TREAT AS SENSITIVE INFORMATION

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

NUCLEAR MANAGEMENT COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

FACILITY OPERATING LICENSE AS AMENDED

License No. DPR-43

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license filed by Wisconsin Public Service Corporation and Wisconsin Power and Light Company (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Kewaunee Nuclear Power Plant (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-50, as amended, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The Nuclear Management Company, LLC (NMC) is technically qualified and the licensees are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensees and NMC have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-43, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

I. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 70, including 10 CFR Section 30.33, 70.23 and 70.31.

2. Facility Operating License No. DPR-43 is hereby issued to NMC, Wisconsin Public Service Corporation and Wisconsin Power and Light Company, to read as follows:

A. This license applies to the Kewaunee Nuclear Power Plant, a pressurized water nuclear reactor and associated equipment (the facility), owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company. The facility is located in Kewaunee County, Wisconsin, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 7 through 31) and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Wisconsin Public Service Corporation and Wisconsin Power and Light Company to possess, and the NMC to use and operate the facility at the designated location in Kewaunee County, Wisconsin, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess, and use at any time special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report as supplemented and amended;

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

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

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

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. 176 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.

(6) Steam Generator Upper Lateral Supports

The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.

(7) License Transfer

(A) WPSC shall take all necessary steps to ensure that the decommissioning trusts are maintained in accordance with the application for approval of the transfer of MG&E's ownership interest in KNPP to WPSC and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order. Additionally, if the MG&E nonqualified fund is not transferred to WPSC, WPSC, or NMC acting on WPSC's behalf, shall explicitly include the status of the MG&E nonqualified fund in all future decommissioning funding status reports that WPSC, or NMC, submit in accordance with 10 CFR 50.75(f)(1).

(B) On the closing date of the transfer of MG&E's interests in KNPP to WPSC, MG&E shall transfer to WPSC all of MG&E's accumulated qualified decommissioning trust funds for KNPP. Immediately following such transfer, the amounts for radiological decommissioning of KNPP in WPSC's decommissioning trusts must, with respect to the interests in KNPP that WPSC would then hold, be at a level no less than the formula amounts under 10 CFR Section 50.75.

D. The NMC shall comply with applicable effluent limitations and other limitations and monitoring requirements, if any, specified pursuant to Section 401(d) of the Federal Water Pollution Control Act Amendments of 1972.

E. This license is effective as of the date of issuance, and shall expire at midnight on December 21, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:

Appendices A and B - Technical Specifications

Date of Issuance: December 21, 1973

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.

BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig for all transients except the hypothetical RCCA Ejection accident, for which the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia) is applied. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

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[1] USAR Section 4
[2] USAR Section 4.3

BASIS - Limiting Safety System Settings - Protective Instrumentation (TS 2.3)

Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.[1]

The power range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[2]

Two sustained rate protective trip functions have been incorporated in the Reactor PROTECTION SYSTEM. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more rod control cluster assemblies (RCCAs) fall into the core. The circuits are independent and ensure immediate reactor trip independent of the initial OPERATING state of the reactor. These trip functions are the LIMITING SAFETY SYSTEM actions employed in the accident analysis.

Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.[3] The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error [2] and transient level overshoot before the reactor trips.

Reactor Coolant Temperature

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: 1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2] is always below the core SAFETY LIMITS shown in the Core Operating Limits Report. If axial peaks are greater than design, as indicated by differences between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.

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[1] USAR Section 14.1.1
[2] USAR Section 14.0
[3] USAR Section 14.3.1

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.

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

BASIS

TS 3.0.a establishes the applicability statement within each individual TS as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the LCO is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a LCO are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the LCO must be met. This time limit is the allowable outage time to restore an inoperable system or component to the OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the TS no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a LCO is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual TSs may include a specified time limit for the completion of a surveillance requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new TS becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new TS becomes applicable if the requirements of the LCO are not met.

TS 3.0.b establishes that noncompliance with a TS exists when the requirements of the LCO are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this TS is to clarify that the implementation of the ACTION requirements within the specified time interval constitutes compliance with a TS. Completion of the remedial measures of the ACTION requirements is not required when compliance with a LCO is restored within the time interval specified in the associated ACTION requirements.

TS 3.0.c provides the standard shutdown sequence to be followed in the event a LIMITING CONDITION FOR OPERATION cannot be met and the condition is not specifically addressed by the associated action requirements. The purpose of this TS is to delineate the time limits for placing the unit in a safe shutdown mode allowed by the TSs.

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

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[1] USAR Section 7.2.2

The requirement for at least one train of residual heat removal when in the REFUELING MODE is to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel < 140°F. The requirement to have two trains of residual heat removal OPERABLE when there is < 23 feet of water above the reactor vessel flange ensures that a single failure will not result in complete loss-of-heat removal capabilities. With the reactor vessel head removed and at least 23 feet of water above the vessel flange, a large heat sink is available. In the event of a failure of the OPERABLE train, additional time is available to initiate alternate core cooling procedures.

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, then the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

Pressure Isolation Valves (TS 3.1.a.4)

The basis for the pressure isolation valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.[(2)]

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the Pressurizer Pressure Control System. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The pressurizer PORVs and associated block valves must be OPERABLE to provide an alternate means of mitigating a design basis steam generator tube rupture. Thus, an inoperable PORV (for reasons other than seat leakage) or block valve is not permitted in the HOT STANDBY and OPERATING MODES for periods of more than 72 hours.

[(2)] Order for Modification of License dated 4/20/81

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests[3] have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will ensure the ability to maintain pressurizer pressure. Surveillance tests are performed to ensure heater OPERABILITY.

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the High Point Vent System is to vent noncondensible gases from the high points of the RCS to ensure that core cooling during natural circulation will not be inhibited. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.[4]

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[3] Hot functional test (PT-RC-31)
[4] Letter from E. R. Mathews to S. A. Varga dated 5/21/82

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,[5] and the calculation methods of Footnote.[6] The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote.[7]

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

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[5] Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

[6] Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, astm designation E262-86.

[7] WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{Im} is the stress intensity factor due to membrane (pressure) stress. K_{It} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{It} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{IR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. The heatup and cooldown limit curves have been developed by combining the most conservative pressure temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data for each of the limiting materials. At any given temperature, the allowable pressure is taken to be the lesser of the values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments including the pressure difference between the gage and beltline weld.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations for each of the limiting materials. Composite limit curves are then constructed for each cooldown rate of interest. Again, adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan(8) and Footnote.(9)

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,(10) weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,(11) WCAP 9878,(12) WCAP-12020,(13) WCAP-14279,(14) and WCAP-14279, Revision 1(10) respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33[1] effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33[1] effective full-power years of fluence (through the end of OPERATING cycle 33[1]). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

<u>Note</u>:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

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(8) "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

(9) 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

(10) C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

(11) S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

(12) S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

(13) S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

(14) E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), then the vent path is considered secured in the open position.

Note

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

Maximum Coolant Activity (TS 3.1.c)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on maximum 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 Reactor Coolant Specific Activity is limited to ≤ 1.0 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[15] are analyzed assuming an RCS activity of 1.0 μCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of 60 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.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 50.67 limits.

The action of reducing average reactor coolant temperature to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

[15] USAR Section 14.0

Leakage of Reactor Coolant (TS 3.1.d)[16]

TS (TS 3.1.d.1)

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

If the reactor coolant activity is $91/\bar{E}$ µCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average X/Q = 2.0×10^{-6} sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

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

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

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

TS 3.1.d.2

Limiting the leakage through any single steam generator to 150 gpd ensures that tube integrity is maintained during a design basis main steam line break or loss-of-coolant accident. Remaining within this leakage rate provides reasonable assurance that no single tube-flaw will sufficiently enlarge to create a steam generator tube rupture as a result of stresses caused by a Loss of Coolant Accident (LOCA) or a main steam line break accident within the time allowed for detection of the accident condition and resulting commencement of plant shutdown. This operational leakage rate is less than the condition assumed in design basis safety analyses and conforms to industry standards established by the Nuclear Energy Institute through its NEI 97-06, "Generic Steam Generator Program Guidelines."

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[16] USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.3

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

TS 3.1.d.4

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

TS 3.1.d.5

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

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

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

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

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

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

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

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is ensured under all OPERATING conditions. [17]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.[18] Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

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

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at ≤ 60% RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.[19]

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in the COLR precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[19]

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

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[17] USAR Section 4.2
[18] USAR Section 9.2
[19] USAR Section 3.2.1

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the value specified in the COLR has been imposed to prevent any unexpected power excursion during normal operation as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than the value specified in the COLR provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special OPERATING precautions will be taken. In addition, the strong negative Doppler coefficient[20] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

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

Analysis has shown that maintaining the moderator temperature coefficient at criticality less than or equal to the value specified in the COLR will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports OPERATING up to 60% power with a moderator temperature coefficient less than or equal to the value specified in the COLR. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than the value specified in the COLR for at least 95% of a cycle's time at HFP to ensure the limitations associated with and anticipated transient without scram (ATWS) event are not exceeded. NRC approved methods[20][21] will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than the value specified in the COLR, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

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(20) USAR Section 3.2.1

(20) "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

(21) "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

In the event that the limits as provided in the COLR are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit as provided in the COLR. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits as provided in the COLR are not developed, then the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits as specified in the COLR. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

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. Also, the Safety Injection pumps can take a suction from the Refueling Water Storage Tank and provide borated water to the Reactor Coolant System.

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

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.

TS 3.3.b.5 provides protection from the possibility of one SI pump reaching runout condition during SI accumulator fill concurrent with a large break LOCA. With both trains of SI and both EDGs operable, the SI system will meet accident analysis.

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, either 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.

The spray additive system may be inoperable for up to 72 hours. The containment spray system would still be available and would remove some iodine from the containment atmosphere in the event of a Design Basis Accident. The 72-hour completion time takes into account the containment spray system capabilities and the low probability of the worst case Design Basis Accident occurring during this period.

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.[8]

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. [9] 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 and Valve SI-4A, etc.). Shared piping and valves are considered to be common to both trains of the systems.

[8] USAR Section 9.3
[9] USAR Section 9.6

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-sink-event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

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

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 $\leq$ 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.

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

BASIS - Instrumentation System (TS 3.5)

Instrumentation has been provided to sense accident conditions and to initiate operation of the engineered safety features.[1] Section 2.3 of these specifications describes the LIMITING SAFETY SYSTEM SETTINGS for the protective instrumentation.

Safety Injection

Safety Injection can be activated automatically or manually to provide additional water to the Reactor Coolant System or to increase the concentration of boron in the coolant.

Safety Injection is initiated automatically by (1) low pressurizer pressure, (2) low main steam line pressure in either loop and (3) high containment pressure. Protection against a loss-of-coolant accident is primarily through signals (1) and (3). Protection against a steam line break is primarily by means of signal (2).

Manual actuation is always possible. Safety Injection signals can be blocked during those OPERATING MODES where they are not "required" for safety and where their presence might inhibit operating flexibility; they are generally restored automatically on return to the "required" OPERATING MODE.

Reactor Trip Breakers

With the addition of the automatic actuation of the shunt trip attachment, diverse features exist to effect a reactor trip for each reactor trip breaker. Since either trip feature being OPERABLE would initiate a reactor trip on demand, the flexibility is provided to allow plant operation on a reactor trip breaker (with either trip feature inoperable) for up to 72 hours. This specification also requires the plant to proceed to the HOT SHUTDOWN condition in accordance with the Kewaunee STANDARD SHUTDOWN SEQUENCE if a reactor trip breaker is bypassed for greater than 8 hours.

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is Safety Injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high containment pressure (Hi-Hi) sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

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[1] USAR Section 7.5

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of Safety Injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident.

Steam Line Isolation

In the event of a steam line break, the steam line isolation valve of the affected line is automatically isolated to prevent continuous, uncontrolled steam release from more than one steam generator. The steam lines are isolated on Hi-Hi containment pressure or high steam flow in coincidence with Lo-Lo T_{avg} and Safety Injection or Hi-Hi steam flow in coincidence with Safety Injection. Adequate protection is afforded for breaks inside or outside the containment even under the assumption that the steam line check valves do not function properly.

Main Feedwater Isolation

Main feedwater isolation actuation occurs as a result of a Hi-Hi steam generator water level to prevent steam generator overfill conditions. Steam generator overfill may result in damage to secondary components; for example, high moisture steam could erode the turbine blades at an accelerated rate.

Setting Limits

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety Injection protects against loss-of-coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal containment pressure for initiation of containment spray and at about 40% for initiation of steam line isolation. Initiation of containment spray and steam line isolation protects against large loss-of-coolant or steam line break accidents as discussed in the safety analysis.

3. The pressurizer low-pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.

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[2] USAR Section 14.3
[3] USAR Section 14.2.5

4. The steam line low-pressure signal is lead/lag compensated and its setpoint is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis.

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the Hi-Hi steam line flow limit is set at approximately 120% of nominal full-load flow at the full-load pressure in order to protect against large steam line break accidents. The coincident Lo-Lo T_{avg} setting limit for steam line isolation initiation is set below its HOT SHUTDOWN value. The safety analysis shows that these settings provide protection in the event of a large steam line break.

6. The setpoints and associated ranges for the undervoltage relays have been established to always maintain motor voltages at or above 80% of their nameplate rating, to prevent prolonged operation of motors below 90% of their nameplate rating, and to prevent prolonged operation of 480 V MCC starter contactors at inrush currents.[4] All safeguard motors were designed to accelerate their loads to operating speed with 80% nameplate voltage, but not necessarily within their design temperature rise. Prolonged operation below 90% of nameplate voltage may result in shortening of motor insulation life, but short-term operation below 90% of nameplate voltage will not result in unacceptable effects due to the service factor provided in the motors and the conservative insulation system used on the motors. Prolonged operation of MCC contactors at inrush currents may result in blown control fuses and inoperable equipment; therefore operation will be limited to a time less than it takes for a fuse to blow.

   The primary safeguard buses undervoltage trip (85.0% of nominal bus voltage) is designed to protect against a loss of voltage to the safeguard bus and assures that safeguard protection action will proceed as assumed in the USAR. The associated time delay feature prevents inadvertent actuation of the undervoltage relays from voltage dips, while assuring that the diesel generators will reach full capacity before the Safety Injection pump loads are sequenced on.

   The safeguard buses second level undervoltage trip (93.6% nominal bus voltage) is designed to protect against prolonged operation below 90% of nameplate voltage of safeguard pumps. The time delay of less than 7.4 seconds ensures that engineered safeguards equipment operates within the time delay assumptions of the accident analyses. The time delay will prevent blown control fuses in 480 V MCC's; the MCC control fuses are the limiting component for long-term low voltage operation. The time delay is long enough to prevent inadvertent actuation of the second level UV relays from voltage dips due to large motor starts (except reactor

[4] USAR section 8.2.3

coolant pump starts with a safeguards bus below 3980 volts). Up to 7.4 seconds of operation of safeguard pumps between 80% and 90% of nameplate voltage is acceptable due to the service factor and conservative insulation designed into the motors.

Each relay in the undervoltage protection channels will fail safe and is alarmed to alert the operator to the failure.

A blackout signal which occurs during the sequence loading following a Safety Injection signal will result in a re-initiation of the sequence loading logic at time step 0 as long as the Safety Injection signal has not been reset. The Kewaunee Emergency Procedures warn the operators that a Blackout Signal occurring after reset of Safety Injection will not actuate the sequence loading and instructs to re-initiate Safety Injection if needed.

Instrument OPERATING Conditions

During plant OPERATIONS, the complete protective instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection Systems, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing OPERATION with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines LIMITING CONDITIONS FOR OPERATION necessary to preserve the effectiveness of the Reactor Control and PROTECTION SYSTEM when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for CHANNEL CALIBRATION and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in another channel.

The OPERABILITY of the instrumentation noted in Table TS 3.5-6 assures that sufficient information is available on these selected plant parameters to aid the operator in identification of an accident and assessment of plant conditions during and following an accident. In the event the instrumentation noted in Table TS 3.5-6 is not OPERABLE, the operator is given instruction on compensatory actions.

BASIS

Containment System Integrity (TS 3.6.a)

The COLD SHUTDOWN condition precludes any energy releases or buildup of containment pressure from flashing of reactor coolant in the event of a system break. The restriction to fuel that has been irradiated during power operation allows initial testing with an open containment when negligible activity exists. The shutdown margin for the COLD SHUTDOWN condition assures subcriticality with the vessel closed even if the most reactive RCC assembly were inadvertently withdrawn. Therefore, the two parts of TS 3.6.a allow CONTAINMENT SYSTEM INTEGRITY to be violated when a fission product inventory is present only under circumstances that preclude both criticality and release of stored energy.

When the reactor vessel head is removed with the CONTAINMENT SYSTEM INTEGRITY violated, the reactor must not only be in the COLD SHUTDOWN condition, but also in the REFUELING shutdown condition. A 5% shutdown margin is specified for REFUELING conditions to prevent the occurrence of criticality under any circumstances, even when fuel is being moved during REFUELING operations.

This specification also prevents positive insertion of reactivity whenever Containment System integrity is not maintained if such addition would violate the respective shutdown margins. Effectively, the boron concentration must be maintained at a predicted concentration of 2,200 ppm[(1)] or more if the Containment System is to be disabled with the reactor pressure vessel open.

Containment Isolation Valves (TS 3.6.b)

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

To be considered OPERABLE, automatic containment isolation valves are required to close within prescribed time limits and to actuate on an automatic isolation signal. Check valves are considered OPERABLE when they have satisfactorily completed their required surveillance testing. Manual isolation components are considered OPERABLE when manual valves are closed, blind flanges are in place, and closed systems are intact.

Penetration flow path(s) may be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in

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[(1)] USAR Table 3.2-1

continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Specification TS 3.6.b.2 pertains to inoperable valves described in TS 3.6.b.3, manual valves assumed to be closed, and normally closed valves that are not assumed, by the USAR, to automatically close. This allows opening of containment isolation valves without entering the LCO or to open containment isolation valves closed as required by TS, provided the administrative controls are in place to ensure valve closure, if needed.

For these LCO(s), separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

In the event a containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated within the specified time constraints. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are 1) a closed and de-activated automatic containment isolation valve, 2) a closed manual valve, 3) a blind flange, and 4) a check valve with flow through the valve secured. For a penetration flow path isolated, the device used to isolate the penetration should be the closest available one to containment. The 24-hour completion time is reasonable, considering the time required to isolate the penetration, perform maintenance, and the relative importance of supporting containment OPERABILITY.

For affected containment penetration flow paths that cannot be restored to OPERABLE status within the required completion time and that have been isolated, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure containment penetrations, requiring isolation following an accident and no longer capable of being automatically isolated, will be in that isolated position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period is specified as "prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN if not performed within the previous 92 days." This is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1-hour Completion Time is consistent with the ACTIONS of LCO 3.0.c. In the event the affected penetration is isolated, the affected penetration must be verified to be isolated on a periodic basis which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of "once per 31 days for verifying each affected penetration flow path is isolated" is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

For those penetrations where one of the isolation devices is a closed system, either inside containment or outside containment, a longer outage time is allowed. This condition is only applicable to those penetration flow paths with a single containment isolation valve and a closed system. This longer outage time is due to a closed system subjected to leakage testing, missile protected, and seismic category I piping. Also, a closed system typically has flow through it during normal operation such that any loss of integrity could be observed through leakage detection system inside containment and system walkdowns outside containment. Thus, a 72-hour completion time is considered appropriate given that certain valves may be located inside containment and the reliability of the closed system.

Isolation devices located in high radiation areas shall be verified closed by use of administrative means. Verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position is small.

De-activation of an automatic containment isolation valve is accomplished by removing or interrupting the valves motive force, thus, preventing a change in the valve position by a single active failure. De-activation may be accomplished by opening the supply breaker for a motor operated valve, isolating air to an air operated valve, removing the supply fuse for a solenoid operated valve, or any other means for ensuring the isolation barrier cannot be affected by a single active failure.

Ventilation Systems (TS 3.6.c)

Proper functioning of the Shield Building Ventilation System is essential to the performance of the Containment System. Therefore, except for reasonable periods of maintenance outage for one redundant train of equipment, the complete system should be in readiness whenever CONTAINMENT SYSTEM INTEGRITY is required. Proper functioning of the Auxiliary Building Special Ventilation System is similarly necessary to preclude possible unfiltered leakage through penetrations that enter the Special Ventilation Zone (Zone SV).

Both the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System are designed to automatically start following a safety injection signal. Each of the two trains of both systems has 100% capacity. If one train of either system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue while repairs are being made. If both trains of either system are inoperable, the plant will be brought to a condition where the air purification system would not be required.

High efficiency particulate air (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 release 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 iodine 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. The performance criteria for the safeguard ventilation fans are stated in Section 5.5 and 9.6 of the USAR. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

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

Accident analysis assumes a charcoal adsorber efficiency of 90%.[2] 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.

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

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 specification have been superseded. The latest versions, MIL-F-51068F and MIL-F-51079D, have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. This is an acceptable situation. 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.

Containment Pressure (TS 3.6.d)

The 2 psi limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident as discussed in Sections 14.2 and 14.3 of the USAR.[3]

The reactor containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

Containment Temperature (TS 3.6.e)

The requirement of a 40°F minimum containment ambient temperature is to assure that the minimum containment vessel metal temperature is well above NDTT + 30° criterion for the shell material.

[3] USAR Section 5

BASIS – Auxiliary Electrical Systems (TS 3.7)

The intent of this TS is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safety features equipment following an accident.

Plant safeguards auxiliary power is normally supplied by two separate external power sources which have multiple off-site network connections [1]: the reserve auxiliary transformer from the 138-Kv portion of the plant substation, and a tertiary winding on the substation auto transformer. Either source is sufficient to supply all necessary accident and post-accident load requirements from any one of four available transmission lines.

Each diesel generator is connected to one 4160-V safety features bus and has sufficient capacity to start sequentially and operate the engineered safety features equipment supplied by that bus. The set of safety features equipment items supplied by each bus is, alone, sufficient to maintain adequate cooling of the fuel and to maintain containment pressure within the design value in the event of a loss-of-coolant accident.

Each diesel generator starts automatically upon low voltage on its associated bus, and both diesel generators start in the event of a safety injection signal.[2] A minimum of 7 days fuel supply for one diesel generator is maintained by requiring 36,000 gallons of fuel oil, thus assuring adequate time to restore off-site power or to replenish fuel. The diesel fuel oil storage capacity requirements are consistent with those specified in ANSI N195-1976/ANS-59.51, Sections 5.2, 5.4, and 6.1.

