adjustment by means of the spray additive system. 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.[(6)(7)]

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 and Refueling Water Storage Tank due to an inadvertent opening of the spray additive valves (CI-1001A and CI-1001B).

---

(2) USAR Section 6.3
(3) USAR Section 6.4
(4) USAR Section 6.4.3
(5) USAR Section 14.3.5
(6) USAR Section 6.4
(7) Westinghouse Chemistry Manual SIP 5-1, Rev. 2, dated 3/77, Section 4.

2. When the Reactor Coolant System temperature is > 350°F, if three auxiliary feedwater trains are discovered to be inoperable, initiate immediate action to restore one auxiliary feedwater train to OPERABLE status and suspend all LIMITING CONDITIONS FOR OPERATION requiring MODE changes until one auxiliary feedwater train is restored to OPERABLE status.

3. The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of the three AFW trains are inoperable, then within two hours, reduce reactor power to ≤ 1673 MWt.

4. When the Reactor Coolant System temperature is > 350°F, any of the following conditions of inoperability may exist during the time interval specified:

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

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

   C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days.

5. When the Reactor Coolant System temperature is > 350°F, an auxiliary feedwater pump low discharge pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded, the associated auxiliary feedwater train shall be declared inoperable and the OPERABILITY requirements of TS 3.4.b.3 and TS 3.4.b.4 applied.

6. If the OPERABILITY requirements of TS 3.4.b.4 above are not met within the times specified, then within 1 hour action shall be initiated to:

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

7. When reactor power is < 15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:

   A. The auxiliary feedwater pump control switches located in the control room may be placed in the "pull out" position.

   B. Valves AFW-2A and AFW-2B may be in a throttled or closed position.

   C. Valves AFW-10A and AFW-10B may be in the closed position.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum usable volume of 41,500 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum usable volume of 41,500 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

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

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is $\leq$ 0.1 μCi/gram.

2. When the Reactor Coolant System temperature is > 350°F, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

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

Two analyses apply to the Loss of Normal Feedwater event:

1. Analysis of the Loss of Normal Feedwater (LONF) event at 1772 MWt.

2. Analysis of the Loss of Normal Feedwater event at 1673 MWt.

One AFW pump provides adequate capacity to mitigate the consequences of the LONF event at 1673 MWt. In the LONF event at 1772 MWt, any two of the three AFW pumps are necessary to provide adequate heat removal capacity.

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following: .

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Amendment No. 172

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses performed at 1772 MWt to fulfill the above functions. The exception is the LONF accident analysis performed at 1772 MWt. Based on the LONF accident analysis at 1772 MWt, two AFW pumps are required to provide adequate capacity.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.4. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The OPERABILITY of the AFW system following a LONF event was analyzed as part of the stretch uprate. As a result of the analysis at 1772 MWt, requirements for three OPERABLE AFW trains prior to increasing power above 1673 MWt were added to the Technical Specifications. In a LONF event, it is assumed that one of the AFW pumps fails. Therefore, to meet single failure criteria, all three pumps are required to be OPERABLE prior to increasing power level above 1673 MWt.

For all design basis accidents other than MSLB and the LONF at 1772 MWt, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2, TS 3.4.b.3, and TS 3.4.b.4 are applied. The two and four hour clocks in TS 3.4.b.3 and TS 3.4.b.4 are started simultaneously. The two hour clock of TS 3.4.b.3 is for the power level restriction. The four hour clock of TS 3.4.b.4 is for starting the shutdown sequence. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated. This provides 72 hours with steam pressure for post-maintenance testing of the turbine AFW pump.

Condensate Storage Tank (TS 3.4.c)

The specified minimum usable water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. Total CST water supply is maintained above a level that includes minimum usable water supply in technical specifications based on the station blackout analysis, allowance for flow to the condenser before isolation, allowance for AFW pump cooling, unusable level, and instrument error in each tank's level instrument.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to ≤ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

---

[1] USAR Section 8.2.4
[2] USAR Section 14.0

BASIS – Refueling Operations (TS 3.8)

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. The boron concentration as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.[2] With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown bounds the assumption used in the dose calculation for the fuel handling accident. A cycle-specific cooling analysis will be performed to verify that the spent fuel pool cooling system can maintain the pool temperature within allowable limits based on the one hundred forty-eight hour decay time. In the unlikely event that the analysis determines this time is not sufficient to maintain acceptable pool temperature, the analysis will determine the additional in core hold time required. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

---

[1] USAR Section 9.5.2
[2] USAR Section 14.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 50.67 for the accidents analyzed.