The plant safeguards 125-V d-c power is normally supplied by two batteries each of which will have a battery charger in service to maintain full charge and to assure adequate power for starting the diesel generators and supplying other emergency loads. A third charger is available to supply either battery.[3]

The arrangement of the auxiliary power sources and equipment and this TS ensure that no single fault condition will deactivate more than one redundant set of safety features equipment items and will therefore not result in failure of the plant protection systems to respond adequately to a loss-of-coolant accident.

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[1] USAR Figure 8.2-1 and 8.2-2
[2] USAR Section 8.2.3
[3] USAR Section 8.2.2 and 8.2.3

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.

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

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.

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.

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

BASIS-Control Rod and Power Distribution Limits (TS 3.10)

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 as specified in the COLR must be observed. In addition, for HOT SHUTDOWN conditions, the shutdown margin as specified in the COLR must be provided for protection against the steam line break accident.

Rod insertion limits are used to ensure adequate trip reactivity, to ensure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident.

The exception to the rod insertion limits in TS 3.10.d.3 is to allow the measurement of the worth of all rods. This measurement is a part of the Reactor Physics Test Program performed at the startup of each cycle. Rod worth measurements 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.

TS 1.0.r, "Shutdown Margin," states the definition of shutdown margin as used in the technical specifications. As a part of this definition is a statement which removes the assumption that the highest reactivity worth rod cluster control assembly (RCCA) is fully withdrawn. This includes the verification that all RCCA's are fully inserted by two independent means. Although not fully independent, this requirement refers to indications which are independent. These independent means include such indicators as the control board individual rod position indicators or the rod position as indicated on the plant process computer system (PPCS) or the condition monitors referenced in TS 3.10.e.

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 analyses. The peak linear power density is chosen to ensure peak clad temperature during a postulated large break loss-of-coolant accident is less than the 2200° F limit. Second, the minimum DNBR in the core must not be less than the DNBR limit in normal operation or during Condition I or II transient events.

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

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

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR 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, with a high probability, remain less than the 2200° F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. The LOCA analyses are performed for Westinghouse 422 V+ fuel, FRA-ANP heavy fuel and for FRA-ANP standard fuel.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent 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.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

$F_Q^{EQ}(Z)$ is the measured $F_Q^N(Z)$ obtained at equilibrium conditions during the target flux determination. $F_Q^{EQ}(Z)$ must satisfy the relationship that is in the COLR.

Because the value of $F_Q^N(Z)$ represents an equilibrium condition, it does not include the variations of $F_Q^N(Z)$ which are present during non-equilibrium situations such as load following or power ascension. To account for these possible variations, the equilibrium value of $F_Q^N(Z)$ is adjusted by an elevation dependent factor, W(z), that accounts for the calculated worst case transient conditions. Core power distribution is controlled under non-equilibrium conditions by operating the core within the core operating limits on axial flux distribution, quadrant power tilt, and control rod insertion.

If a power distribution measurement indicates that the $F_Q^{EQ}(Z)$ transient relationship's margin to the limit has decreased since the previous evaluation then TS 3.10.b.6.C provides two options of either increasing the $F_Q^{EQ}(Z)$ transient relationship by the appropriate penalty factor or increasing the power distribution surveillance to once every 7 EFPD until two successive flux maps indicate that the $F_Q^{EQ}(Z)$ transient relationship's margin to the limit has not decreased. IF $F_Q^{EQ}(Z)$ with the penalty factor applied is greater than the limit, then TS 3.10.b.6 is not satisfied and TS 3.10.b.7 should be applied to maintain the normal surveillance interval. Based on TS 3.10.b.7.A, the axial flux distribution (AFD) limits are reduced by 1% for each 1% that the $F_Q^{EQ}(Z)$ transient relationship exceeds its limit within the allowed time of 4 hours.

The contingency actions of TS 3.10.b.6 and TS 3.10.b.7 are to ensure that $F_Q^N(Z)$ does not exceed its limit for any significant period of time without detection. Satisfying limits on $F_Q^N(Z)$ ensures that the safety analyses remain bounding and valid.

$F_{\Delta H}^{N}$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^{N}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^{N}$ is based on an integral and is used as such in DNBR 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 $F_{\Delta H}^{N}$.

The $F_{\Delta H}^{N}$ limit is determined from safety analyses of the limiting DNBR transient events. The safety analyses are performed for FRA-ANP heavy fuel, FRA-ANP standard fuel, and Westinghouse 422 V+ fuel. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is greater than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^{N}$ limit.

The use of $F_{\Delta H}^{N}$ in TS 3.10.b.5.C 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 needs to be included in TS 3.10.b.1 for FRA-ANP fuel.[1]

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:

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 specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and reactor power is greater than 50 percent or RATED POWER.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit 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.

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[1] 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.

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 two 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 two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core power distribution 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. Power distribution 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, then the reactor power level will be reduced to ≤50%.

If a misaligned rod has caused a tilt ratio > 1.09, then the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0. If after eight hours the rod has not been realigned, then 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, then 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, then the generator may remain connected to the grid.

If the tilt ratio is > 1.09, and it is not due to a misaligned rod, then 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 Insertion Limits (TS 3.10.d)

The allowed completion time of two hours for restoring the control banks to within the insertion limits provides an acceptable time for evaluation and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Operation beyond the rod insertion limits is allowed for a short-time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss-of-flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The time limits of six hours to achieve HOT STANDBY and an additional six hours to achieve HOT SHUTDOWN allow for a safe and orderly shutdown sequence and are consistent with most of the remainder of the Technical Specifications.

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 (IRPI), 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, then 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.

Core Average Temperature (TS 3.10.k)

The core average temperature limit is consistent with full power operation within the nominal operational envelope. Either Tavg control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A higher Tavg will cause the reactor core to approach DNB limits.

Reactor Coolant System Pressure (TS 3.10.l)

The RCS pressure limit is consistent with operation within the nominal operational envelope. Either pressurizer pressure control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A lower pressure will cause the reactor core to approach DNB limits.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant system (RCS) flow limit, as specified in the COLR, is consistent with the minimum RCS flow limit assumed in the safety analysis adjusted by the measurement uncertainty. The safety analysis assumes initial conditions for plant parameters within the normal steady state envelope. The limits placed on the RCS pressure, temperature, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed transients.

The RCS flow normally remains constant during an operational fuel cycle with all reactor coolant pumps running. At least two plant computer readouts from the loop RCS flow instrument channels are averaged per reactor coolant loop and the sum of the reactor coolant loop flows are compared to the limit. Operating within this limit will result in meeting the DNBR criterion in the event of a DNB-limited event.

DNBR Parameters (TS 3.10.n)

The DNBR related safety analyses make assumptions on reactor temperature, pressure, and flow. In the event one of these parameters does not meet the TS 3.10.k, TS 3.10.l or TS 3.10.m limits, an analysis can be performed to determine a power level at which the MDNBR limit is satisfied.

BASIS - Control Room Post-Accident Recirculation System (TS 3.12)

The Control Room Post-Accident Recirculation System is designed to filter the Control Room atmosphere during Control Room isolation conditions. The Control Room Post-Accident Recirculation System is designed to automatically start upon SIS or high radiation signal.

If the system is found to be inoperable, there is no immediate threat to the Control Room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 7 days, the reactor is placed in HOT STANDBY until the repairs are made.

Accident analysis assumes a charcoal adsorber efficiency of 90%.[1] 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.

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

BASIS - Shock Suppressors (Snubbers) (TS 3.14)

Shock suppressors (snubbers) are designed to prevent unrestrained pipe motion under dynamic loads, as might occur during seismic activity or severe plant transients, while allowing normal thermal motion during startup or shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic event or other events initiating dynamic loads. It is therefore required that all snubbers designed to protect the reactor coolant and other safety-related systems or components be operable during reactor operation. The intent of this TS is to prohibit startup or continued operation with defective safety-related shock suppressors.

Because the protection afforded by snubbers is required only during low probability events, TS 3.14.b allows a period of 72 hours for repairs or feasible alternative action before reactor shutdown is required.

BASIS - Surveillance Requirements (TS 4.0)

TS 4.0.a establishes the requirements that surveillances must be met during the operational MODES or other conditions for which the requirements of the LIMITING CONDITIONS FOR OPERATION (LCO) apply unless otherwise stated in an individual surveillance requirement. The purpose of this TS is to ensure that surveillances are performed to verify the OPERABILITY of systems and components and that parameters are within specified limits. This ensures safe operation of the facility when the plant is in a MODE or other specified condition for which the associated LCOs are applicable. Surveillance requirements do not have to be performed when the facility is in an operational MODE for which the requirements of the associated LCO do not apply unless otherwise specified. Surveillance requirements do not have to be performed on inoperable equipment because the action requirements define the remedial measures that apply. However, the surveillance requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

TS 4.0.b establishes the limit for which the specified time interval for surveillance requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operation conditions that may not be suitable for conducting the surveillance (e.g., transient conditions or other ongoing surveillance or maintenance activities). It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of TS 4.0.b is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the surveillance requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

TS 4.0.c establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the allowed surveillance interval. A delay period of up to 24 hours or up to the limit of the allowed surveillance interval, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed in accordance with TS 4.0.b, and not at the time that the allowed surveillance interval was not met.

This delay period provides adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with required actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements. When a surveillance with an allowed interval based not on time intervals, but upon specified unit conditions, OPERATING situations, or requirements of regulations (e.g., prior to entering OPERATING MODE after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, TS 4.0.c allows for the full delay period of up to the allowed surveillance interval to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

TS 4.0.c provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of MODE changes imposed by required actions.

Failure to comply with allowed surveillance intervals for SRs is expected to be an infrequent occurrence. Use of the delay period established by TS 4.0.c is flexibility which is not intended to be used as an operational convenience to extend surveillance intervals.

While up to 24 hours or the limit of the allowed interval is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the completion times of the required actions for applicable LCO conditions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this Specification, or within the completion time of the actions, restores compliance with TS 4.0.a.

TS 4.0.d establishes the requirements that all applicable surveillance must be met before entry into an operational MODE or other condition of operation specified in the applicability statement. The purpose of the TS is to ensure that system and component operability requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of the TS, the applicable surveillance requirements must be performed within the specified surveillance interval to ensure that the LCOs are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with action requirements, the provisions of TS 4.0.d do not apply because this would delay placing the facility in a lower MODE of operation.

BASIS - Operational Safety Review (TS 4.1)

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, the minimum checking frequency of once per shift is regarded as adequate for reactor and steam system instrumentation when the plant is in operation.

Calibration

Calibration shall be performed to ensure the accuracy of information presented.

The nuclear flux (linear level) channels shall be calibrated at least daily against a heat balance standard to verify drift and effects of changing rod patterns.

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at each REFUELING shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each REFUELING shutdown for the process system channels is considered acceptable.

Testing

Experience with this type of instrumentation has shown that the testing frequency as specified in Table TS 4.1-1 will assure the required level of performance.

Fuel Inspection

Two fuel assemblies per region will be selected as reference assemblies on which base line data will be taken prior to initial fuel loading. During each refueling, visual inspections will be made on a representative sample of assemblies and in addition on any suspect assembly. Any observed unexplained anomalies in the suspected assembly will determine the necessity to recheck the reference assemblies against the original base line data.

Seismic

The seismic instrumentation will be checked for proper operation once per operating cycle or once every 18 months, whichever occurs first. In the event of a seismic disturbance, written administrative procedures will be put into effect covering operation of the plant. Inspection of crucial areas and components will be made immediately with the results of this inspection documented. In the absence of any unusual observations the plant will continue to be operated.

Guard Pipes

Visual inspections will be made of the accessible portions of the hot process pipeline guard pipes once during each operating cycle or once every 18 months, whichever occurs first.

BASIS

Kewaunee Nuclear Power Plant design was not designed to Section XI of the ASME Code; therefore, 100% compliance may not be practically achievable. However, the design process did consider access for in-service inspection, and made modifications within design limitations to provide maximum access. To the extent practical, NMC performs inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components in accordance with Section XI of the ASME Code. If an inspection required by the Code is impractical, NMC requests Commission approval for deviation from the requirement.

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 inspection and plugging requirements for Kewaunee Nuclear Power Plant steam generator tubes. Fulfilling these requirements assures that KNPP steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria of 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 the reactor coolant pressure boundary to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires the Reactor Coolant System and associated auxiliary, control, and protection systems to be designed with sufficient margin to ensure that design limits of the reactor coolant pressure boundary are not exceeded during normal operation, including during anticipated operational transients. Furthermore, GDC 32, "Inspection of Reactor Coolant System Pressure Boundary," requires components that are part of the reactor coolant pressure boundary to be designed to permit periodic inspection and testing of critical areas in order to assess their structural and leak tight integrity.

The NRC has developed guidance for steam generator tube inspection 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 before revising them. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines steam generator tube minimum wall thickness.

Technical Specification 4.2.b.1

If the steam generators are performing in an adequately similar manner, it is appropriate to limit the inspection to one steam generator per inspection interval on an alternating basis. This offers economic savings as well as reduction of radiation exposure and outage duration.

Technical Specification 4.2.b.2

Inspection of the steam generator tubes provides evaluation of their service condition. Operational experience has shown that certain types of steam generators are susceptible to generic degradation mechanisms. It has also revealed site-specific steam generator tube degradation mechanisms. The Kewaunee inspection program assesses both generic and site-specific tube degradations.

Kewaunee uses various eddy current (EC) testing methodologies to inspect steam generator tubes. EC technology has improved considerably since Kewaunee began commercial operation in 1974, and NMC is committed to use advanced EC methods and technology, as appropriate, to assure accurate assessment of steam generator tube service condition.

Technical Specification 4.2.b.3

Kewaunee Nuclear Power Plant steam generator tube inspections are typically conducted during refueling outages. Criteria used to select tubes for inspection are based, in part, on tube service condition determined during previous inspections, and on operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tubes results in expansion of the current inspection as well as increased frequency of subsequent inspections. In this manner, steam generator tube surveillance remains consistent with tube service condition.

Several operational events or transients require consequent steam generator tube inspections. These inspections must be performed after occurrence of excessive primary-to-secondary leakage or after transients that impose large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

Procedures, calculations, and analyses found in WCAP-15325,[1] combined with conservative allowances, such as general corrosion and measurement error, are the bases for the tube plugging criteria set forth in TS 4.2.b.4. Tubes that exceed the limits established by these criteria must be removed from service by plugging.

Steam generator tube plugging is a common method of preventing excessive primary-to-secondary steam generator tube leakage. This method is relatively uncomplicated and isolates a defective tube from the reactor coolant system by installing mechanical devices to block its hot and cold leg tubesheet openings.

Technical Specification 4.2.b.5 (Deleted)

Technical Specification 4.2.b.6 (Deleted)

Technical Specification 4.2.b.7

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

[1] WCAP 15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Replacement Steam Generators."

BASIS

Background - Containment Tests (TS 4.4)

The Containment System is designed to provide protection for the public from the consequences of a Design Basis Accident.[1] The Design Basis Accident is an instantaneous double-ended rupture of the cold leg of the Reactor Coolant System. Pressure and temperature behavior subsequent to the accident was determined by calculations evaluating the combined influence of the energy sources, the heat sinks and engineered safety features. The assumptions and effects for containment vessel leakage rate are detailed in the USAR[2] and further amplified in one of its Appendices.[3]

The total containment system consists of two systems. The Primary Containment System consists of a steel structure and its associated engineered safety features systems. The Primary Containment System, also referred to as the Reactor Containment Vessel, is a low-leakage steel shell, including all of its penetrations, designed to confine the radioactive materials that could be released by accidental loss of integrity of the Reactor Coolant System pressure boundary. It is designed for a maximum internal/test pressure of 46 psig and a temperature of 268°F.

The Secondary Containment System consists of the Shield Building, its associated engineered safety features systems, and a Special Ventilation Zone in the Auxiliary Building. The Shield Building is a medium-leakage concrete structure surrounding the Reactor Containment Vessel and is designed to provide a means for collection and filtration of fission-product leakage from the Reactor Containment Vessel following the Design Basis Accident. A 5-ft. annular space is provided between the Reactor Containment Vessel and the Shield Building. The Shield Building Ventilation System is the engineered safety feature utilized for the collection and filtration of fission-product leakage from the containment vessel.

The Special Ventilation Zone of the Auxiliary Building provides a medium-leakage boundary which confines leakage that could conceivably bypass the Shield Building annulus. The safety system associated with the Auxiliary Building Special Ventilation Zone is the Auxiliary Building Special Ventilation System (ABSVS). One of the functions of the ABSVS is to collect and filter any potential fission products that may bypass the Shield Building annulus.

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[1] USAR Section 14.3
[2] USAR Section 14.3.5
[3] USAR Appendix H

Maintaining CONTAINMENT SYSTEM INTEGRITY in an OPERABLE state requires, among other conditions, that all the requirements of TS 4.4.a and b, leakage rate testing (Containment Leakage Rate Testing Program), are satisfied. The testing process will include: (1) an overall containment leak rate evaluation (Type A); (2) a determination of the leakage through pressure containing or leakage limiting boundaries (Type B); and (3) an evaluation of the leak rate through containment isolation valves (Type C).[4] These tests are intended to check all possible paths for containment atmosphere to reach the outside atmosphere.

Shield Building Ventilation System (TS 4.4.c)

Pressure drop across the combined HEPA filters and charcoal adsorbers of < 10 inches of water and an individual HEPA bank pressure drop of < 4 inches of water at the system design flow rate (±10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle establishes system performance capability. This pressure drop is approximately 3 inches of water when the filters are clean.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 (Rev. 1) dated July 1976, except that ASTM D3803-89 standard will be used to fulfill the guidelines of Table 2, item 5, "Radioiodine removal efficiency." The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated July 1976.

If painting, fire, or chemical release occurs, the charcoal adsorber will be laboratory tested to determine whether it was contaminated from the fumes, chemicals, or foreign materials. Replacement of the charcoal adsorber can then be evaluated.

Operation of the systems every month will demonstrate operability of the filters and adsorber system. Operation of the Shield Building Ventilation System will result in a discharge to the environment. This discharge is made after at least three samples of the building atmosphere have been analyzed to determine the concentration of activity in the atmosphere.

[4] 10 CFR Part 50, Appendix J, Option B

Auxiliary Building Special Ventilation System (TS 4.4.d)

Demonstration of the automatic initiation capability is necessary to assure system performance capability.[5]

Pressure drop across the combined HEPA filters and charcoal adsorbers of < 10 inches of water and an individual HEPA bank pressure drop of < 4 inches of water at the system design flow rate (±10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle establishes system performance capability. This pressure drop is approximately 3 inches of water when the filters are clean.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 (Rev. 1) dated July 1976, except that ASTM D3803-89 standard will be used to fulfill the guidelines of Table 2, item 5, "Radioiodine removal efficiency." The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated July 1976.

If painting, fire, or chemical release occurs, the charcoal adsorber will be laboratory tested to determine whether it was contaminated from the fumes, chemicals, or foreign materials. Replacement of the charcoal adsorber can then be evaluated.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

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

Vacuum Breaker Valves (TS 4.4.e)

The vacuum breaker valves are 18 inch butterfly valves with air to open, spring to close operators. The valve discs are center pivot and rotate when closing to an EPT base material seat. When closed, the disc is positioned fully on the seat regardless of flow or pressure direction. Testing these valves in a direction opposite to that which would occur post-LOCA verifies leakage rates of both the vacuum breaker valves and the check valves downstream.

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[5] USAR Section 9.6

Isolation Device Positions (TS 4.4.f)

TS 4.4.f.1 ensures each 36 inch containment purge valve is verified sealed closed at 31-day intervals.[6] This Surveillance is designed to ensure that an inadvertent or spurious opening of a containment purge valve does not cause a gross breach of containment. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit off-site doses. Therefore, these valves are required to be in the sealed closed position when critical. A containment purge valve that is sealed closed must be closed with its control switch sealed in the close position. In this application, the term "sealed" has no connotation of leak tightness. The frequency is a result of a NRC initiative, Generic Issue B-24, related to containment purge valve use during plant operations.

TS 4.4.f.2 ensures the 2-inch vent/purge valves are closed as required or, if open, open for an allowable reason. If a 2-inch vent/purge valve is open in violation of this TS, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The TS is not required to be met when the 2-inch vent/purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The 2-inch vent/purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day frequency is consistent with other containment isolation valve requirements discussed.

TS 4.4.f.3.A requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The TS helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary are within design limits. This TS does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The TS specifies that containment isolation valves that are open under administrative controls are not required to meet the TS during the time the valves are open. This TS does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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[6] Letter from Steven A. Varga (NRC) to C.W. Giesler (WPSC) dated April 22, 1983

TS 4.4.f.3.B requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions, is closed. The TS helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the frequency of "prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The TS specifies that containment isolation valves that are open under administrative controls are not required to meet the TS during the time they are open. This TS does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

TS 4.4.f.3.C modifies TS 4.4.f.3 for valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted when above COLD SHUTDOWN for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

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.

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

BASIS – Periodic Testing of Emergency Power Systems (TS 4.6)

Each diesel generator can start and be ready to accept full load within 10 seconds, and will sequentially start and supply the power requirements for one complete set of engineered safety features equipment in approximately one minute.[1] This test will be conducted during each REFUELING outage to ensure that the diesel generator will start and assume required loads in accordance with the timing sequence listed in USAR Table 8.2-1 after the initial starting sequence.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency will be detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function. Its possible failure to respond is, of course, anticipated by providing two diesel generators, each supplying through an independent bus, a complete and adequate set of engineered safety features equipment. Further, both diesel generators are provided as backup to multiple sources of external power, and this multiplicity of sources should be considered with regard to adequacy of test frequency.

Monthly Diesel Generator Surveillance (TS 4.6.a.1)

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be indicated by an alarm without need for test startup. Monthly tests are performed in accordance with the intent of IEEE 387-1977, paragraph 6.6.1.

REFUELING Interval Diesel Generator Surveillance (TS 4.6.a.2)

The REFUELING interval diesel generator surveillance demonstrates that the Emergency Power System, and its control system, will function automatically to provide engineered safety equipment power in the event of loss of off-site power coincident with a safety injection signal. This test demonstrates proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment to demonstrate OPERABILITY of the diesel generators. This test is initiated by simultaneously unblocking safety injection and simulating a loss-of-voltage signal. This surveillance is performed to meet the intent of IEEE 387-1977 paragraph 6.6.2. (Note also that Reg. Guide 1.108 addresses diesel generator surveillance.)

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[1] USAR Section 8.2

REFUELING Interval Diesel Generator Inspection, TS 4.6.a.3

Inspections are performed at REFUELING outage intervals in order to maintain the diesel generators in accordance with the manufacturers' recommendations. The inspection procedure is periodically updated to reflect experience gained from past inspections and new information as it is available from the manufacturer.

18-Month Load Rejection Test, TS 4.6.a.4

The load rejection test demonstrates the capability of rejecting the maximum rated load without overspeeding or attaining voltages which would cause the diesel generator to trip, mechanical damage, or harmful overstresses.

Operating Cycle Short-Term Load Test, TS 4.6.a.5

Loading the diesel generators to their short-term rating will demonstrate their capability to provide a continuous source of emergency AC power during a load perturbation of up to 112% of the diesel generator's continuous rating.

Station Batteries, TS 4.6.b

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

BASIS

The main steam isolation valves (MSIVs) serve to limit the cooldown rate of the Reactor Coolant System and the reactivity insertion that could result from a main steam line break incident. They also serve to limit the amount of mass and energy released into containment from the unfaulted steam generator during a main steam line break incident. Their ability to close upon signal should be verified at each REFUELING outage. The MSIV closure time assumption used in the main steam line break incident analysis can be found in Section 14.2.5 of the USAR.

BASIS

The Auxiliary Feedwater System (AFW) mitigates the consequences of any event that causes a loss of normal feedwater. The design basis of the AFW System is to remove decay and residual heat by delivering the minimum required flow to at least one steam generator until the Reactor Coolant System (RCS) is cooled to the point of placing the Residual Heat Removal System into operation.

In accordance with ASME Code Section XI, Subsection IWP, an in-service test of each auxiliary feedwater pump shall be run nominally every 3 months (quarterly) during normal plant operation. It is recommended that this test frequency be maintained during shutdown periods if this can be reasonably accomplished, although this is not mandatory. If the normally scheduled test is not performed during a plant shutdown, then the motor-driven pumps shall be demonstrated OPERABLE within 1 week exceeding 350°F; and the turbine-driven pump shall be demonstrated OPERABLE within 72 hours of exceeding 350°.

Quarterly testing of the AFW pumps is used to detect degradation of the component. This type of testing may be accomplished by measuring the pump's developed head at one point of the pump characteristic curve. This verifies that the measured performance is within an acceptable tolerance of the original pump baseline performance.

TS 3.4.b requires all three AFW pumps be OPERABLE prior to heating the RCS average temperature > 350°F. It is acceptable to heat the RCS to > 350°F with the turbine-driven pump inoperable for a limited time period of 72 hours. The wording of TS 3.4.b.4.B and TS 4.8.b allows delaying the testing until the steam flow is consistent with the conditions under which the performance acceptance criteria were generated.

The discharge valves of the two motor-operated pumps are normally open, as are the suction valves from the condensate storage tanks and the two valves on a cross tie line that directs the turbine-driven pump discharge to either or both steam generators. The only valve required to function upon initiation of auxiliary feedwater flow is the steam admission valve on the turbine-driven pump. Proper opening of the steam admission valve will be demonstrated each time the turbine-driven pump is tested.

BASIS - REACTIVITY ANOMALIES[1]

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

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[1] USAR Section 3.2

BASIS

Pressure drop across the combined HEPA filters and charcoal adsorbers of <10 inches of water and 4 inches across any HEPA filter bank at the system design flow rate (±10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle establishes system performance capability. This pressure drop is approximately 2 inches of water when filters are clean.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 (Rev. 1) dated July 1976, except that ASTM D3803-89 standard will be used to fulfill the guidelines of Table 2, item 5, "Radioiodine removal efficiency." The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least 2 inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated July 1976.

If painting, fire, or chemical release occurs such that the charcoal adsorbers become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Degradation of the HEPA filters due to painting, fire or chemical release in a communicating ventilation zone would be detected by an increased pressure drop across the filters. Should the filters become contaminated, engineering judgment would be used to determine if further leakage and/or efficiency testing was required.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

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

BASIS

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those quantities of radioactive by-product materials of interest to this specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

The Eberline Model 1000 Multi-Source Calibrator and the J. L. Shepherd Model 89-400 are totally enclosed instrument calibrating assemblies for which leak testing of the enclosed sources is not practical. Leak testing of these sources would require disassembly of the calibration assembly shield, controls, etc., resulting in personnel exposure without corresponding benefits.

BASIS

Control Room Post-Accident Recirculation System

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water and 4 inches across any HEPA filter bank at the system design flow rate (± 10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A filter test frequency of once per operating cycle establishes system performance capability.

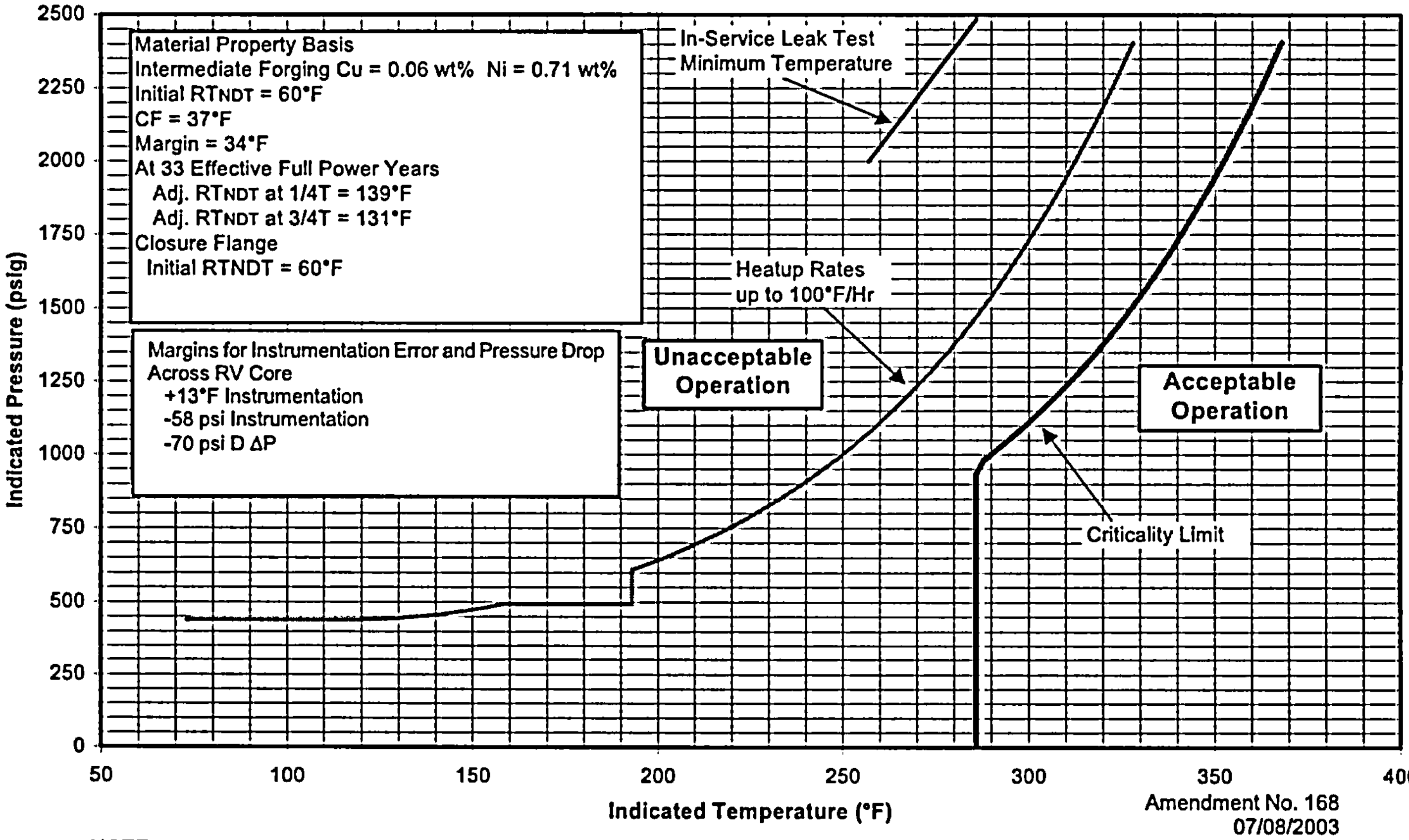
The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 (Rev. 1) dated July 1976, except that ASTM D3803-89 standard will be used to fulfill the guidelines of Table 2, item 5, "Radioiodine removal efficiency." The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced.

Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated July 1976. If painting, fire, or chemical release occurs such that the charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

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

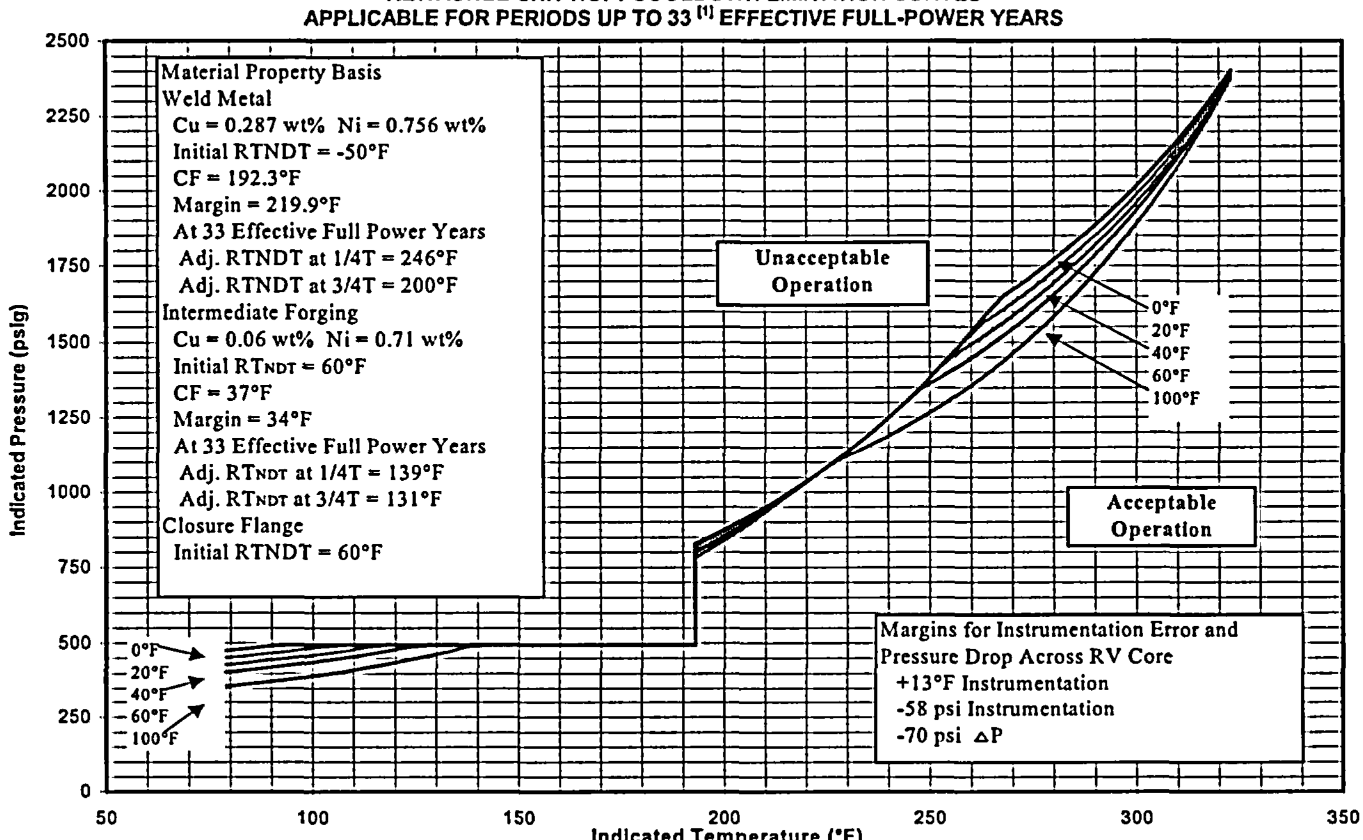
FIGURE TS 3.1-1
KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



Amendment No. 168
07/08/2003

NOTE:
[1]The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-2
KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33 [1] EFFECTIVE FULL-POWER YEARS



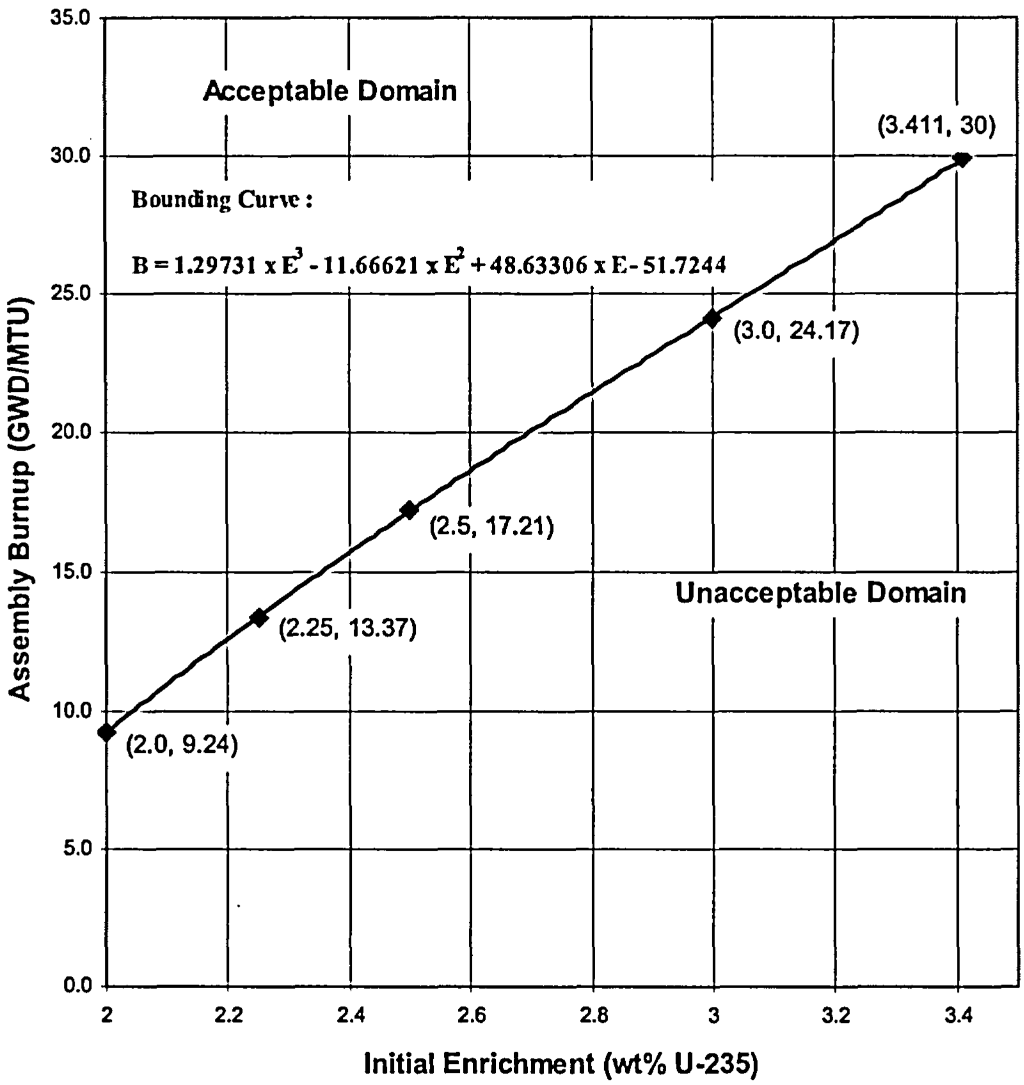
Amendment No. 168
07/08/2003

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 5.4-1

MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF

NOMINAL INITIAL ENRICHMENT TO PERMIT STORAGE IN THE TRANSFER CANAL



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, then 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.

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 non-automatic Containment System isolation valves and blind flanges are closed, except as provided in TS 3.6.b.

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, except as provided in TS 3.6.b.

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.c.

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 the 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 as necessary, such that it responds with acceptable range and accuracy to known values of the parameter that 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 Δk/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%	~T_{oper}	< 2
OPERATING	< 0.25%	~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 1772 MWt.

n. REPORTABLE EVENT

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

o. <u>RADIOLOGICAL EFFLUENTS</u>

1. 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.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in: (1) the calculation of off-site doses due to radioactive gaseous and liquid effluents, (2) the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and (3) the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain: (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. 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 ensure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, Federal and State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

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

5. 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.

p. <u>DOSE EQUIVALENT I-131</u>

DOSE EQUIVALENT I-131 is that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated based on dose conversion factors derived from ICRP-30.

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0059	I-132
0.1692	I-133
0.0010	I-134
0.0293	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

1. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

2. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the nominal zero power design temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

<u>APPLICABILITY</u>

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

<u>OBJECTIVE</u>

To maintain the integrity of the fuel cladding.

<u>SPECIFICATION</u>

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.

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

APPLICABILITY

Applies to the maximum limit on Reactor Coolant System pressure.

OBJECTIVE

To maintain the integrity of the Reactor Coolant System.

SPECIFICATION

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

2.3 LIMITING SAFETY SYSTEM SETTINGS - PROTECTIVE INSTRUMENTATION

APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

A. Source Range (high setpoint)	within span of source range instrumentation
B. Intermediate range (high setpoint)	≤ 40% of RATED POWER
C. Power range (low setpoint)	≤ 25% of RATED POWER
D. Power range (high setpoint)	≤ 109% of RATED POWER
E. Power range fast flux rate trip (positive)	15%Δq/5 sec
F. Power range fast flux rate trip (negative)	10%Δq/5 sec

2. Pressurizer

A. High pressurizer pressure	≤ 2385 psig
B. Low pressurizer pressure	≥ 1875 psig
C. High pressurizer water level	≤ 90% of full scale

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

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.

6. Reactor Trip Interlocks

   Protective instrumentation settings for reactor trip interlocks shall be as follows:

   A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.

   B. Above 10% of RATED POWER, the single loop loss-of-flow trip is made functional.

7. Other Trips

   A. Undervoltage ≥ 75% of normal voltage

   B. Turbine trip

   C. Manual trip

   D. Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

3.0 LIMITING CONDITIONS FOR OPERATION

APPLICABILITY

a. Compliance with the LIMITING CONDITIONS FOR OPERATION contained in the succeeding TSs is required during the operational MODES or other conditions specified therein; except that upon failure to meet the LIMITING CONDITIONS FOR OPERATION, the associated ACTION requirements shall be met.

b. Noncompliance with a TS shall exist when the requirements of the LIMITING CONDITIONS FOR OPERATION and associated ACTION requirements are not met within the specified time intervals. If the LIMITING CONDITIONS FOR OPERATION is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

c. 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.

   Exceptions to these requirements are stated in the individual TSs.

   Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the ACTIONS required by LCO 3.0.c. is not required.

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

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those LIMITING CONDITIONS FOR OPERATION of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.

   C. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures ≤ 200°F unless the secondary water temperature of each steam generator is < 100°F above each of the RCS cold leg temperatures.

2. Decay Heat Removal Capability

   A. At least two of the following four heat sinks shall be OPERABLE whenever the average reactor coolant temperature is ≤ 350°F but > 200°F.

      1. Steam Generator 1A
      2. Steam Generator 1B
      3. Residual Heat Removal Train A
      4. Residual Heat Removal Train B

   If less than the above number of required heat sinks are OPERABLE, then corrective action shall be taken immediately to restore the minimum number to the OPERABLE status.

B. Two residual heat removal trains shall be OPERABLE whenever the average reactor coolant temperature is ≤ 200°F and irradiated fuel is in the reactor, except when in the REFUELING MODE with the minimum water level above the top of the vessel flange ≥ 23 feet, one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

   a) One OPERABLE residual heat removal pump

   b) One OPERABLE residual heat removal heat exchanger

   c) An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, then corrective action shall be taken immediately to return it to the OPERABLE status.

3.1.a.3 Pressurizer Safety Valves

LCO 3.1.a.3 Two pressurizer safety valves shall be OPERABLE

APPLICABILITY: Reactor Coolant System Temperature Greater than the Low Temperature Overpressure Protection (LTOP) Enabling Temperature (200°F)

ACTIONS

- NOTE -
During a hydro test of the RCS, the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge pump are set for the test pressure plus 35 psi to protect the system.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable	A.1 Restore to OPERABLE status	15 Minutes
	OR	
	A.2 Be in HOT SHUTDOWN	12 Hours
B. Both pressurizer safety valves are inoperable	B.1 Restore one pressurizer safety valve to an OPERABLE status	15 Minutes
	OR	
	B.2 Be in a condition with the LTOP system OPERABLE or reactor vessel head removed	48 Hours

4. Pressure Isolation Valves

    A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY MODES, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

    B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[1]

    C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, then an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

    A. Two PORVs and their associated block valves shall be OPERABLE during HOT STANDBY and OPERATING modes.

        1. With one or both PORVs inoperable because of excessive seat leakage, within one hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, action shall be initiated to:

            - Achieve HOT STANDBY within 6 hours
            - Achieve HOT SHUTDOWN within the following 6 hours

        2. With one PORV inoperable due to causes other than excessive seat leakage, within one hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve. Restore the PORV to OPERABLE status within the following 72 hours or action shall be initiated to:

            - Achieve HOT STANDBY within 6 hours
            - Achieve HOT SHUTDOWN within the following 6 hours

        3. With both PORVs inoperable due to causes other than excessive seat leakage, within one hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and

            - Achieve HOT STANDBY within 6 hours
            - Achieve HOT SHUTDOWN within the following 6 hours

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[1] Manual valves shall be locked in the closed position. Motor operated valves shall be placed in the closed position with their power breakers locked out.

4. With one block valve inoperable, within one hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the block valve to OPERABLE status within 72 hours; otherwise action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours

5. With both block valves inoperable, within one hour restore the block valves to OPERABLE status or place their associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour; otherwise, action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours

6. Pressurizer Heaters

   A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is > 350°F.

7. Reactor Coolant Vent System

   A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be OPERABLE and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.

   B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:

      1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.

      2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

      If OPERABILITY is not restored within 30 days, then within one hour action shall be initiated to:

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

   C. If no Reactor Coolant System vent paths are OPERABLE, then restore at least one vent path to OPERABLE status within 72 hours. If OPERABILITY is not restored within 72 hours, then within one hour action shall be initiated to:

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

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33[1] effective full-power years.

   A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

   C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

4. The overpressure protection system for low temperature operation shall be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F, and the reactor vessel head is installed. The system shall be considered OPERABLE when at least one of the following conditions is satisfied:

   A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of ≤ 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.

      1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within five days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional eight hours.

      2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within eight hours.

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[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

B. A vent pathway shall be provided with an effective flow cross section ≥ 6.4 square inches.

1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

A. ≤ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

A. With the specific activity of the reactor coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding 60 µCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

C. With the specific activity of the reactor coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 $or > \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

d. Leakage of Reactor Coolant

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

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 150 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within four hours or be in COLD SHUTDOWN within the next 36 hours.