The spent fuel pool sweep system will be operated for the first month after reactor is shutdown for refueling during fuel handling and crane operations with loads over the pool. The potential consequences of a postulated fuel handling accident without the system are a very small fraction of the guidelines of 10 CFR Part 50.67 after one month decay of the spent fuel. Heavy loads greater than one fuel assembly are not allowed over the spent fuel.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A fuel handling accident in containment does not cause containment pressurization. One containment door in each personnel air lock can be closed following containment personnel evacuation and the containment ventilation and purge system has the capability to initiate automatic containment ventilation isolation to terminate a release path to the atmosphere.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.

Accident analysis assumes a charcoal adsorber efficiency of 90%.[3] To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 90%, this equates to a methyl iodide penetration of 10%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 5%. Thus, the acceptance criteria of 95% efficient will be used for the charcoal adsorbers.

Although committing to ASTM D3803-89, it was recognized that ASTM D3803-89 Standard references Military Standards MIL-F-51068D, Filter, Particulate High Efficiency, Fire Resistant, and MIL-F-51079A, Filter, Medium Fire Resistant, High Efficiency. These specifications have been revised and the latest revisions are, MIL-F-51068F and MIL-F-51079D. These revisions have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. ASME AG-1 is an acceptable substitution. Consequently, other referenced standards can be substituted if the new standard or methodology is shown to provide equivalent or superior performance to those referenced in ASTM D3803-89.

---

[3] USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

TABLE TS 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety injection[1]	≤ 4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment spray b. Steam line isolation of both lines	≤ 23 psig ≤ 17 psig
3	Pressurizer Low Pressure	Safety injection[1]	≥ 1815 psig
4	Low Steam Line Pressure	Safety injection[1] Lead time constant Lag time constant	≥ 500 psig ≥ 12 seconds ≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and "Lo-Lo" T_{avg}	Steam line isolation of affected line[2]	≤ d/p corresponding to 0.745 x 10^6 lb/hr at 1005 psig ≥ 540°F
6	High-High Steam Flow in a Steam Line Coincident with Safety Injection	Steam line isolation of affected line[2]	≤ d/p corresponding to 4.4 x 10^6 lb/hr at 735 psig
7	Forebay Level	Trip circ. water pumps	

[1] Initiates containment isolation, feedwater line isolation, shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.

[2] Confirm main steam isolation valves closure within 5 seconds when tested. d/p = differential pressure

BASIS

System Tests (TS 4.5.a)

The Safety Injection System and the Containment Vessel Internal Spray System are principal plant safety systems that are normally in standby during reactor operation. Complete system tests cannot be performed when the reactor is OPERATING because a safety injection signal causes containment isolation, and a Containment Vessel Internal Spray System test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore to combine system tests to be performed during periodic shutdowns with more frequent component tests, which can be performed during reactor operation.

The system tests demonstrate proper automatic operation of the Safety Injection and Containment Vessel Internal Spray Systems. A test signal is applied to initiate automatic action, resulting in verification that the components received the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[(1)]

The Internal Containment Spray (ICS) System is designed to provide containment cooling in the event of a loss-of-coolant accident or steam line break accident, thereby ensuring the containment pressure does not exceed its design value of 46 psig at 268°F (100% R.H.).[(2)] With the KNPP ICS system design, 76 properly functioning spray nozzles per train will adequately provide the required ICS flow rate for post accident cooling.

Component Tests - Containment Fancoil Units (TS 4.5.a.3)

Testing of the containment fancoil unit emergency discharge and backdraft dampers is performed to assure the integrity of the duct work post-LOCA.

Component Tests - Pumps (TS 4.5.b.1)

During reactor operation, the instrumentation which is depended upon to initiate safety injection and containment spray is checked daily and the initiating logic circuits are tested monthly (in accordance with TS 4.1). In addition, the active components (pumps and valves) are to be tested quarterly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The quarterly test interval is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

(1) USAR Section 6.2
(2) USAR Section 6.4

Table TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY
3. Refueling Water Storage Tank Water Sample[7]	Boron Concentration	Monthly[8]
4. Deleted		
5. Accumulator	Boron Concentration	Monthly
6. Spent Fuel Pool	Boron Concentration	Monthly[9]
7. Secondary Coolant	a. Gross Beta or Gamma Activity b. Iodine Concentration	Weekly Weekly when gross beta or gamma activity ≥ 0.1 μCi/gram

[7] A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during REFUELING outages.

[8] And after adjusting tank contents.

[9] Sample will be taken monthly when fuel is in the pool.