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

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

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

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is > 250°F.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, then corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, then the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise a safety review by the Plant Operations Review Committee shall be made before starting.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is ≤ 250°F.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, then the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.

6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F.

f. Minimum Conditions for Criticality

1. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.

2. The reactor shall be maintained subcritical by at least 1% Δk/k until normal water level is established in the pressurizer.

3. When the reactor is critical the moderator temperature coefficient shall be as specified in the COLR, except during LOW POWER PHYSICS TESTING. The maximum upper moderator temperature coefficient limit shall be ≤ 5 pcm/°F for power levels ≤ 60% RATED POWER and ≤ 0 pcm/°F for power levels > 60% RATED POWER.

4. If the limits of 3.1.f.3 cannot be met, then power operation may continue provided the following actions are taken:

   A. Within 24 hours, develop and maintain administrative control rod withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits specified in TS 3.1.f.3. These withdrawal limits shall be in addition to the insertion limits specified in TS 3.10.d.

   B. If the actions specified in TS 3.1.f.4.A are not satisfied, then be in HOT STANDBY within the next 6 hours.

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.

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 TESTING 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 ± 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, the following conditions of inoperability may exist during the time interval specified:

   A. One accumulator may have a boron concentration < 1900 ppm for 72 hours.

   B. One accumulator may be inoperable for a reason other than TS 3.3.a.2.A for 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. Emergency Core Cooling System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for LOW POWER PHYSICS TESTING and except as provided by TS 3.3.b.2 and TS 3.3.b.4.

   A. 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 Refueling Water Storage Tank upon a Safety Injection signal and after manual transfer taking suction from the containment sump.

   B. Isolation valves SI-9A, SI-11A, SI-11B, and as a minimum either SI-4A or SI-4B are in the open position with their power breaker locked out.

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.

3. The reactor shall not be made critical unless the following conditions are satisfied except for LOW POWER PHYSICS TESTING and as provided by TS 3.3.b.4.

   A. The Refueling Water Storage Tank shall contain at least 272,500 gallons of water.

   B. The Refueling Water Storage Tank has a boron concentration of at least 2400 ppm.

4. During power operation or recovery from an inadvertent trip, the following conditions of inoperability may exist during the time interval specified.

   A. The calculated Refueling Water Storage Tank boron concentration may be < 2400 ppm for 8 hours.

   B. The Refueling Water Storage Tank may be inoperable for a reason other than that stated in TS 3.3.b.4.A for 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.

5. When the reactor is critical, an OPERABLE SI train may be used to fill one SI Accumulator, for a duration of less than one hour, provided the redundant SI train is also OPERABLE. The provisions of TS 3.7.c are not applicable.

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.

2. Spray Additive System

   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.2.A.3.

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

      2. Valves and piping are capable of adding NaOH solution from the additive tank to a containment spray system.

      3. During power operation or recovery from inadvertent trip, the spray additive system may be out of service for 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 COLD SHUTDOWN within an additional 36 hours.

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.

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

a. Main Steam Safety Valves (MSSVs)

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of two MSSVs per steam generator are OPERABLE.

2. The reactor shall not be made critical unless five MSSVs per steam generator are OPERABLE.

3. If the conditions of TS 3.4.a.1 or TS 3.4.a.2 cannot be met within 48 hours, then 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.

b. Auxiliary Feedwater System

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

    A. Auxiliary feedwater train "A" and auxiliary feedwater train "B" are OPERABLE and capable of taking suction from the Service Water System and delivering flow to the associated steam generator.

    B. The turbine-driven auxiliary feedwater train is OPERABLE and capable of taking suction from the Service Water System and delivering flow to both steam generators, OR

    The turbine-driven auxiliary feedwater train is declared inoperable.

    C. The auxiliary feedwater pump low discharge pressure trip channels are OPERABLE.

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.

3.5 INSTRUCTIONMENTATION SYSTEM

APPLICABILITY

Applies to reactor protection and engineered safety features instrumentation systems.

OBJECTIVE

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded, and to delineate the conditions of the reactor protection instrumentation and engineered safety features circuits necessary to ensure reactor safety.

SPECIFICATIONS

a. Setting limits for instrumentation which initiate operation of the engineered safety features shall be as stated in Table TS 3.5-1.

b. For on-line testing or in the event of failure of a subsystem instrumentation channel, plant operation shall be permitted to continue at RATED POWER in accordance with Tables TS 3.5-2 through TS 3.5-5.

c. If for Tables TS 3.5-2 through TS 3.5-5, the number of channels of a particular subsystem in service falls below the limits given in Column 3, or if the values in Column 4 cannot be achieved, operation shall be limited according to the requirement shown in Column 6, as soon as practicable.

d. In the event of subsystem instrumentation channel failure permitted by TS 3.5.b, Tables TS 3.5-2 through TS 3.5-5 need not be observed during the short period of time (approximately 4 hours) the operable subsystem channels are tested, where the failed channel must be blocked to prevent unnecessary reactor trip.

e. The accident monitoring instrumentation in Table TS 3.5-6 shall be OPERABLE whenever the plant is above HOT SHUTDOWN. In the event the limits given in Columns 1 and 2 cannot be maintained, operator action will be in accordance with the respective notes. A change in operational MODES or conditions is acceptable with an inoperable accident monitoring instrumentation channel(s).

3.6 CONTAINMENT SYSTEM

APPLICABILITY

Applies to the integrity of the Containment System.

OBJECTIVE

To define the operating status of the Containment System.

SPECIFICATION

a. CONTAINMENT SYSTEM INTEGRITY shall not be violated if there is fuel in the reactor which has been used for power operation, except whenever either of the following conditions remains satisfied:

   1. The reactor is in the COLD SHUTDOWN condition with the reactor vessel head installed, or

   2. The reactor is in the REFUELING shutdown condition.

b. Containment Isolation Valves

   1. When CONTAINMENT SYSTEM INTEGRITY is required, all containment isolation valves and blind flanges shall be OPERABLE, except as permitted by TS 3.6.b.2 and TS 3.6.b.3.

   2. Containment Penetration flow paths can be unisolated intermittently under administrative controls. This TS does not apply to the 36" containment purge valves when they are required to be sealed closed.

   3. When CONTAINMENT SYSTEM INTEGRITY is required, the following conditions of inoperability may exist during the time interval specified. Separate entry is allowed into TS 3.6.b.3 for each penetration flowpath.

      A. For one or more penetration flow paths with two containment isolation valves per penetration with one containment isolation valve inoperable:

         1. Return the valve to OPERABLE status within 24 hours or isolate the affected penetrations flow path by use of at least one:

            a) Closed and de-activated automatic valve, or

            b) Closed manual valve, or

            c) Blind flange, or

d) Check valve with flow through the valve secured

2. Verify the affected flow path is isolated:

   a) For isolation devices outside containment, at least once per 31 days, or

   b) For isolation devices inside containment, prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN if not performed within the previous 92 days.

B. For one or more penetration flow paths with two containment isolation valves per penetration with two containment isolation valves inoperable:

1. Return at least one isolation valve to an OPERABLE status within 1 hour or isolate the affected flow path by use of at least one:

   a) Closed and de-activated automatic valve, or

   b) Closed manual valve, or

   c) Blind flange.

2. Verify the affected flow path is isolated:

   a) For isolation devices outside containment, at least once per 31 days, or

   b) For isolation devices inside containment, prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN if not performed within the previous 92 days.

C. For one or more penetration flow paths with one containment isolation valve and a closed system per penetration with one containment isolation valve inoperable:

1. Return the valve to OPERABLE status within 72 hours or isolate the affected penetrations flow path by use of at least one:

   a) Closed and de-activated automatic valve, or

   b) Closed manual valve, or

   c) Blind flange.

2. Verify the affected flow path is isolated:

   a) For isolation devices outside containment, at least once per 31 days, or

   b) For isolation devices inside containment, prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN if not performed within the previous 92 days.

D. Valves and blind flanges in high radiation areas may be verified, as required by TS 3.6.b.3.A.2, TS 3.6.b.3.B.2, and TS 3.6.b.3.C.2, by use of administrative means.

4. If CONTAINMENT SYSTEM INTEGRITY is required and the OPERABILITY requirements of TS 3.6.b.3 are not met within the times specified, then initiate action to:

   A. Achieve HOT STANDBY within the next 6 hours,

   B. Achieve HOT SHUTDOWN within the following 6 hours, and

   C. Achieve COLD SHUTDOWN within the subsequent 36 hours.

c. All of the following conditions shall be satisfied whenever CONTAINMENT SYSTEM INTEGRITY, as defined by TS 1.0.g, is required:

   1. Both trains of the Shield Building Ventilation System, including filters and heaters shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Shield Building Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

   2. Both trains of the Auxiliary Building Special Ventilation System including filters and heaters shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Auxiliary Building Special Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

3. Performance Requirements

   A. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

   B. The results of laboratory carbon sample analysis from the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System carbon shall show ≥ 95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C, 95% RH for the Shield Building Ventilation System and 30°C, 95% RH for the Auxiliary Building Special Ventilation System.

   C. Fans shall operate within ± 10% of design flow when tested.

d. If the internal pressure of the reactor containment vessel exceeds 2 psi, the condition shall be corrected within 8 hours or the reactor shall be placed in a subcritical condition.

e. The reactor shall not be taken above the COLD SHUTDOWN condition unless the containment ambient temperature is > 40°F.

3.7 AUXILIARY ELECTRICAL SYSTEMS

APPLICABILITY

Applies to the availability of electrical power for the operation of plant auxiliaries.

OBJECTIVE

To define those conditions of electrical power availability necessary to provide 1) safe reactor operation and 2) continuing availability of engineered safety features.

SPECIFICATION

a. The reactor shall not be made critical unless all of the following requirements are satisfied:

    1. The reserve auxiliary transformer is fully operational and energized to supply power to the 4160-V buses.

    2. A second external source of power is fully operational and energized to supply power to emergency buses 1-5 and 1-6.

    3. The 4160-V buses 1-5 and 1-6 are both energized.

    4. The 480-V buses 1-52 and 1-62 and their MCC's are both energized from their respective station service transformers.

    5. The 480-V buses 1-51 and 1-61 are both energized from their respective station service transformers.

    6. Both station batteries and both DC systems are OPERABLE, except during testing and surveillance as described in TS 4.6.b.

    7. Both diesel generators are OPERABLE. The two underground storage tanks combine to supply at least 35,000 gallons of fuel oil for either diesel generator and the day tanks for each diesel generator contain at least 1,000 gallons of fuel oil.

    8. At least one pair of physically independent transmission lines serving the substation is OPERABLE. The three pairs of physically independent transmission lines are:

        A. R-304 and Q-303
        B. F-84 and Y-51
        C. R-304 and Y-51

b. During power operation or recovery from inadvertent trip, any 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.

   1. Either auxiliary transformer may be out of service for a period not exceeding 7 days provided the other auxiliary transformer and both diesel generators are OPERABLE.

   2. One diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure OPERABILITY and the engineered safety features associated with this diesel generator are OPERABLE.

   3. One battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain OPERABLE with one charger carrying the d-c supply system.

   4. If the conditions in TS 3.7.a.8 cannot be met, power operation may continue for up to 7 days provided at least two transmission lines serving the substation are OPERABLE.

   5. Three off-site power supply transmission lines may be out of service for a period of 7 days provided reactor power is reduced to 50% of rated power and the two diesel generators shall be tested daily for OPERABILITY.

   6. One 4160-V or 480-V engineered safety features bus may be out of service for 24 hours provided the redundant bus and its loads remain OPERABLE.

c. When its normal or emergency power source is inoperable, a system, train or component may be considered OPERABLE for the purpose of satisfying the requirements of its applicable LIMITING CONDITION FOR OPERATION, provided:

   1. Its corresponding normal or emergency power source is OPERABLE; and

   2. Its redundant system, train, or component is OPERABLE.

3.8 REFUELING OPERATIONS

APPLICABILITY

Applies to operating limitations during REFUELING OPERATIONS.

OBJECTIVE

To ensure that no incident occurs during REFUELING OPERATIONS that would affect public health and safety.

SPECIFICATION

a. During REFUELING OPERATIONS:

1. Containment Closure

   a. The equipment hatch shall be closed and at least one door in each personnel air lock shall be capable of being closed[1] in 30 minutes or less. In addition, at least one door in each personnel air lock shall be closed when the reactor vessel head or upper internals are lifted.

   b. Each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously.

3. The reactor will be subcritical for 148 hours prior to movement of its irradiated fuel assemblies. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment whenever core geometry is being changed. When core geometry is not being changed at least one neutron flux monitor shall be in service.

4. At least one residual heat removal pump shall be OPERABLE.

5. When there is fuel in the reactor, a minimum boron concentration as specified in the COLR shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor. The required boron concentration shall be verified by chemical analysis daily.

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[1] Administrative controls ensure that:
- Appropriate personnel are aware that both personnel air lock doors are open,
- A specified individual(s) is designated and available to close the air lock following a required evacuation of containment, and
- Any obstruction(s) (e.g., cables and hoses) that could prevent closure of an open air lock can be quickly removed.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

7. Heavy loads, greater than the weight of a fuel assembly, will not be transported over or placed in either spent fuel pool when spent fuel is stored in that pool. Placement of additional fuel storage racks is permitted, however, these racks may not traverse directly above spent fuel stored in the pools.

8. The containment ventilation and purge system, including the capability to initiate automatic containment ventilation isolation, shall be tested and verified to be operable immediately prior to and daily during REFUELING OPERATIONS.

9. a. The spent fuel pool sweep system, including the charcoal adsorbers, shall be operating during fuel handling and when any load is carried over the pool if irradiated fuel in the pool has decayed less than 30 days. If the spent fuel pool sweep system, including the charcoal adsorber, is not operating when required, fuel movement shall not be started (any fuel assembly movement in progress may be completed).

   b. Performance Requirements

      1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

      2. The results of laboratory carbon sample analysis from spent fuel pool sweep system carbon shall show $\geq$95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C and 95% RH.

      3. Fans shall operate within $\pm$10% of design flow when tested.

10. The minimum water level above the vessel flange shall be maintained at 23 feet.

11. A dead-load test shall be successfully performed on both the fuel handling and manipulator cranes before fuel movement begins. The load assumed by the cranes for this test must be equal to or greater than the maximum load to be assumed by the cranes during the REFUELING OPERATIONS. A thorough visual inspection of the cranes shall be made after the dead-load test and prior to fuel handling.

12. A licensed senior reactor operator will be on-site and designated in charge of the REFUELING OPERATIONS.

b. If any of the specified limiting conditions for REFUELING OPERATIONS are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be performed.

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 SHUTDOWN MARGIN shall be at least that as specified in the COLR

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 shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ not within limits:

   A. Perform the following:

      i. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER.

      ii. Reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within 72 hours.

      iii. Verify $F_{\Delta H}^N$ within limits within 24 hours.

   B. If the actions of TS 3.10.b.2.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

   i. 50% of RATED POWER,
   ii. 75% of RATED POWER, and
   iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within its limit:

   A. Reduce the thermal power ≥ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

   B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

   C. Verify the $F_Q^N(Z)$ equilibrium relationship and the F_Q^{EQ} (Z) transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

4. 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. (Note: time requirements may be extended by 25%)

   A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

   B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. The measured F_Q^{EQ} (Z) hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the relationship of F_Q^{EQ} (Z) specified in the COLR according to the following schedules with allowances for a 25% grace period:

   A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter.

   B. Once within 12 hours of 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.6.A.

C. If a power distribution map measurement indicates that the F_Q^{EQ} (Z) transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation, then either of the following actions shall be taken:

i. F_Q^{EQ} (Z) transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit, or

ii. Repeat the determination of the F_Q^{EQ} (Z) transient relationship once every seven effective full-power days until either i. above is met, or two successive maps indicate that the F_Q^{EQ} (Z) transient relationship's margin to the transient limit has not decreased.

7. If, for a measured F_Q^{EQ}, the transient relationships of F_Q^{EQ} (Z) specified in the COLR are not within limits, then take the following actions:

A. Reduce the axial flux difference limits ≥ 1% for each 1% the F_Q^{EQ} (Z) transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Verify the F_Q^N(Z) equilibrium relationship and the F_Q^{EQ} (Z) transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, reduce thermal power to less than 50% RATED POWER within 30 minutes.

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 two 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, then 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, then 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, then the reactor shall immediately be brought to a no load condition (≤5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be withdrawn to within the limits, specified in the COLR, when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limits are specified in the COLR. If any one of the control bank insertion limits is not met:

   A. Within one hour, initiate boration to restore control bank insertion to within the limits specified in the COLR, and

   B. Restore control bank insertion to within the limits specified in the COLR within two hours of exceeding the insertion limits.

   C. If any one of the conditions of TS 3.10.d.2.A or TS 3.10.d.2.B cannot be met, then within one hour action shall be initiated to:

      - Achieve HOT STANDBY within 6 hours
      - Achieve HOT SHUTDOWN within the following 6 hours

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin, as specified in the COLR, 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%, then the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If peaking factors are not determined within four 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 four hours, and TS 3.10.b applied. If the peaking factors are not determined within four 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 eight 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 eight 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 eight 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. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ 178,000 gallons per minute average and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, at or above 90% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

3.11 CORE SURVEILLANCE INSTRUMENTATION

APPLICABILITY

Applies to the operability of the movable detector instrumentation system and the core thermocouple instrumentation system.

OBJECTIVE

To specify operability requirements for the movable detector and core thermocouple systems.

SPECIFICATIONS

a. The movable detector system shall be operable following the initial fuel loading and each subsequent reloading and the power distribution confirmed before the reactor is operated at >75% power. If the system is not completely operable, the measurement error allowance due to incomplete mapping shall be substantiated by the licensee.

b. A minimum of 2 movable detector thimbles per quadrant, and sufficient detectors, drives, and readout equipment to map these thimbles, shall be available during re-calibration of the excore axial offset detection system.

c. A minimum of 4 thermocouples per quadrant shall be available for readout if the reactor is operated above 85% with one excore nuclear power channel out of service.

d. The licensee shall utilize his best effort to maintain the movable detector system and the core thermocouple system in an operable state so that surveillance of the core power distribution may be performed. If more than one half of either system is inoperable for 7 consecutive days of power operation, the Commission shall be informed within 30 days. Additional reports of the status of the systems shall be made every 30 days until the system is repaired. Power operation may be continued until the next refueling period provided best efforts are utilized to restore the operability of the system or systems.

BASIS

The moveable detector system is used to measure the core fission power density distribution. A power map made with this system following each fuel loading will confirm the proper fuel arrangement within the core. The moveable detector system is designed with substantial redundancy so that part of the system could be out of service without reducing the value of a power map. If the system is severely degraded, large measurement uncertainty factors must be applied. The uncertainty factors would necessarily depend on the operable configuration.

Two detector thimbles per quadrant are sufficient to provide data for the normalization of the excore detector system's axial power offset feature.

The core thermocouples provide an independent means of measuring the balance of power among the core quadrants. If one excore power channel is out of service, it is prudent to have available an independent means of determining the quadrant power balance.

The moveable detector system and the thermocouple system are not integral parts of the Reactor Protection System. These systems are, rather, surveillance systems which may be required in the event of an abnormal occurrence such as a power tilt or a control rod misalignment. Since such occurrences cannot be predicted a priori, it is prudent to have the surveillance systems in an operable state.

3.12 CONTROL ROOM POST-ACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to the OPERABILITY of the Control Room Post-Accident Recirculation System.

OBJECTIVE

To specify OPERABILITY requirements for the Control Room Post-Accident Recirculation System.

SPECIFICATION

a. The reactor shall not be made critical unless both trains of the Control Room Post-Accident Recirculation System are OPERABLE.

b. Both trains of the Control Room Post-Accident Recirculation System, including filters, shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Control Room Post-Accident Recirculation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

c. During testing the system shall meet the following performance requirements:

   1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filter and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

   2. The results of the laboratory carbon sample analysis from the Control Room Post-Accident Recirculation System carbon shall show ≥ 95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C, and 95% RH.

   3. Fans shall operate within ±10% of design flow when tested.

3.14 SHOCK SUPPRESSORS (SNUBBERS)

APPLICABILITY

Applies to the OPERABILITY of shock suppressors which are related to plant safety.

OBJECTIVE

To ensure that shock suppressors, which are used to restrain safety-related piping under dynamic load conditions, are functional during reactor operation.

SPECIFICATION

a. The reactor shall not be made critical unless all safety-related shock suppressors are OPERABLE except as noted in 3.14.b.

b. During power operation or recovery from inadvertent trip, if any safety-related shock suppressor is found inoperable one of the following actions shall be taken within 72 hours:

   1. The inoperable shock suppressor shall be restored to an OPERABLE condition or replaced with a spare shock suppressor of similar specifications; or

   2. The fluid line restrained by the inoperable shock suppressor shall, if feasible, be isolated from other safety-related systems if otherwise permitted by the TS and thereafter operation may continue subject to any limitations by the TS for that fluid line; or

   3. Actions shall be initiated to shut down the reactor and the reactor shall be in a HOT SHUTDOWN condition within 36 hours.

4.0 SURVEILLANCE REQUIREMENTS

APPLICABILITY

a. Surveillance requirements shall be met during the operational MODES or other conditions specified for individual LIMITING CONDITIONS FOR OPERATION (LCO) unless otherwise stated in an individual surveillance requirement. Failure to meet a surveillance requirement, whether such failure is experienced during the performance of the surveillance or between performances of the surveillances, shall be failure to meet the OPERABILITY requirements for the LCO. Failure to perform a surveillance within the allowed surveillance interval, defined by TS 4.0.b, shall be a failure to meet the OPERABILITY requirements for the LCO except as provided in TS 4.0.c. Surveillance requirements do not have to be performed on inoperable equipment.

b. Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

c. If it is discovered that a surveillance was not performed within its allowed surveillance interval, then compliance with the requirement to declare the OPERABILITY requirements for the LCO not met may be delayed from the time of discovery up to 24 hours, or up to the limit of the allowed surveillance interval, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

   If the surveillance is not performed within the delay period, the OPERABILITY requirements for the LCO must immediately be declared not met, and the applicable conditions(s) must be entered.

   When the surveillance is performed within the delay period and the surveillance is not met, the OPERABILITY requirements for the LCO must immediately be declared not met, and the applicable conditions(s) must be entered.

d. Entry into an operational MODE or other specified condition shall not be made unless the surveillance requirement(s) associated with a LIMITING CONDITION FOR OPERATION have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational MODES as required to comply with action requirements.

   Exceptions to these requirements are stated in the individual Technical Specifications.

4.1 OPERATIONAL SAFETY REVIEW

APPLICABILITY

Applies to items directly related to safety limits and LIMITING CONDITIONS FOR OPERATION.

OBJECTIVE

To assure that instrumentation shall be checked, tested, and calibrated, and that equipment and sampling tests shall be conducted at sufficiently frequent intervals to ensure safe operation.

SPECIFICATION

a. Calibration, testing, and checking of protective instrumentation channels and testing of logic channels shall be performed as specified in Table TS 4.1-1.

b. Equipment and sampling tests shall be conducted as specified in Table TS 4.1-2 and TS 4.1-3.

c. Deleted

d. Deleted

e. Deleted

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, 3, and MC components.

SPECIFICATION

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

1. In-service inspection of ASME Code Class 1, Class 2, Class 3, and Class MC 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 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 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 50.55a(f), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(f)(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.

[(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 requirements of the inspection program.

Imperfection is a deviation from the dimension, finish, or contour required by a design drawing or specification.

Degradation means service-induced cracking, wastage, wear or corrosion of a tube wall.

% Degradation is the amount in percent of tube wall thickness affected or removed by degradation.

Degraded Tube means a tube containing degradation that is ≥ 20% of nominal wall thickness.

Defect means an imperfection that violates criteria used to determine acceptability of a tube for continued use in operation.

Tube Inspection means the detailed examination of a steam generator tube from the point of entry (e.g., hot leg side) around the U-bend to the level of the top tube support plate of the opposite leg (cold leg).

Tube is a single hollow metal cylinder that is an element of an array of similar cylinders inside each steam generator, through which Reactor Coolant flows, and by which heat is transferred from the Reactor Coolant to the secondary system feedwater. Taken as a whole, steam generator tubes form a major portion of the reactor coolant pressure boundary.

Plugged Tube is a tube that has been removed from service by installing a mechanical device in each end of the tube to seal the tube in a manner that isolates it from the reactor coolant system.

1. Steam Generator Sample Selection and Inspection

   In-service inspection of steam generators may be limited to one steam generator per inspection period on an alternating basis. The tubes shall be selected for inspection as set forth in TS 4.2.b.2.a, provided that previous inspections indicate the two steam generators are performing in an acceptably similar manner.

2. Steam Generator Tube Sample Selection and Inspection

   Each in-service inspection:

   a. Shall include a number of tubes that is at least equal to 3% of the total number of non-plugged tubes contained in both steam generators. Tubes shall be selected for inspection on a random basis except as noted in TS 4.2.b.2.b.

   b. Shall concentrate the inspection by selecting 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. Shall include all non-plugged tubes in which previous inspections revealed degradation that exceeded 20% of nominal wall thickness. For these tubes, only the area previously identified as degraded must be inspected, unless their inspection is also performed to satisfy requirements of TS 4.2.b.2.a and TS 4.2.b.2.b above.

   d. May not require inspection of the full length of each tube during the second and third sample inspections but may concentrate the inspection only on those portions of the tubes previously found degraded.

   e. Shall perform a tube inspection on each selected tube. If the eddy current inspection probe will not pass through the entire length of a tube, including the U-bend, it shall be so recorded and the tube shall be characterized as degraded. An adjacent tube shall also be inspected.

   f. Shall classify sample inspection results as belonging to one of the following three categories, and actions shall accordingly be taken as described in Table TS 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 Between 5% and 10% of the total tubes inspected are degraded tubes, or one or more tubes, but not more than 1% of the total tubes inspected, are defective.

C-3 More than 10% of the total tubes inspected are degraded tubes, or more than 1% of the inspected tubes are defective.

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

3. Inspection Frequency

In-service inspection of steam generator tubes shall be performed at the following intervals:

a. In-service inspections may be performed during refueling outages, but shall be performed at intervals not to exceed 24 calendar months, except that the inspection interval may be extended to a maximum of 40 months if:

   1. two consecutive inspections following service under AVT conditions, not including the pre-service inspection, yield results that fall into the C-1 category, or

   2. two consecutive inspections demonstrate that previously documented degradation sites have not continued to deteriorate and no new degradation is found.

   NOTE: A one-time inspection interval extension of a maximum of once per 40 months is allowed following the inspection performed during the Spring 2003 Outage. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category.

b. If the result of a steam generator in-service inspection conducted in accordance with Table TS 4.2-2 falls into Category C-3, the inspection interval shall be reduced to 20 months. The 20 month interval shall apply until a subsequent inspection meets the conditions set forth in TS 4.2.b.3.a for extending the interval to 40 months.

c. Additional, unscheduled in-service inspections of each steam generator shall be performed using the criteria set forth in Table 4.2-2 for a "1st SAMPLE INSPECTION" during shutdowns consequent to:

   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.d, or

   2. A seismic event having a magnitude greater than the Operating Basis Earthquake, or

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

   4. A main steam line or feedwater line break, where the Reactor Coolant System cooldown rate exceeded 100°F/hr.

d. If there is a significant change in steam generator chemistry control methodology, the steam generators shall be operated at power for three months while using the new treatment and shall then be inspected during the next outage of sufficient duration.

4. Plugging Limit Criteria

Any tube with tube wall degradation of 50% or more shall be plugged before returning the steam generator to service. If significant general tube thinning occurs, this criterion is reduced to 40% wall degradation.

5. Deleted

6. Deleted

7. Reports

   a. Following each in-service inspection of steam generator tubes during which tubes are plugged, the number of tubes plugged shall be reported to the Commission within 60 days.

b. The results of each steam generator tube in-service inspection shall be included in the Annual Operating Report for the reporting period that included completion of the inspection. The report shall include:

   1. Number of tubes inspected and extent of inspection.

   2. Location of each tube wall degradation and its percent of wall penetration.

   3. Identification of tubes plugged.

c. If a steam generator tube inspection result falls into Category C-3, the Commission shall be promptly notified according to requirements of 10 CFR 50.72(b)(3)(ii). A Licensee Event Report shall then be filed with the Commission as described by Specification 4.2.b.7.a and as set forth in 10 CFR 50.73(a)(2)(ii).

4.4 CONTAINMENT TESTS

APPLICABILITY

Applies to integrity testing of the steel containment, shield building, auxiliary building special ventilation zone, and the associated systems including isolation valves.

OBJECTIVE

To verify that leakage from the containment system is maintained within allowable limits in accordance with 10 CFR Part 50, Appendix J.

SPECIFICATION

a. Integrated Leak Rate Tests (Type A)

   Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

   As a one-time change, the Type A test frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is changed to "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in April 1994.

b. Local Leak Rate Tests (Type B and C)

   Perform required air lock, penetration, and containment isolation valve leakage testing in accordance with the Containment Leakage Rate Testing Program.

c. Shield Building Ventilation System

   1. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

      a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is < 10 inches of water and the pressure drop across any HEPA filter bank is < 4 inches of water at the system design flow rate (±10%).

      b. Automatic initiation of each train of the system.

      c. Operability of heaters at rating and the absence of defects by visual observation.

2. Shield Building Ventilation System Filter Testing

   a. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   b. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

   d. Each train shall be operated with the heaters on at least 10 hours every month.

3. An air distribution test on these HEPA filter banks will be performed after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at design flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.[1]

4. Each train shall be determined to be operable at the time of its periodic test if it produces measurable indicated vacuum in the annulus within 2 minutes after initiation of a simulated safety injection signal and obtains equilibrium discharge conditions that demonstrate the Shield Building leakage is within acceptable limits.

[1] In WPS letter of August 25, 1976 to Mr. Al Schwencer (NRC) from Mr. E. W. James, we relayed test results for flow distribution for tests performed in accordance with ANSI N510-1975. This standard refers to flow distribution tests performed upstream of filter assemblies. Since the test results upstream of filters were inconclusive due to high degree of turbulence, tests for flow distribution were performed downstream of filter assemblies with acceptable results (within 20%). The safety evaluation attached to Amendment 12 references our letter of August 25, 1976 and acknowledges acceptance of the test results.

d. Auxiliary Building Special Ventilation System

1. Periodic tests of the Auxiliary Building Special Ventilation System, including the door interlocks, shall be performed in accordance with TS 4.4.c.1 through TS 4.4.c.3, except for TS 4.4.c.2.d.

2. Each train of Auxiliary Building Special Ventilation System shall be operated with the heaters on at least 15 minutes every month.

3. Each system shall be determined to be operable at the time of periodic test if it starts with coincident isolation of the normal ventilation ducts and produces a measurable vacuum throughout the special ventilation zone with respect to the outside atmosphere.

e. Containment Vacuum Breaker System

The power-operated valve in each vent line shall be tested during each refueling outage to demonstrate that a simulated containment vacuum of 0.5 psig will open the valve and a simulated accident signal will close the valve. The check and butterfly valves will be leak tested in accordance with TS 4.4.b during each refueling, except that the pressure will be applied in a direction opposite to that which would occur post-LOCA.

f. Containment Isolation Device Position Verification

1. When the reactor is critical, verify each 36 inch containment purge and vent isolation valve is sealed closed every 31 days.

2. When the reactor is critical, verify each 2 inch containment vent isolation valve is closed every 31 days, except when the 2 inch containment vent isolation valves are open for pressure control, ALARA, or air quality considerations for personnel entry, or Surveillances that require the valves to be open.

3. Containment isolation manual valves and blind flanges shall be verified closed as specified in TS 4.4.f.3.a and TS 4.4.f.3.b, except as allowed by TS 4.4.f.3.c.

   a. When greater than COLD SHUTDOWN, verify each containment isolation manual valve and blind flange that is located outside containment and required to be closed during accident conditions is closed every 31 days, except for containment isolation valves that are locked, sealed, or otherwise secured closed or open as allowed by TS 3.6.b.2.

b. Prior to entering INTERMEDIATE SHUTDOWN from COLD SHUTDOWN, if not performed in the previous 92 days, verify each containment isolation manual valve and blind flange that is located inside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are locked sealed or otherwise secured closed or open as allowed by TS 3.6.b.2.

c. Valves and blind flanges in high radiation areas may be verified by use of administrative means.

4.5 EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT AIR COOLING SYSTEM TESTS

APPLICABILITY

Applies to testing of the Emergency Core Cooling System and the Containment Air Cooling System.

OBJECTIVE

To verify that the subject systems will respond promptly and perform their design functions, if required.

SPECIFICATION

a. System Tests

1. Safety Injection System

A. System tests shall be performed once per operating cycle or once every 18 months, whichever occurs first. With the Reactor Coolant System pressure ≤ 350 psig and temperature ≤ 350°F, a test safety injection signal will be applied to initiate operation of the system.

B. The test will be considered satisfactory if control board indication or visual observations indicate that all components have received the safety injection signal in the proper sequence and timing. That is, the appropriate pump motor breakers shall have opened and closed, and all valves shall have completed their travel.

2. Containment Vessel Internal Spray System

A. System tests shall be performed once every operating cycle or once every 18 months, whichever occurs first. The test shall be performed with the isolation valves in the supply lines at the containment blocked closed.

B. Verify a minimum of 76 spray nozzles per train are functioning properly by using an air or smoke test at a test interval not to exceed 10 years.

C. The test will be considered satisfactory if control board indications or visual observations indicate all components have operated satisfactorily.

3. Containment Fancoil Units

   Each fancoil unit shall be tested once every operating cycle or once every 18 months, whichever occurs first, to verify proper operation of the motor-operated service water outlet valves and the fancoil emergency discharge and associated backdraft dampers.

b. Component Tests

   1. Pumps

      A. The safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started and operated quarterly during power operation and within 1 week after the plant is returned to power operation, if the test was not performed during plant shutdown.

      B. Acceptable levels of performance are demonstrated by the pumps' ability to start and develop head within an acceptable range.

   2. Valves

      A. The containment sump outlet valves shall be tested during the pump tests.

      B. The accumulator check valves shall be checked for OPERABILITY during each major REFUELING outage. The accumulator block valves shall be checked to assure "valve open" requirements during each major REFUELING outage.

      C. Deleted

      D. Spray additive tank valves shall be tested during each major REFUELING outage.

      E. Deleted

      F. Residual Heat Removal System valve interlocks shall be tested once per operating cycle.

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

<u>APPLICABILITY</u>

Applies to periodic testing and surveillance requirements of the emergency power system.

<u>OBJECTIVE</u>

To verify that the emergency power sources and equipment are OPERABLE.

<u>SPECIFICATION</u>

The following tests and surveillance shall be performed:

a. Diesel Generators

1. Manually-initiated start of each diesel generator, and assumption of load by the diesel generator. This test shall be conducted monthly, loading the diesel generator to at least 2600 KW (nominal) for a period of at least 1 hour.

2. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, all initiated by a simulated loss of all normal a-c station service power supplies together with a simulated safety injection signal. This test will be conducted at each REFUELING interval to assure that each diesel generator will start and assume required loads to the extent possible within 1 minute, and operate for ≥ 5 minutes while loaded with the emergency loads.

3. Each diesel generator shall be inspected at each major REFUELING outage.

4. Diesel generator load rejection test in accordance with IEEE 387-1977, Section 6.4.5, shall be performed at least once per 18 months.

5. Each diesel generator shall be loaded to 2950 KW (nominal) for 2 hours every operating cycle.

6. Safeguard bus undervoltage and safeguard bus second level undervoltage relays shall be calibrated at least once per operating cycle.

b. Station Batteries

1. The voltage of each cell shall be measured to the nearest hundredth volt each month. An equalizing charge shall be applied if the lowest cell in the battery falls < 2.13 volts. The temperature and specific gravity of a pilot cell in each battery shall be measured.

2. The following additional measurements shall be made quarterly: the specific gravity and height of electrolyte in every cell and the temperature of every fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration.

4. The batteries shall be subjected to a load test during the first REFUELING and once every 5 years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

4.7 MAIN STEAM ISOLATION VALVES

APPLICABILITY

Applies to periodic testing of the main steam isolation valves.

OBJECTIVE

To verify the ability of the main steam isolation valves to close upon signal.

SPECIFICATION

The main steam isolation valves shall be tested once per operating cycle. A closure time of 5 seconds or less shall be verified.

4.8 AUXILIARY FEEDWATER SYSTEM[1]

APPLICABILITY

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

OBJECTIVE

To verify the OPERABILITY of the auxiliary feedwater equipment and its ability to respond properly when required.

SPECIFICATION

a. The OPERABILITY of the motor-driven auxiliary feedwater pumps as required by TS 3.4.b.1.A shall be demonstrated quarterly during power operation and within one week after the pumps are required to be operable by the Technical Specifications, if the test surveillance interval expired during the shutdown period.

b. The OPERABILITY of the turbine-driven auxiliary feedwater pump as required by TS 3.4.b.1.B shall be demonstrated quarterly during power operation and within 72 hours after exceeding 350°F, if the test surveillance interval expired during the shutdown period.

c. The valves on the discharge side of the turbine-driven pump that direct flow to either steam generator shall be tested by operator action whenever the turbine-driven pump is tested.

d. The service water supply valves to the auxiliary feedwater pump suctions shall be tested by operator action following the auxiliary feedwater pump tests.

e. These tests shall be considered satisfactory if control board indication or visual observation of the equipment demonstrate that all components have operated properly.

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[1] USAR Section 6.6

4.9 REACTIVITY ANOMALIES

APPLICABILITY

Applies to potential reactivity anomalies.

OBJECTIVE

To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of 1% in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Commission within 30 days.

4.12 SPENT FUEL POOL SWEEP SYSTEM

APPLICABILITY

Applies to testing and surveillance requirements for the spent fuel pool sweep system in TS 3.8.a.9.

OBJECTIVE

To verify the performance capability of the spent fuel pool sweep system.

SPECIFICATION

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is < 10 inches of water and the pressure drop across any HEPA bank is < 4 inches of water at the system design flow rate (± 10%).

   2. Automatic initiation of each train of the system.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

c. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the system. The test shall be performed at design flow rate (± 10%). The results of the test shall show the air distribution is uniform within ± 20%[1].

[1] In WPS letter of August 25, 1976 to Mr. Al Schwencer (NRC) from Mr. E. W. James, we relayed test results for flow distribution for tests performed in accordance with ANSI N510-1975. This standard refers to flow distribution tests performed upstream of filter assemblies. Since the test results upstream of filters were inconclusive due to high degree of turbulence, tests for flow distribution were performed downstream of filter assemblies with acceptable results (within 20%). The safety evaluation attached to Amendment 12 references our letter of August 25, 1976 and acknowledges acceptance of the test results.

4.13 RADIOACTIVE MATERIALS SOURCES

APPLICABILITY

Applies to the possession, leak test, and record requirements for radioactive material sources required for operation of the facility.

OBJECTIVE

To ensure that radioactive material sources which are beneficial to facility operation are available to the facility and these sources are verified to be free from leakage.

SPECIFICATION

a. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or the State.

b. Sources which contain by-product material that exceeds the quantities listed in 10 CFR 30.71, Schedule B, and all other sources containing > 0.1 microcuries shall be leak tested in accordance with this TS.

c. Any source specified by TS 4.13.b which is determined to be leaking shall be immediately withdrawn from use, repaired or disposed of in accordance with the Commission's regulations. Leaking is defined as the presence of .005 microcuries of the source's radioactive material on the test sample.

d. Each sealed source with a half-life > 30 days, and in any form other than gas, shall be tested for leakage at intervals not to exceed 6 months, except for:

   1. Startup sources inserted in the reactor vessel,
   2. Fission detectors following exposure to core flux,
   3. Irradiation sample sources inserted in the reactor vessel,
   4. Sources enclosed within the Eberline Model 1000 Multi-Source Gamma Calibrator,
   5. Sources enclosed within the Shepherd Model 89-400 Self-Contained Calibrator, and
   6. Hydrogen-3 sources.

e. Sources specified by TS 4.13.b which are in storage and not being used are exempt from the testing of TS 4.13.d. Prior to use or transfer to another licensee of such a source, the leakage test of TS 4.13.d shall be current.

f. Startup sources and fission detectors shall be leak tested prior to initial insertion into the reactor vessel or prior to being subjected to core flux.

g. A complete inventory of radioactive materials sources shall be maintained current at all times.

4.14 TESTING AND SURVEILLANCE OF SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to periodic testing and surveillance requirements of safety related shock suppressors.

Objective

To verify operability of shock suppressors.

Specification

The following surveillance and testing is required for hydraulic shock suppressors required to be operable by Specification 3.14:

a. All hydraulic shock suppressors whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected to verify integrity of hydraulic fittings, reservoirs and cylinders and mechanical integrity of linkage connections to piping and anchors. These inspections shall be in accordance with the following schedule:

Number of hydraulic shock suppressors found inoperative during inspection or during inspection interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 - 4	124 days ± 25%
5 - 7	62 days ± 25%
≥ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step per inspection interval.

All hydraulic shock suppressors whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

Shock suppressors are categorized as "accessible" or "inaccessible". For the purpose of this inspection these two groups may be considered independently and scheduled accordingly.

b. A representative sample of 10% of the safety related shock suppressors shall be functionally tested for operability including verification of proper piston movement, lockup, and bleed at each refueling. For each shock suppressor or subsequent shock suppressor found inoperable by this testing requirement, an additional 10% shall be tested until no more failures are found or all shock suppressors have been tested. Those shock suppressors designated to be difficult to remove or in a high radiation area during shutdown need not be selected for functional testing. The Anchor Holth suppressors used on the steam generators are exempt from functional testing requirements.

BASIS

All safety related hydraulic shock suppressors are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests, or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed to an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories have shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement and snubbing action. Ten percent of the safety-related snubbers represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. The Anchor Holth suppressors used on the steam generators are exempt from the functional test requirement due to the impracticability of functionally testing 900 Kip suppressors.

4.16 Reactor Coolant Vent System Tests

Applicability

Applies to the surveillance testing requirements of the reactor coolant vent system.

Objectives

To assure that the capability exists to vent non-condensible gases from the reactor coolant system, if required.

Specification

a. Vent Path Operability

At least once per operating cycle or once every 18 months, whichever occurs first, each reactor coolant system vent path shall be demonstrated operable by:

1) Cycling each solenoid operated valve in each vent path through at least one complete cycle of full travel.

2) Verifying that unobstructed flow exists through the reactor coolant vent system paths during the normal filling and venting operations following refueling.

Basis

The cycling of each solenoid operated valve once each refueling ensures that the valves are capable of opening, if required, to vent the reactor coolant system. More frequent cycling of these valves is not practical since it would provide unnecessary challenges to the reactor coolant pressure boundary during plant operation.

Flow verification is performed to assure that there are no blockages in the reactor coolant system vent piping that would prevent venting of non-condensible gases from the reactor coolant system. Flow verification is performed following each refueling by qualitatively assuring flow exists through the system during the postrefueling filling and venting of the RCS.

4.17 CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to testing and surveillance requirements for the Control Room Postaccident Recirculation System in TS 3.12.

OBJECTIVE

To verify the performance capability of the Control Room Postaccident Recirculation System.

SPECIFICATION

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches of water and the pressure drop across any HEPA bank is < 4 inches of water at the system design flow rate (± 10%).

   2. Automatic initiation of the system on a high radiation signal and a safety injection signal.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

   4. Each train shall be operated at least 10 hours each month.

5.0 DESIGN FEATURES

5.1 SITE

APPLICABILITY

Applies to the location and extent of the reactor site.

OBJECTIVE

To define those aspects of the site which affect the overall safety of the installation.

SPECIFICATION

The Kewaunee Nuclear Power Plant is located on property owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company at a site on the west shore of Lake Michigan, approximately 30 miles east-southeast of the city of Green Bay, Wisconsin.

The minimum distance from the center line of the reactor containment to the site exclusion radius as defined in 10 CFR 100.3 is 1200 meters.

5.2 CONTAINMENT

APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

OBJECTIVE

To define the significant design features of the Containment System.

SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the design basis accident are within the guidelines of 10 CFR Part 50.67. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel reactor containment vessel designed for the peak pressure of the design basis accident.

   B. A concrete shield building which surrounds the containment vessel, providing a shield building annulus between the two structures.

   C. A Shield Building Ventilation System that causes leakage from the reactor containment vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System that serves the special ventilation zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

b. Reactor Containment Vessel

1. The reactor containment vessel is designed for the peak internal pressure of the design basis accident plus the loads resulting from an earthquake producing 0.06g horizontally and 0.04g vertically. It is also designed to withstand an external pressure 0.8 psi greater than the internal pressure.

2. Penetrations of the containment vessel for piping, electrical conductors, ducts and access hatches are provided with double barriers against leakage.

3. The automatically actuated containment valves are designed to close upon high containment pressure and on a safety injection signal. The actuation system is designed so that no single component failure will prevent containment isolation, if required.

c. Shield Building

The shield building is a reinforced concrete structure with a wall thickness of 2.5 feet and a dome thickness of 2 feet. It is designed for the same seismic conditions as the reactor containment vessel and is designed to resist a 3 psi internal pressure due to tornadoes.

d. Shield Building Ventilation System

In the event of a loss-of-coolant accident, the Shield Building Ventilation System will relieve the initial thermal expansion of air through particulate and charcoal filters and will then cause a vacuum to be produced throughout the shield building annulus. A momentary positive pressure no greater than 0.5 psi will result during the thermal expansion. Once vacuum is achieved, the system causes the air within the annulus to be recirculated through the filters while vacuum is maintained. The filtered mixture of annulus air plus leakage is vented through the Containment System vent by the discharge fan that maintains vacuum at a vent rate determined by in-leakage to the shield building.[1]

e. Auxiliary Building Special Ventilation Zone and Special Ventilation System

A limited amount of containment leakage could potentially escape through certain penetrations in the event of leakage in the isolation valves, as described in the Basis of TS 3.6. The leakage escaping into that portion of the auxiliary building which is designed for medium leakage and controlled access would be processed by the Auxiliary Building Special Ventilation System. When actuated, the system will draw all in-leakage air from this special ventilation zone and exhaust it through particulate and charcoal filters to the auxiliary building vent.[2]

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[1] USAR Section 5.5
[2] USAR Section 9.6

5.3 REACTOR CORE

APPLICABILITY

Applies to the reactor core.

OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy, ZIRLO™, or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

5.4 FUEL STORAGE

APPLICABILITY

Applies to the capacity and storage arrays of new and spent fuel.

OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

SPECIFICATION

a. Criticality

1. The spent fuel storage racks are designed and shall be maintained with the following:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties

2. The new fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties

   c. $k_{eff} < 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties

3. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor REFUELING cavity and REFUELING canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

The spent fuel storage pool is designed with a storage capacity of 1205 assemblies and shall be limited to no more than 1205 fuel assemblies.

c. Canal Rack Storage

Fuel assemblies stored in the canal racks shall meet the minimum required fuel assembly burnup as a function of nominal initial enrichment as shown in Figure TS 5.4-1. These assemblies shall also have been discharged prior to or during the 1984 REFUELING outage.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

a. The Manager - Kewaunee Plant shall be responsible for overall plant operation and shall delegate in writing the succession of this responsibility during his absence.

b. The Manager - Kewaunee Plant, or his designee, shall approve prior to implementation, each proposed test, experiment or modification to structures, systems or components that affect nuclear safety.

6.2 ORGANIZATION

a. Off-Site Staff

The off-site organization for plant management and technical support shall be as described in the Operational Quality Assurance Program Description.

b. Facility Staff

The plant organization shall be as described in the Operational Quality Assurance Program Description.

1. Each on-duty shift complement shall consist of at least:

   A. One Shift Manager (SRO)
   B. Two licensed Reactor Operators
   C. Two Nuclear Auxiliary Operators
   D. Deleted
   E. One Radiation Technologist

2. While above COLD SHUTDOWN, the on-duty shift complement shall consist of the personnel required by TS 6.2.b.1 and an additional SRO.

3. In the event that one of the shift members becomes incapacitated due to illness or injury or the Radiation Technologist has to accompany an injured person to the hospital, reactor operations may continue with the reduced complement until a replacement arrives. In all but severe weather conditions, a replacement is required within two hours.

4. At least one licensed operator shall be in the control room when fuel is in the reactor.

5. Two licensed operators, one of which shall be an SRO, shall be present in the control room when the unit is in an operational MODE other than COLD SHUTDOWN or REFUELING.

6. REFUELING OPERATIONS shall be directed by a licensed SRO assigned to the REFUELING OPERATION who has no other concurrent responsibilities during the REFUELING OPERATION.

7. When the reactor is above the COLD SHUTDOWN condition, a qualified Shift Technical Advisor shall be within 10 minutes of the control room.

c. Organizational Changes

Changes not affecting safety may be made to the off-site and facility staff organizations. Such changes that are described in the Technical Specifications shall be reported to the Commission in the form of an application for license amendment within 60 days of the implementation of the change.

6.3 PLANT STAFF QUALIFICATIONS

a. Qualification of each member of the Plant Staff shall meet or exceed the minimum acceptable levels of ANSI N18.1-1971 for comparable positions, except for:

1. The Radiation Protection Manager who shall meet or exceed the recommendation of Regulatory Guide 1.8, Revision 1-R, September 1975, or their equivalent as further clarified in Attachment 1 to the Safety Evaluation Report enclosed with Amendment No. 46 to Facility Operating License DPR-43.

2. The education and experience eligibility requirements for operator license applicants, changes thereto, shall be those previously reviewed and approved by the NRC, specifically those referenced in NRC Safety Evaluation letter dated October 2, 2003 (K-03-140).

b. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in the design of the Kewaunee Plant and plant transient and accident analysis.

6.4 TRAINING

A retraining and replacement training program for the Plant Staff shall be maintained and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI-N18.1-1971 and 10 CFR Part 55.

6.7 SAFETY LIMIT VIOLATION

The following actions shall be taken in the event a SAFETY LIMIT is violated:

a. The reactor shall be shut down and operation shall not be resumed until authorized by the Commission.

b. The Report shall be prepared in accordance with 10 CFR 50.72 and 10 CFR 50.73.

6.8 PROCEDURES

a. Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements and recommendations of Section 5.2.2, 5.2.5, 5.2.15 and 5.3 of ANSI N18.7-1976.

b. Changes to procedures are made in accordance with the provisions of ANSI N18.7-1976 Section 5.2.2, except temporary changes which clearly do not change the intent of the procedure shall, as a minimum, be approved by two individuals knowledgeable in the area affected one of which holds an active SRO license at Kewaunee.

c. Procedures are reviewed in accordance with the provisions of ANSI N18.7-1976, Section 5.2.15. The biennial review requirement is accomplished through alternate programs as described in the OQAPD.

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) nine 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. Deleted

B. As per applicable, portions of Regulatory Guide 1.16, 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 person 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). 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]

D. This report shall document the results of specific activity analysis in which the reactor coolant exceeded the limits of TS 3.1.c.1.A during the past year. The following information shall be included:

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations.

(3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded.

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level.

(5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

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[1] This tabulation supplements the requirements of Section 20.2206(b) of 10 CFR Part 20.
[2] Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (U.S. NRC) dated January 5, 1981.

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.

4. Core Operating Limits Report (COLR)

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

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.8.a.5	Refueling Boron Concentration
(6)	TS 3.10.a	Shutdown Margin
(7)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(8)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(9)	TS 3.10.b.4	$F_Q^{EQ}(Z)$ Limits
(10)	TS 3.10.b.5.C.i	$F_Q^{EQ}(Z)$ penalty
(11)	TS 3.10.b.9	Axial Flux Difference Target Band
(12)	TS 3.10.b.11.A	Axial Flux Difference Envelope
(13)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(14)	TS 3.10.d.2	Control Bank Insertion Limits
(15)	TS 3.10.k	Core Average Temperature
(16)	TS 3.10.l	Reactor Coolant System Pressure
(17)	TS 3.10.m.1	Reactor Coolant Flow

B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents.

(1) Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods For Application To Kewaunee" Report, dated August 21, 1979, report date September 29, 1978

(2) Kewaunee Nuclear Power Plant – Review For Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC No. MB0306) dated September 10, 2001.

(3) S.M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of –Coolant Accident Analysis, Volume I, Rev. 2, and Volume II-V, Rev.1, and WCAP-14747 (Non-Proprietary) March 1998.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(10) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(11) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(12) S.I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Rev. 1 (Proprietary and WCAP-14450-NP-A, Rev. 1 (Non-Proprietary), October 1999.

(13) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

(14) WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989.

(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," Rev. 1, May 2000.

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

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

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. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

   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. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

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.

6.10 RECORD RETENTION

a. The following records shall be retained for at least five years:

1. Records and logs of plant operation, including power levels and periods of operation at each power level.

2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment pertaining to nuclear safety.

3. Reports of all REPORTABLE EVENTS.

4. Records of periodic checks, inspections, and calibrations required by these Technical Specifications.

5. Records of nuclear safety-related tests or experiments.

6. Records of radioactive shipments.

7. Records of changes to OPERATING procedures.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

10. Records of Quality Assurance activities required by the Operational Quality Assurance Program (OQAP) except where it is determined that the records should be maintained for a longer period of time.

b. The following records shall be retained for the duration of the Plant Operating License.

1. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.

2. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.

3. Records of plant radiation and contamination surveys.

4. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of transient or operational cycles for these facility components.

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of meetings of the JOSRC and PORC.

10. Records for environmental qualification.

11. Records of reviews performed for changes made to the ODCM and the PCP.

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 in-plant iodine concentrations under accident conditions. This program shall include the following:

1. Training of personnel
2. Procedures for monitoring
3. Provisions for maintenance of sampling and analysis equipment

6.12 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements.

b. Integrated leak test requirements for each system at a frequency not to exceed REFUELING cycle intervals.

6.13 HIGH RADIATION AREA

a. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is > 100 mrem/hr, but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a radiation work permit (RWP).[(1)] Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following.

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

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

   3. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

b. In addition to the requirements of 6.13.a., areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose > 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose > 1000 mrem[(2)] that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

[(1)] Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

[(2)] Measurement made at 30 centimeters from source of radioactivity.

6.15 SECONDARY WATER CHEMISTRY

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

6.16 RADIOLOGICAL EFFLUENTS

a. Written procedures shall be established, implemented and maintained covering the activities referenced below:

   1. Process Control Program (PCP) implementation

   2. OFF-SITE DOSE CALCULATION MANUAL (ODCM) implementation

   3. Quality Assurance Program for effluent and environmental monitoring

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

   1. Radioactive Effluent Controls Program

      A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program shall: (1) be contained in the ODCM, (2) be implemented by OPERATING procedures, and (3) include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

      A. Limitations on the OPERABILITY of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM.

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

      C. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM.

      D. Limitations on the annual and quarterly doses or dose commitment to a MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50.

      E. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.

F. Limitations on the OPERABILITY and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31 day period would exceed 2% of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

G. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1.

H. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

I. Limitations on the annual and quarterly doses to MEMBER(S) OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

J. Limitations on the annual dose or dose commitment to any MEMBER(S) OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

2. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide: (1) representative measurement of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall: (1) be contained in the ODCM (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

A. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

B. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census.

C. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

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 documented and records of reviews performed shall be retained as required by TS 6.10.b.11. The documentation shall contain:

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

      B. A determination that the change will maintain the overall conformance of the soldified waste product to existing requirements of Federal, State, or other applicable regulations.

   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 documented and records of reviews performed shall be retained as required by TS 6.10.b.11. This documentation shall contain:

      A. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change.

      B. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

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

   3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. The date the changes were made shall be indicated. In addition, a method such as redlining should be used to clearly identify the changes.

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 that 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 that 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.

    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 periodic USAR update.

6.20 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The provisions of TS 4.0.b do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The provisions of TS 4.0.c are applicable to the Containment Leakage Rate Testing Program.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident is less than the containment internal test pressure, P_a. The maximum allowable leakage rate (L_a) is 0.5 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig.

For penetrations which extend into the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.10L_a$. For penetrations which are exterior to both the shield building and the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.01L_a$. If leak rates are exceeded, repairs and retest shall be performed to demonstrate reduction of the combined leak rate to these values.

Leakage rate acceptance criteria:

a. The containment leakage rate acceptance criterion is $\leq 1.0L_a$.

b. Prior to unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.6L_a$ for Type B and C tests and $< 0.75L_a$ for the Type A test.

c. The personnel and emergency air lock leakage rates, when combined with the cumulative Type B and C leakage, shall be $< 0.6L_a$. For each air lock door seal, the leakage rate shall be $< 0.005L_a$ when tested to ≥ 10 psig.

6.21 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

The Bases Control Program shall be established, implemented and maintained. This program provides a means for processing changes to the bases of these Technical Specifications.

a. Changes to the bases of the TS shall be made under appropriate administrative controls and reviews.

b. Changes to bases may be made without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license.

   2. A change to the USAR or bases that requires NRC approval pursuant to 10 CFR 50.59.

c. Proposed changes that meet the criteria of 6.21.b.1 and 6.21.b.2 above shall be reviewed and approved by the NRC prior to implementation.

d. The Bases Control Program shall contain provisions to ensure that the bases are maintained consistent with the USAR.

e. Changes to the bases implemented without prior NRC approval shall be provided to the NRC on a frequency not to exceed that of 10 CFR 50.71(e).

TABLE TS 1.0-1

FREQUENCY NOTATIONS

NOTATION	FREQUENCY
Shift (S)	At least once per 12 hours
Daily (D)	At least once per 24 hours
Weekly (W)	At least once per 7 days
Monthly (M)	At least once per 31 days
Quarterly (Q)	At least once per 92 days
Semiannual (SA)	At least once per 184 days
Refueling (R)	At least once per 18 months
Reactor Startup (S/U)	Prior to each reactor startup
N.A.	Not Applicable

TABLE TS 3.1-2

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NO.	MAXIMUM[(1)(2)] ALLOWABLE LEAKAGE BASED ON NORMAL OPERATING PRESSURE
Reactor Vessel, Core Flooding Line (Upper Plenum Injection)	SI-304A	≤ 5.0 gallons per minute
	SI-303A	≤ 5.0 gallons per minute
	SI-304B	≤ 5.0 gallons per minute
	SI-303B	≤ 5.0 gallons per minute
Loop B 12" Accumulator Discharge Line	SI-22B	≤ 5.0 gallons per minute

(1) Leakage rates ≤ 1.0 gpm are considered acceptable.

Leakage rates > 1.0 gpm but ≤ 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

Leakage rates > 1.0 gpm but ≤ 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

Leakage rates greater than 5.0 gpm are considered unacceptable.

(2) Minimum test differential pressure shall not be < 150 psid.

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 0.745×10^6 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 4.4×10^6 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

TABLE TS 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
8	Containment Purge and Vent System Radiation Particulate Detector Radioactive Gas Detector	Containment ventilation isolation	≤ value of radiation levels in exhaust duct as defined in footnote [3]
9	Safeguards Bus Undervoltage [4]	Loss of power	85.0% ± 2% nominal bus voltage ≤ 2.5 seconds time delay
10	Safeguards Bus Second Level Undervoltage [5]	Degraded grid voltage	93.6% ± 0.9% of nominal bus voltage ≤ 7.4 seconds time delay

[3] The setting limits for max radiation levels are derived from ODCM Specification 3.4.1 and Table 2.2, and USAR Section 6.5.
[4] This undervoltage protection channel ensures ESF equipment will perform as assumed in the USAR.
[5] This undervoltage protection channel protects ESF equipment from long-term low voltage operation.

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Manual	2	1	1	-		Maintain HOT SHUTDOWN
2	Nuclear Flux Power Range[1]						
	Low setting	4	2	3	1	P-10	Maintain HOT SHUTDOWN
	High setting	4	2	3	1		
	Positive rate	4	2	3	1		
	Negative rate	4	2	3	1		
3	Nuclear Flux Intermediate Range	2	1	1	-	P-10	Maintain HOT SHUTDOWN[3]
4	Nuclear Flux Source Range	2	1	1	-	P-6	Maintain HOT SHUTDOWN[3]
5	Overtemperature ΔT	4	2	3	1		Maintain HOT SHUTDOWN
6	Overpower ΔT	4	2	3	1		Maintain HOT SHUTDOWN
7	Low Pressurizer Pressure	4	2	3	1	P-7	Maintain HOT SHUTDOWN
8	High Pressurizer Pressure	3	2	2	-		Maintain HOT SHUTDOWN

Amendment No. 137
06/09/98

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
9	Pressurizer High Water Level	3	2	2	-	P-7	Maintain HOT SHUTDOWN
10	Low Flow In One Loop	3/loop	2/loop (any loop)	2	-	P-8	Maintain HOT SHUTDOWN
	Low Flow Both Loops	3/loop	2/loop (both loops)	2	-	P-7	Maintain HOT SHUTDOWN
11	Deleted						
12	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	-		Maintain HOT SHUTDOWN
13	Undervoltage 4-kV	2/bus	1/bus (both buses)	1/bus	-	P-7	Maintain HOT SHUTDOWN
14	Underfrequency 4-kV Bus(4)	2/bus	1/bus (both buses)	1/bus	-		Maintain HOT SHUTDOWN
15	Deleted						
16	Steam Flow/Feedwater Flow Mismatch	2	1	1	-		Maintain HOT SHUTDOWN

Amendment No. 94
11/12/91

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

		1	2	3	4	5	6
NO.	**FUNCTIONAL UNIT**	**NO. OF CHANNELS**	**NO. OF CHANNELS TO TRIP**	**MINIMUM OPERABLE CHANNELS**	**MINIMUM DEGREE OF REDUNDANCY**	**PERMISSIBLE BYPASS CONDITIONS**	**OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET**
17	Reactor Trip Breaker (RTB)	2	1	2	-	The RTBs may be bypassed for up to 8 hrs. for surveillance testing or maintenance	Maintain HOT SHUTDOWN and open the RTBs
	(Independently Test Shunt and Undervoltage Trip Attachments)	2/bkr	1	2	-		After 72 hours maintain HOT SHUTDOWN and open the RTBs

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

<u>NOTES</u>

(1) One additional channel may be taken out of service for zero power physics testing.

(2) Deleted

(3) When a block condition exists, maintain normal operation.

(4) Underfrequency on the 4-kV buses trips the Reactor Coolant Pump breakers, which in turn trips the reactor when power is above P-7.

P-6 1 of 2 Intermediate Range Nuclear Instrument Channels Indicates > 10^{-5}% power

P-7 3 of 4 Power Range Nuclear Instrument channels < 10% power AND
2 of 2 Turbine Impulse Pressure Channels < 10% power

P-8 3 of 4 Power Range Nuclear Instrument Channels < 10% power

P-10 2 of 4 Power Range Nuclear Instrument Channels > 10% power

TABLE TS 3.5-3

EMERGENCY COOLING

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Safety Injection						
	a. Manual	2	1	1	-		HOT SHUTDOWN[1]
	b. High Containment Pressure	3	2	2	-		HOT SHUTDOWN[1]
	c. Low Steam Pressure/Line	3	2	2	-	Primary pressure < 2000 psig	HOT SHUTDOWN[1]
	d. Pressurizer Low Pressure	3	2	2	-	Primary pressure < 2000 psig	HOT SHUTDOWN[1]
2	Deleted						
3	Containment Spray						
	a. Manual	2	2	2	[2]		HOT SHUTDOWN[3]
	b. Hi-Hi Containment Pressure (Containment Spray)	3 sets of 2	1 of 2 in each set	1 per set	1/set		HOT SHUTDOWN[3]

(1) If minimum conditions are not met within 24 hours, steps shall be taken to place the plant in COLD SHUTDOWN condition.
(2) Must actuate 2 switches.
(3) If minimum conditions are not met within 24 hours, steps shall be taken to place the plant in COLD SHUTDOWN condition.

TABLE TS 3.5-3

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
4	Motor-Driven Auxiliary Feedwater Pumps						
	a. Either Steam Generator Lo-Lo Level	3/loop	2/loop	2/loop	-		Maintain HOT SHUTDOWN
	b. Loss of Main Feed Water[4]	1	1	1			Maintain HOT SHUTDOWN
	c. Safety Injection	(Refer to Item 1 of this table)					
	d. 4 KV Buses 1-5 and 1-6 under voltage	2/bus[5]	1/bus	1/bus[6]			Maintain HOT SHUTDOWN or operate diesel generators
5	Turbine-Driven Auxiliary Feedwater Pumps						
	a. Both Steam Generator Lo-Lo Level	3/loop	2/loop	2/loop	-		Maintain HOT SHUTDOWN
	b. 4 KV Buses 1-1 and 1-2 under voltage	(Refer to Item 13 of Table TS 3.5-2)					

[4] Tripping of both main feedwater pump breakers starts both motor-driven auxiliary feedwater pumps.
[5] Each channel consists of one instantaneous and one time-delay relay connected in series.
[6] When one component of a channel is taken out of service, that component shall be in the tripped condition.

TABLE TS 3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Containment Isolation						
	a. Safety Injection	Refer to Item No. 1 of Table TS 3.5-3					HOT SHUTDOWN[1]
	b. Manual	2	1	1	-		HOT SHUTDOWN
2	Steam Line Isolation						
	a. Hi-Hi Steam Flow with Safety Injection	2/loop	1	1	-		HOT SHUTDOWN[1]
	b. Hi Steam Flow and 2 of 4 Lo-Lo T_{avg} with Safety Injection	2/loop	1	1	-		HOT SHUTDOWN[1]
	c. Hi-Hi Containment Pressure	3	2	2	-		HOT SHUTDOWN[1]
	d. Manual	1/loop	1/loop	1/loop	-		HOT SHUTDOWN

[1] If minimum conditions are not met within 24 hours, steps shall be taken to place the plant in a COLD SHUTDOWN condition.

TABLE TS 3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
3	Containment Ventilation Isolation						
	a. High Containment Radiation	2	1	1	-	-	These channels are not required to activate containment ventilation isolation when the containment purge and ventilation system isolation valves are maintained closed.[2]
	b. Safety Injection	Refer to Item 1 of Table TS 3.5-3					
	c. Containment Spray	Refer to Item 3 of Table TS 3.5-3					
4	Main Feedwater Isolation						
	a. Hi-Hi Steam Generator Level	3	2	2	1		HOT SHUTDOWN

[2] The detectors are required for Reactor Coolant System leak detection as referenced in TS 3.1.d.5.

TABLE TS 3.5-5

INSTRUMENT OPERATION CONDITIONS FOR SAFEGUARDS BUS POWER SUPPLY FUNCTIONS

		1	2	3	4	5	6
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Safeguards Bus Undervoltage	2/bus[1]	1/bus	1/bus[2]	–		Maintain HOT SHUTDOWN or operate the diesel generator
2	Safeguards Bus Second Level Undervoltage	1/bus[3]	1/bus	–	–		When one of the two time-delay relays is out of service, place that relay in the tripped condition

[1] Each channel consists of one instantaneous and one time-delayed relay connected in series.
[2] When one component of a channel is taken out of service, that component shall be in the tripped condition.
[3] Each channel consists of two time-delayed relays connected in series.

TABLE TS 3.5-6

ACCIDENT MONITORING INSTRUMENTATION OPERATING CONDITIONS FOR INDICATION

NO.	FUNCTIONAL UNIT	REQUIRED TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE[1]
1	Auxiliary Feedwater Flow to Steam Generators (Narrow Range Level Indication Already Required OPERABLE by Table TS 3.5-2, Item 12)	1/steam generator[2]	
2	Reactor Coolant System Subcooling Margin	2[2]	1
3	Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve[2]	1/valve
4	Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve[2]	1/valve
5	Pressurizer Safety Valve Position (One Channel Temperature, and One Acoustic Sensor per Valve)	2/valve[2]	1/valve
6	Containment Water Level (Wide Range)	2[2]	1
7	Containment Hydrogen Monitor	2[3]	1
8	Containment Pressure Monitor (Wide Range)	2[2]	1
9	Reactor Vessel Level Indication	2[4]	1
10	Core Exit Thermocouples[5]	4 thermocouple/core quadrant[4]	2 thermocouple/core quadrant[6]
11	Steam Generator Level (Wide Range)	2/steam generator[4]	1/steam generator

[1] With the number of OPERABLE accident monitoring instrumentation channels less than the minimum channels OPERABLE requirements, either restore the minimum number of channels to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

[2] With the number of OPERABLE accident monitoring instrumentation channels less than the required total number of channels shown, either restore the inoperable channels to OPERABLE status within 14 days, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

TABLE TS 3.5-6

ACCIDENT MONITORING INSTRUMENTATION OPERATING CONDITIONS FOR INDICATION

[3] With the number of OPERABLE accident monitoring instrumentation channels less than the required total number of channels shown, either restore the inoperable channels to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

[4] With the number of OPERABLE accident monitoring instrumentation channels less than the required total number of channels shown, either restore the inoperable channels to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

[5] Refer also to TS 3.11.c and TS 3.11.d

[6] For the purposes of accident monitoring instrumentation, thermocouples on the axis may be included in either adjacent quadrant.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
1. Nuclear Power Range	Each shift(a) Effective Full Power Month(c)	Daily(a) Effective Full Power Quarter(c)	Monthly(b) Quarterly(d)	(a) Heat balance (b) Signal to ΔT; bistable action (permissive, rod stop, trips) (c) Upper and lower chambers for axial off-set using incore detectors. The check and calibration for axial offset shall also be performed prior to > 75% power following any core alteration. (d) Permissives P8 and P10 and the 25% reactor trip are tested quarterly.
2. Nuclear Intermediate Range	Each shift(a,c)	Not applicable	Prior to each startup if not done previous week(b)	(a) Once/shift when in service (b) Log level; bistable action (permissive, rod stop, trips) (c) Channel check required in all plant modes
3. Nuclear Source Range	Each shift(a,c)	Not applicable	Prior to each startup if not done previous week(b)	(a) Once/shift when in service (b) Bistable action (alarm, trips) (c) Channel check required in all plant modes
4. Reactor Coolant Temperature	Each shift (c)	Each refueling cycle	Monthly(a) Monthly(b)	(a) Overtemperature ΔT (b) Overpower ΔT (c) Channel check not required below HOT SHUTDOWN
5. Reactor Coolant Flow	Each shift	Each refueling cycle	Monthly	

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
6. Pressurizer Water Level	Each shift	Each refueling cycle	Monthly	
7. Pressurizer Pressure	Each shift	Each refueling cycle	Monthly	
8. a. 4-KV Voltage and Frequency	Not applicable	Each refueling cycle	Monthly	Reactor protection circuits only
b. 4-KV Voltage (Loss of Voltage)	Not applicable	Each refueling cycle	Monthly	Safeguards buses only
c. 4-KV Voltage (Degraded Grid)	Not applicable	Each refueling cycle	Monthly	Safeguards buses only
9. Analog Rod Position	Each shift(a,b)	Each refueling cycle	Each refueling cycle	(a) With step counters (b) Following rod motion in excess of 24 steps when computer is out of service
10. Rod Position Bank Counters	Each shift(a,b)	Not applicable	Each refueling cycle	(a) With analog rod position (b) Following rod motion in excess of 24 steps when computer is out of service

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
11. a. Steam Generator Low Level	Each shift	Each refueling cycle	Monthly	
b. Steam Generator High Level	Each shift	Each refueling cycle	Monthly	
12. Steam Generator Flow Mismatch	Each shift	Each refueling cycle	Monthly	
13. Deleted				
14. Residual Heat Removal Pump Flow	Each shift (when in operation)	Each refueling cycle	Not applicable	
15. Deleted				
16. Refueling Water Storage Tank Level	Weekly	Annually	Not applicable	
17. Deleted				

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
18. a. Containment Pressure (SIS signal)	Each shift	Each refueling cycle	Monthly(a)	(a) Isolation Valve Signal
b. Containment Pressure (Steamline Isolation)	Each shift(a)	Each refueling cycle(a)	Monthly(a)	(a) Narrow range containment pressure (-3.0, +3.0 psig excluded)
c. Containment Pressure (Containment Spray Act)	Each shift	Each refueling cycle	Monthly	
d. Annulus Pressure (Vacuum Breaker)	Not applicable	Each refueling cycle	Each refueling cycle	
19. Radiation Monitoring System	Daily (a,b)	Each refueling cycle (a)	Quarterly (a)	(a) Includes only channels R11 thru R15, R19, R21, and R23 (b) Channel check required in all plant modes
20. Deleted				
21. Containment Sump Level	Not applicable	Not applicable	Each refueling cycle	
22. Accumulator Level and Pressure	Each shift	Each refueling cycle	Not applicable	
23. Steam Generator Pressure	Each shift	Each refueling cycle	Monthly	

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
24. Turbine First Stage Pressure	Each shift	Each refueling cycle	Monthly	
25. Portable Radiation Survey Instruments	Monthly (a)	Annually	Quarterly	(a) Channel check required in all plant modes
26. Protective System Logic Channel Testing	Not applicable	Not applicable	Monthly	Includes auto load sequencer
27. Deleted				
28. Deleted				
29. Seismic Monitoring System	Each refueling cycle	Each refueling cycle	Not applicable	
30. Fore Bay Water Level	Not applicable	Each refueling cycle	Each refueling cycle	
31. AFW Flow Rate	(a)	Each refueling cycle	Not applicable	(a) Flow rate indication will be checked at each unit startup and shutdown
32. PORV Position Indication	Monthly	Each refueling cycle	Not applicable	
a. Back-up (Temperature)	Monthly	Each refueling cycle	Not applicable	
33. PORV Block Valve Position Indicator	Monthly	Each refueling cycle	Not applicable	

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
34. Safety Valve Position Indicator (Acoustic)	Monthly	Each refueling cycle	Not applicable	
a. Back-up (Temperature)	Monthly	Each REFUELING cycle	Not applicable	
35. FW Pump Trip (AFW Initiation)	Not applicable	Not applicable	Each refueling cycle	
36. Reactor Coolant System Subcooling Monitor	Monthly	Each refueling cycle	Each refueling cycle	
37. Containment Pressure (Wide Range)	Daily	Each refueling cycle	Not applicable	
38. Deleted				
39. Containment Water Level (Wide Range)	Not applicable	Not applicable	Each refueling cycle	
40. Reactor Vessel Level Indication	Monthly	Each refueling cycle	Not applicable	
41. Core Exit Thermocouples	Monthly	Each refueling cycle	Not applicable	
42. Steam Generator Level (Wide Range)	Monthly	Each refueling cycle	Not applicable	

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Each refueling cycle	
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY
1. Reactor Coolant Samples	a. Gross Radioactivity Determination (excluding tritium)	5/week[1]
	b. DOSE EQUIVALENT I-131 Concentration	1/14 days[2]
	c. Tritium activity	Monthly
	d. Chemistry (Cl, F, O_2)[3]	3/week[4]
	e. $\bar{E}$ Determination	1/6 months[5]
	f. RCS Isotopic analysis for Iodine	Once per 4 hours in accordance with TS 3.1.c.2.C.
2. Reactor Coolant Boron[6]	Boron Concentration[3]	2/week

[1] Maximum time between tests is 3 days.
[2] Sample required only when in the OPERATING MODE.
[3] Test required in all plant modes.
[4] Maximum time between tests is 4 days.
[5] Sample after a minimum of 2 EFPD and 20 days of OPERATING MODE operation have elapsed since the reactor was last subcritical for ≥ 48 hours.
[6] A reactor coolant boron concentration sample does not have to be taken when the core is completely unloaded.

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.

TABLE TS 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

EQUIPMENT TESTS[1]	TEST	FREQUENCY
1. Control Rods	od drop times of all full length rods Partial movement of all rods not fully inserted in the core	Each REFUELING outage Quarterly when at or above HOT STANDBY
1a. Reactor Trip Breakers	ndependent test[2] shunt and undervoltage trip attachments	Monthly
1b. Reactor Coolant Pump Breakers-Open-Reactor Trip	PERABILITY	Each REFUELING outage
1c. Manual Reactor Trip	pen trip reactor[3] trip and bypass breaker	Each REFUELING outage
2. Deleted		
3. Deleted		
4. Containment Isolation Trip	PERABILITY	Each REFUELING outage
5. Refueling System Interlocks	PERABILITY	Prior to fuel movement each REFUELING outage
6. Deleted		
7. Deleted		
8. RCS Leak Detection	PERABILITY	Weekly[4]
9. Diesel Fuel Supply	uel Inventory[5]	Weekly
10. Deleted		
11. Fuel Assemblies	isual Inspection	Each REFUELING outage
12. Guard Pipes	isual Inspection	Each REFUELING outage
13. Pressurizer PORVs	PERABILITY	Each REFUELING cycle
14. Pressurizer PORV Block Valves	PERABILITY	Quarterly[6]
15. Pressurizer Heaters	PERABILITY[7]	Each REFUELING cycle
16. Containment Purge and Vent Isolation Valves	PERABILITY[8]	Each REFUELING cycle

(1) Following maintenance on equipment that could affect the operation of the equipment, tests should be performed to verify OPERABILITY.
(2) Verify OPERABILITY of the bypass breaker undervoltage trip attachment prior to placing breaker into service.
(3) Using the Control Room push-buttons, independently test the reactor trip breakers shunt trip and undervoltage trip attachments. The test shall also verify the undervoltage trip attachment on the reactor trip bypass breakers.
(4) When reactor is at power or in HOT SHUTDOWN condition.
(5) Inventory of fuel required in all plant modes.
(6) Not required when valve is administratively closed.
(7) Test will verify OPERABILITY of heaters and availability of an emergency power supply.
(8) This test shall demonstrate that the valve(s) close in $\leq$ 5 seconds.

TABLE TS 4.2-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G. (2)	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G. (2)	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., (2) plug defective tubes and inspect 2S tubes in the other S.G. (2) Prompt notification of the Commission. (1)	The other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in other S.G. and plug defective tubes. Prompt notification of the Commission. (1) (2)	N/A	N/A

S = 6%/n Where n is the number of steam generators inspected during an inspection.

Notes:

1. Refer to Specification 4.2.b.7.c
2. As allowed by TS 4.2.b.2.d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those portions of the tubes where imperfections were previously found.

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APPENDIX A

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Note:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

Nuclear Safety Related	☑ Yes ☐ No	PORC Review Required	☑ Yes ☐ No	SRO Approval Of Temporary Changes Required	☑ Yes ☐ No

1.0 Purpose

1.1 This procedure provides instruction for calibrating the Refueling Water Level Indication System Transmitters.

1.2 The Reactor Coolant System (RCS) transports heat from the Reactor Core to the Steam Generators to produce steam. The Refueling Water Level Indication System provides indication only.

2.0 Operational Considerations

2.1 Initial Conditions

2.1.1 This procedure may be performed any time containment entry is possible, but is normally performed during a refueling outage.

2.1.2 Perform this procedure before entering Mid Loop operations.

2.2 Control Room Indications

2.2.1 There are no Control Room alarms associated with this procedure.

2.2.2 The following Control Room instruments will be unavailable while performing this procedure:

- 41337 Reactor Cavity Level Indicator
- L9053A Refueling Water LVL A WR Computer Point
- L9054A Refueling Water LVL B WR Computer Point
- L9055A Refueling Water LVL NR Computer Point

2.3 Technical Specifications And Limiting Conditions For Operation

2.3.1 None.

2.4 Commitments

- 87-146 Install Independent Refueling Water Level Indication System.

3.0 Precautions

3.1 Work under this procedure is performed in a radiological controlled area. Obtain a Radiation Work Permit (RWP).

3.2 This system contains fluids under high pressure.

3.3 This system contains fluids under high temperature.

3.4 The transmitters share a common sensing line with ICCMS Reactor Vessel Level transmitters 21152, 21153, 21154 and 21155. Restoring the Refueling Water transmitters to service may cause a momentary loss of ICCMS Reactor Vessel Level indication.

3.5 Observe Foreign Material Exclusion as required by NAD 1.36 "Foreign Material Exclusion."

4.0 General Instructions

4.1 <u>Overview</u>

4.1.1 This procedure requires two technicians.

4.1.2 The following equipment is affected by this procedure:

- 21158 — Refueling Water Level Narrow Range Transmitter, Containment Bldg., 130°/30', 611'
- 21159 — Refueling Water Level B Wide Range Transmitter, Containment Bldg., 130°/30', 611'
- 24068 — Refueling Water Level A Wide Range Transmitter, Containment Bldg., 239°/18', 596'

4.1.3 The instruments affected by this procedure are independent devices. Instructions may be performed without modification when performed under a work request.

4.1.4 A 10 ohm ±0.01% resistor is temporarily installed at the transmitter. Actual measurements will be in volts.

4.1.5 If an actual result differs from a required result, immediately notify the Shift Supervisor before continuing. This is basis for test failure.

4.1.6 Complete a Kewaunee Assessment Process (KAP) Form for nonconforming conditions as required by NAD 12.2, Surveillance Procedures.

4.1.7 Underlined step numbers designate steps requiring independent verification of performance. A second person shall independently verify completing the step.

4.2 References

4.2.1 Drawings

- Flow XK-100-10
- Logic E-2041
- Interconnection XK-390-4, XK-15064-9
- Plumbing DM 17.6-712, DM 17.6-714

4.2.2 Technical manuals and instruction books

- 21158, 21159 2566-1, Rosemount
- 24068 390-183, Foxboro

4.2.3 Bill of materials

- None

5.0 Equipment Required

5.1 Measuring And Test Equipment

- Fluke 45 precision voltmeter (2)
- Precision test resistor
- Deadweight Tester/Calibrator (0 to 1000 In-H_2O)

5.2 Protective And Safety Equipment

- As required by WPSC Safety Rule Book

DATE PERFORMED ________ INITIALS

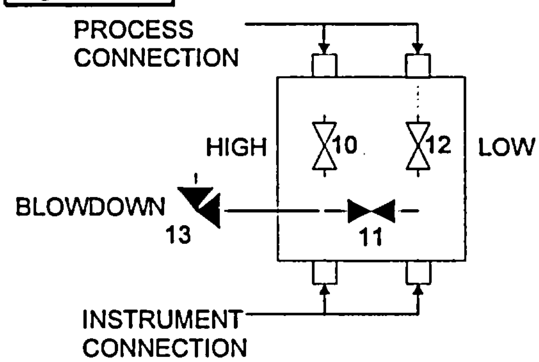
6.0 Procedure

6.1 LT-21158

Note

Figure 1shows valves and connections for LT-21158. Valve 10 represents the HP side and valve 12 the LP side.

Figure 1



6.1.1 Isolate LT-21158:

6.1.1.1 Close valve 12. ______

6.1.1.2 Close valve 10. ______

6.1.2 Open valve 11. ______

6.1.3 Drain the transmitter. ______

6.1.4 Close valve 11. ______

6.1.5 At LT-21158, terminal (+), lift field wire RVLNR-P. ______

6.1.6 Between terminal (+) and lifted field wire RVLNR-P, connect a precision voltmeter and test resistor. ______

6.1.7 Connect the deadweight tester/calibrator. ______

DATE PERFORMED __________ INITIALS

6.1.8 At Relay Rack RR-162, Nest 1 Slot 1 (I/E 48093001), input terminal pair 1 Termination Module, connect a precision voltmeter. ______

6.1.9 Using Table 6-1, verify loop integrity by recording the voltage at the transmitter and Relay Rack RR-162. ______

Table 6-1

Output greater than 180 mvdc (18 madc)	
Transmitter output (LT-21158)	mvdc
Rack (RR-162, Nest 1 Slot 1 terminal pair 1 termination module)	mvdc
Difference between transmitter and rack	mvdc
Max allowable difference	**0.20 mvdc**

M&TE	**DESCRIPTION**
	Fluke 45 precision voltmeter*
	Fluke 45 precision voltmeter*
	Precision test resistor

**300 mvdc medium resolution*

DATE PERFORMED ________ INITIALS

6.1.10 Using Table 6-2, perform a calibration test of LT-21158. ______

Table 6-2

Input ($In-H_2O$)		Output (mvdc)			LT-21158
Test Tee				Accuracy: ± 0.32 mvdc	
		DESIRED	AS FOUND	ACCEPTANCE RANGE	FINAL
1	74	41.30		40.98 to 41.62	
2	82	82.60		82.28 to 82.92	
3	90	123.90		123.58 to 124.22	
4	97	160.00		159.68 to 160.32	
5	105	201.30		200.98 to 201.62	
5	105	201.30		200.98 to 201.62	
4	97	160.00		159.68 to 160.32	
3	90	123.90		123.58 to 124.22	
2	82	82.60		82.28 to 82.92	
1	74	41.30		40.98 to 41.62	

M&TE ID	DESCRIPTION
	Deadweight tester/Calibrator
	Precision voltmeter
	Precision test resistor

6.1.11 Open valve 11. ______

6.1.12 Remove all test equipment. ______

<u>6.1.13</u> At LT-21158, terminal (+), land field wire RVLNR-P. ______

Independent Verification ______

<u>6.1.14</u> Verify valve 13 is closed. ______

Independent Verification ______

DATE PERFORMED ________ INITIALS

6.1.15 Slowly open valve 10. ______

Independent Verification ______

6.1.16 Close valve 11. ______

Independent Verification ______

6.1.17 Slowly open valve 12. ______

Independent Verification ______

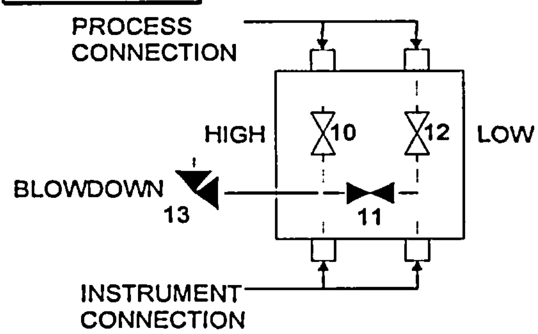
DATE PERFORMED __________ INITIALS

6.2 LT-21159

Note

Figure 2 shows valves and connections for LT-21159. Valve 10 represents the HP side and valve 12 the LP side.

Figure 2



6.2.1 Isolate LT-21159:

6.2.1.1 Close valve 12. ______

6.2.1.2 Close valve 10. ______

6.2.2 Open valve 11. ______

6.2.3 Drain the transmitter. ______

6.2.4 Close valve 11. ______

6.2.5 At LT-21159, terminal (+), lift field wire RVLWR-P. ______

6.2.6 Between terminal (+) and lifted field wire RVLWR-P, connect a precision voltmeter and test resistor. ______

6.2.7 Connect the deadweight tester/calibrator. ______

DATE PERFORMED __________ INITIALS

6.2.8 At Relay Rack RR-162, Nest 1 Slot 1 (I/E 48093002), input terminal pair 2 Termination Module, connect a precision voltmeter. _______

6.2.9 Using Table 6-3, verify loop integrity by recording the voltage at the transmitter and Relay Rack RR-162. _______

Table 6-3

Output greater than 180 mvdc (18 madc)	
Transmitter output (LT-21159)	mvdc
Rack (RR-162, Nest 1 Slot 1 terminal pair 2 termination module)	mvdc
Difference between transmitter and rack	mvdc
Max allowable difference	**0.20 mvdc**

M&TE	DESCRIPTION
	Fluke 45 precision voltmeter*
	Fluke 45 precision voltmeter*
	Precision test resistor

**300 mvdc medium resolution*

DATE PERFORMED ________ INITIALS

6.2.10 Using Table 6-4, perform a calibration test of LT-21159. ______

Table 6-4

Input (In-H_2O)		Output (mvdc)			LT-21159
Test Tee				Accuracy: ± 0.32 mvdc	
		DESIRED	AS FOUND	ACCEPTANCE RANGE	FINAL
1	15	40.10		39.78 to 40.42	
2	178	80.20		79.88 to 80.52	
3	340	120.10		119.78 to 120.42	
4	503	160.20		159.88 to 160.52	
5	659	198.60		198.28 to 198.92	
5	659	198.60		198.28 to 198.92	
4	503	160.20		159.88 to 160.52	
3	340	120.10		119.78 to 120.42	
2	178	80.20		79.88 to 80.52	
1	15	40.10		39.78 to 40.42	

M&TE ID	DESCRIPTION
	Deadweight tester/Calibrator
	Precision voltmeter
	Precision test resistor

6.2.11 Open valve 11. ______

6.2.12 Remove all test equipment. ______

6.2.13 At LT-21159, terminal (+), land field wire RVLWR-P. ______

Independent Verification ______

6.2.14 Verify valve 13 is closed. ______

Independent Verification ______

6.2.15 Slowly open valve 10. ______

Independent Verification ______

DATE PERFORMED ________ INITIALS

6.2.16 Close valve 11. ______

Independent Verification ______

6.2.17 Slowly open valve 12. ______

Independent Verification ______

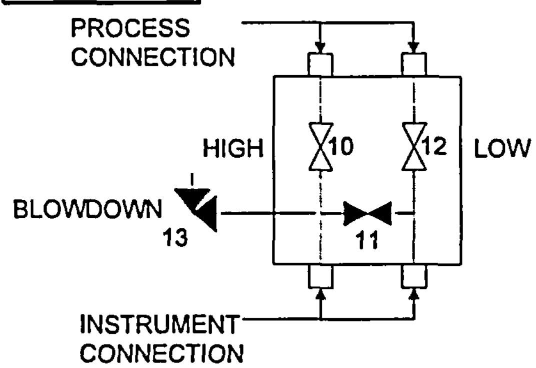
DATE PERFORMED __________ INITIALS

6.3 LT-24068

Note

Figure 3 shows valves and connections for LT-24068. Valve 10 represents the HP side and valve 12 the LP side.

Figure 3



6.3.1 Isolate LT-24068:

6.3.1.1 Close valve 12. ______

6.3.1.2 Close valve 10. ______

6.3.2 Open valve 11. ______

6.3.3 Drain the transmitter. ______

6.3.4 Close valve 11. ______

6.3.5 Connect the required test equipment. ______

6.3.6 At Relay Rack RR-163, TP/DP24068, connect a precision voltmeter. ______

DATE PERFORMED ________ INITIALS

6.3.7 Using Table 6-5, verify loop integrity by recording the voltage at the transmitter and Relay Rack RR-163. ______

Table 6-5

Output greater than 450 mvdc (45 madc)		
Transmitter output (LT-24068)		mvdc
Rack (RR-163, TP/DP24068)		mvdc
Difference between transmitter and rack		mvdc
Max allowable difference		**0.67 mvdc**
M&TE	**DESCRIPTION**	
	Fluke 45 precision voltmeter*	
	Fluke 45 precision voltmeter*	

**1000 mvdc slow resolution*

DATE PERFORMED __________ INITIALS

6.3.8 Using Table 6-6, perform a calibration test of LT-24068. _______

Table 6-6

Input (In-H_2O)		Output (mvdc)			LT-24068
Test Tee				Accuracy: ± 3.0 mvdc	
		DESIRED	AS FOUND	ACCEPTANCE RANGE	FINAL
1	200	100.0		97.0 to 103.0	
2	362	200.0		197.0 to 203.0	
3	525	300.0		297.0 to 303.0	
4	687	400.0		397.0 to 403.0	
5	850	500.0		497.0 to 503.0	
5	850	500.0		497.0 to 503.0	
4	687	400.0		397.0 to 403.0	
3	525	300.0		297.0 to 303.0	
2	362	200.0		197.0 to 203.0	
1	200	100.0		97.0 to 103.0	

M&TE ID	DESCRIPTION
	Deadweight tester/Calibrator
	Precision voltmeter

6.3.9 Open valve 11. _______

6.3.10 Remove all test equipment. _______

6.3.11 Verify valve 13 is closed. _______

Independent Verification _______

6.3.12 Slowly open valve 10. _______

Independent Verification _______

DATE PERFORMED _________ INITIALS

6.3.13 Close valve 11. ______

Independent Verification ______

6.3.14 Slowly open valve 12. ______

Independent Verification ______

6.4 Notify the Control Room Operators of your completing this procedure. ______
(Operator)

7.0 Acceptance Criteria

Reactor Condition (power level):		
Channels Satisfactorily Tested		
Channel	Yes	No
21158		
21159		
24068		

M&TE ID	DESCRIPTION	CAL DUE
	Fluke 45 precision voltmeter	
	Fluke 45 precision voltmeter	
	Precision test resistor	
	Precision voltmeter	
	Precision voltmeter	
	Deadweight tester/calibrator	

7.1 This test is complete and acceptable once the following items have been completed and signed.

7.1.1 Section 6.0 has been completed.

7.1.2 Independent Verifications have been completed.

7.1.3 All problems encountered during the test have been properly resolved and signed, except as noted by Kewaunee Assessment Process (KAP) Forms.

7.2 Were all As Found data entries within the allowable range?

☐ Yes ☐ No Initials: _____ ☐ N/A

7.3 Are all Final data entries within the allowable range?

☐ Yes ☐ No Initials: _____ ☐ N/A

7.4 Were any problems or discrepancies noted while performing this procedure?

☐ Yes KAP No. ______________ ☐ No

8.0 Review

8.1 Performed by: ______________________ Initials: _____ Date: __________

______________________ Initials: _____ Date: __________

8.2 Shift Supervisor: ______________________ Date: __________

Group Supervisor: ______________________ Date: __________

RM-45	ANNUNCIATOR NUMBER:	47011-B

SER PT:
See Comments

COMPUTER PT:
G0001G - G0023G

	A	B	C	D	E	F	G	H	I	J	K	L	M
1		X											
2													
3													
4													
5													

BETA WINDOWBOX 01

RADIATION INDICATION HIGH

INSTRUMENT:
Radiation Channels 1-23, and RE-29026

SETPOINT:
Setpoints Listed in E-2021

RECOMMENDED ACTION:

1. IF alarm is NOT due to a planned evolution , THEN INFORM Health Physics and request assistance in identifying and isolating radiation source.

2. GO TO A-RM-45.

COMMENTS:

1. SER Points:

206 - R1 Control Room Radiation High
207 - R2 Containment Radiation High
209 - R4 Charging Pump Room Radiation High
210 - R5 Spent Fuel Pool Area Radiation High

211 - R6 Sampling Room Radiation High
212 - R7 In-Core Seal Table Area Radiation High
214 - R9 Letdown Line Area Radiation High
215 - R10 New Fuel Pit Radiation High

216 - R11 Cntmt Air Part Mon Radiation High
217 - R12 Cntmt Gaseous Mon Radiation High
218 - R13 Aux Building Vent Radiation High

219 - R14 Aux Building Vent Radiation High
220 - R15 Cond Air Ejector Radiation High
221 - R16 Cntmt Fan Coils SW Radiation High
222 - R17 Component Cooling Radiation High

223 - R18 Liq Waste Disposal Radiation High
224 - R19 S/G BD Sample Radiation High
225 - R20 Aux Bldg HX SW Radiation High
226 - R21 Cntmt Stack Mon Radiation High

227 - R22 RHR Pump Pit Radiation High
228 - R23 C/R Ventilation Radiation High
485 - R39 Sludge Intercept Filter Area Rad High

REFERENCES:
FLOW: None
LOGIC E-2019, E-2021
OTHER: E-2013, E-2018, E-3745

NUCLEAR SAFETY RELATED	YES [ ] NO [X]	PORC REVIEW REQUIRED	YES [ ] NO [X]	SRO APPROVAL OF TEMPORARY CHANGES REQUIRED	YES [X] NO [ ]

REVIEWED BY [signature] DATE 10/28/97

APPROVED BY [signature] DATE ~~10/28/97~~

REV DATE 10/28/97
REV D

MI-87	ANNUNCIATOR NUMBER: 47023-K

SER PT:
See Comments

COMPUTER PT:
None

INSTRUMENT:
See Comments

SEISMIC TROUBLE

SETPOINT:
See Comments

RECOMMENDED ACTION:

1. DISPATCH the STA to Relay Room panel RR-159 to determine extent of seismic activity.

2. On RR-159, VERIFY alarm by checking activation of seismic recorder with TRIGGER, OBE OR DBE lights lit.

3. VERIFY actual seismic event as indicated by either of the following:

   a. Actual physical ground shaking.
   b. UW - Milwaukee Seismic Center verification (telephone numbers are in ETD 02, Kewaunee Nuclear Plant Emergency Telephone Directory, Private Agency call list).

4. WHEN event is complete, THEN on SWP-300 drawer at top of RR-159, PRESS RESET button to allow detection of subsequent events.

5. IF SER Point 333 is activated OR alarm is result of a monitor malfunction, PERFORM the following:

   a. INFORM Point Beach that Kewaunee Seismic Monitor is out-of-service. [PCR015806]
   b. CHECK Point Beach Seismic Monitor is in-service. [PCR015806]
   c. REQUEST Shift Manager review EPIP-AD-02 Chart M for impact on ability to determine event classifications. [PCR014701]

6. GO TO E-0-05.

CONTINUED

REFERENCES:
FLOW: None
LOGIC: None
OTHER: USAR 4.1.4, M-749, XK-992-35-2, 3
Tech Manual 99235-1

NUCLEAR SAFETY RELATED	PORC REVIEW REQUIRED	SRO APPROVAL OF TEMPORARY CHANGES REQUIRED
YES ☐ NO ☒	YES ☐ NO ☒	YES ☒ NO ☐

REVIEWED BY [signature] DATE AUG 0 5 2004

APPROVED BY [signature] DATE AUG 0 5 2004

REV DATE 8/05/04
REV E

MI-87	ANNUNCIATOR NUMBER: 47023-K

SER PT:
See Comments

COMPUTER PT:
None

INSTRUMENT:
See Comments

SEISMIC TROUBLE

SETPOINT:
See Comments

(continued)

COMMENTS:

SER POINTS	RR-159 Lights	INSTRUMENT	SETPOINT
330-Seismic Monitor Event	Seismic System TRIGGER	28081	.03g Horiz/Vert Ground Acceleration
331-Seismic Monitor Operational Basis Earthquake	OBE Limit Exceeded	28081	.06g Horiz, .04g Vert Ground Acceleration
332-Seismic Monitor Design Basis Earthquake	DBE Limit Exceeded	28081	.12g Horiz, .08g Vert Ground Acceleration
333-Seismic Monitor Loss of AC power	Facility AC Power (Lights Off)	BRB-123 Ckt 15	

REFERENCES:
FLOW: None
LOGIC: None
OTHER: USAR 4.1.4, M-749, XK-992-35-2, 3
Tech Manual 99235-1

NUCLEAR SAFETY RELATED	PORC REVIEW REQUIRED	SRO APPROVAL OF TEMPORARY CHANGES REQUIRED
YES ☐ NO ☒	YES ☐ NO ☒	YES ☒ NO ☐

REVIEWED BY [signature] DATE AUG 05 2004

APPROVED BY [signature] DATE AUG 05 2004

REV DATE 8/05/04
REV E

CW-04	ANNUNCIATOR NUMBER: 47051-N

SER PT:
See Comments

COMPUTER PT:
L9075A

INSTRUMENT:
LC-26829 LC-26831
LC-26830 LC-26832

FOREBAY LEVEL

LOW

SETPOINT:
See Comments

RECOMMENDED ACTION:

1. GO TO E-CW-04.

COMMENTS:

1. SER Points:

   278 - Forebay Level Low
   279 - Forebay Level Low Low

2. Setpoints:

   a. Forebay level low: 1/4 Channel < 47% (567.5')
   b. Forebay Level low low: 1/4 Channels < 42% (566')

3. 2/4 Channels < 42% will cause the running Circ Water pumps to trip.

REFERENCES:
FLOW: M-215 OTHER: E-2184
LOGIC: E-1614

NUCLEAR SAFETY RELATED	YES ☐ NO ☒	PORC REVIEW REQUIRED	YES ☐ NO ☒	SRO APPROVAL OF TEMPORARY CHANGES REQUIRED	YES ☒ NO ☐

REVIEWED BY [signature] DATE APR 11 2002

APPROVED BY [signature] DATE APR 11 2002

REV DATE
04/11/02
REV C

TB-54	ANNUNCIATOR NUMBER: 47051-W

SER PT:
359

COMPUTER PT:
P0300A

INSTRUMENT:
PS-16039

CONDENSER

VACUUM

LOW

SETPOINT:
5" Hg Absolute

RECOMMENDED ACTION:

1. IF Reactor trips, GO TO E-0.
2. GO TO E-AR-09.

COMMENTS:

None

REFERENCES:
FLOW: None
LOGIC: E-2057
OTHER: E-1469

NUCLEAR SAFETY RELATED	YES ☐ NO ☒	PORC REVIEW REQUIRED	YES ☐ NO ☒
SRO APPROVAL OF TEMPORARY CHANGES REQUIRED	YES ☒ NO ☐		

REVIEWED BY [signature] DATE SEP 27 2001

APPROVED BY [signature] DATE SEP 27 2001

REV DATE
09/27/01
REV B

Design Description
DCR-1811
Refueling Containment Loop Seal

When containment system integrity is not required during the refueling mode, containment penetration 42N will be used as a containment cable passthrough. During fuel shuffle no direct air path from containment atmosphere to the outside atmosphere is permitted. The penetration is normally used as a containment vessel pressurization test line. During normal full power operation it is sealed with a blind flange on the inside of containment.

The loop seal will consist of ten inch OD piping, mounted on the auxiliary building operating floor, 633 foot elevation. It will incorporate a twenty-four inch water column seal, an approximate 475 pound piping weight and a three-piece construction to facilitate handling and cable pulling. Provisions will be made for draining the loop seal and for possible overflow. At the end of refueling the loop seal will be broken down for storage and penetration 42N will be returned to its normal configuration for full power operation.

MJOrtmayer
01/06/87

COPY

Safety Evaluation Basis

DCR-1811
Refueling Containment Loop Seal

The containment vessel pressurization test line and penetration 42N is shown in the FSAR on Figure 11.1-4.

Technical Specifications 3.8.A.1 states, "In addition, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmoshpere shall have a closed isolation valve or an operable automatic isolation valve." The modification will utilize a 24 inch water loop seal thus preventing a direct air path from containment to the outside.

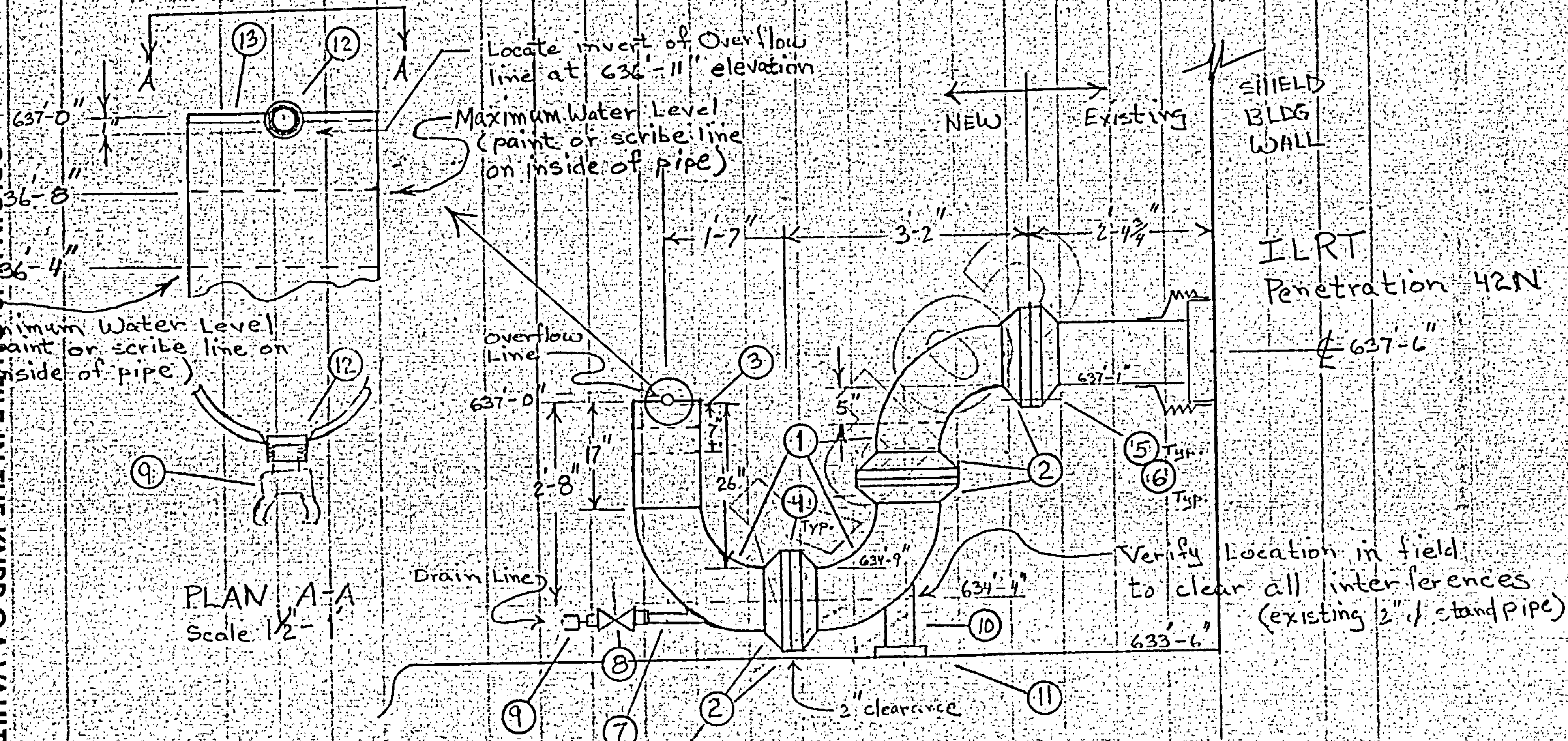
The design basis accident which bounds the loop seal modification is the fuel handling accident, of which no unreviewed safety question exists. The gaseous activity associated with the fuel handling accident and resulting off-site exposure are well below the 10 CFR 100 guidelines. The accident is assumed to occur in the spent fuel pool and any accident occuring in containment would have substantially lower off-site radiation levels.

MJOrtmayer

1-7-87

REFUELING CNTMT LX SEAL

DCR-1811



ELEVATION VIEW
Aux. Bldg. Operating Floor 633'-6" EL.

Scale ½" = 1'

Reference DWGs.
M.350

Sketch 1811-1

2MJO2.1

Design Basis

(T/S 3.8.a.1)

September 22, 1986

Green Bay

DCR - 1811, Containment Cable/Hose Pass Through Utilizing the ILRT Penetration 42N During Refueling

D J Ropson

cc - K H Weinhauer w/o attachments
D J Will

Reference: Letter to C. A. Schrock from R. E. Draheim dated 01-11-85
Letter to R. E. Draheim from K. L. Hull dated 02-05-85

As assigned Responsible Engineer for DCR 1811, I have been reviewing engineering work previously performed under DCR 1604 which this project originated as. At that time, two items were identified which needed resolution before design engineering work could proceed:

1) The design basis requirements for the penetration seal in light of Technical Specification Paragraph 3.8.a.1.

2) If a water loop seal design is selected, the possible detrimental effects submergence would have on cables.

I have enclosed some previous correspondence on these matters for your review, please refer to them. Also enclosed as an attachment to this letter, is my analysis of Item One.

Would you please review with the appropriate members of your staff, Item One, and provide me a letter of clarification on this Technical Specification paragraph. A design basis must be established which will enable me a foundation to base my design on.

Mark Ortmayer
MJOrtmayer:jal

Attachment

Jan 11, 1995

C.A. Schrock

Per your generous offer, I would like to request some assistance on the ILRT containment penetration modification. Specifically I would like to know:

1. What DP does this refueling integrity penetration seal have to be designed for?
2. Do we need a double seal or will a single boundary suffice?
3. Do we have to test the seal? How often? What DP? What leakage acceptance criteria can we use?
4. Is it acceptable to use QA-3 piping as part of this containment boundary?

Per C. Tomes Westinghouse cable does not have a jacket qualified for submergence so I believe we will go with some kind of mechanical (RTV, flamastic, etc.) system. The answers to the above questions will be the basis for the safety evaluation. The physical modification will not be too difficult.

February 5, 1985

Green Bay

DCR 1604 Refueling Cable/Hose Vessel
Penetration on ILRT Penetration 42N

R E Draheim

cc - C A Schrock
K H Weinhauer
D J Ropson
R P Repshas

As you are aware, this project is modified to address two separate individual concerns; 1) a potential USQ with the Containment Pressurization Piping forces on the Containment Vessel penetration and 2) a refueling cable/hose penetration that will provide the required degree of Containment Integrity during the fuel shuffle.

Having spent as much time and effort as I can to research the requirements for this project and to investigate the design alternatives I recommend the following schedule due to problems with design criteria.

1) We proceed to modify the Aux Bldg. piping to satisfy the vessel design criteria for piping penetrations. This will consist of a removable spool of pipe with suitable flanges and gaskets. This will permit the uncoupling and a blind flange seal on both the vessel piping penetrations and downstream piping. The pressurization piping will be connected only during the ILRT, when we have reduced design criteria, since we are in cold or refueling shutdown. FEI will perform the analysis and we will complete this modification as soon as practical.

The refueling cable/hose penetration portion should wait until we can establish satisfactory design criteria. To date these are the problems.

1) The design pressure for the penetration has been verbally communicated to be 5 psi. This requirement, though extremely conservative, is objectionable to maintenance since it eliminates the loop seal similar to PI. I cannot at this time recommend we proceed based on the arbitrary 5 psi criteria when PI has a 2 foot deep water seal.

2) In contacting the vendors for the Eddy Current and ISI, either personally or through the WPSC responsible individual, they have objections to submerging their cables in a loop seal. Their reasons are: a) these cables are not qualified for submergence; b) they object to pulling their cable through the water and c) they've indicated water may change the signal quality of the communication cables between the sensors and the data acquisition/processing equipment.

Maintenance has indicated all cables can be wrapped with waterproof protection. However, this still does not address the signal quality effects. I have discussed this with Dan Cole and he's indicated there should be no effect from the loop seal water on signal quality. However, without proof or further investigation, we risk invalidating any inspection work which used a loop seal penetration for data transmittal.

Finally, indiscussions with Dan Cole, FEI is willing to wait on the design for the refueling penetration at no extra cost.

On this basis, I recommend we proceed as follows:

1) Perform a detailed study to establish and document the design pressure requirement.

2) In the future, require all cables, regardless of vendor, to be submersible or specifically address the vendor concerns on a case by case basis.

3) Proceed with the design when agreement can be reached between the responsible parties.

If you have any questions, please contact me.

Kurt L Hull RE

KLHull:las

Attachment I

Letter to Dan Ropson

from

Mark Ortmayer

Dated September 22, 1986

Basis and Interpretation of

Technical Specifications Paragraph 3.8.a.1

Basis and Interpretation of Technical Specification Paragraph 3.8.a.1

Paragraph 3.8.a.1 requires that during refueling operations, the equipment hatch and at least one door in each air lock be closed. In addition, each penetration of containment shall have a closed isolation valve or an operable automatic isolation valve.

The basis for these requirements is the fuel handling accident as discussed in the FSAR Section 14.2.1 Activity release characteristics are such that a sudden release of the gaseous fission products held in the voids between the pellets and cladding would occur. It is assumed the accident could occur either inside the containment building or in the auxiliary building. Both of these areas have there separate ventilation systems. In calculating off-site exposure, it was assumed the accident occurs in the spent fuel pool with the total release discharging to the atmosphere. No credit was taken for the SFP Sweep System filtration.

If a fuel handling accident occurred in containment, you would want to bottle up the containment. The containment building vent stack radiation monitor on a high-level alarm automatically closes the purge supply and exhaust ducts. What must be avoided is a pressure differential between containment and the outside which would permit a direct release path. Possible sources of pressurization are the containment vent supply fan (which was the basis for the Prairie Island loop seal design) or the POST LOCA air pressurization line. Long term sources (on the order of days, which are not relevant to this case) would be instrument air leakage or air temperature changes.

If the basis used in determining Tech. Spec. 3.8.a.1 is the containment vent supply fan, such a fan would typically have a total static pressure of approximately 8 inches water column. A loop seal of 28 inches (or 1 psi) would have a considerable safety factor. Based on this premise, one could say a mechanical seal device that seals to 28 inches water column with some small finite leak rate would also be acceptable.

I am then of the opinion that you would meet the intent of Tech. Spec. 3.8.a.1 if a device would seal to a water column of 28 inches with no or minimal leakage. The fuel handling accident would be bounded by this basis.

November 5, 1986

Green Bay

DGR-1811, Containment Cable/Hose Pass Through Utilizing the ILRT Penetration 42N During Refueling

Reference 1) Letter from M J Ortmayer to D J Ropson dated September 22, 1986

M J Ortmayer

cc - D J Ropson

Reference 1 requested a letter of clarification concerning the interpretation of Technical Specification 3.8.a.1.

T.S. 3.8.a.1 states:

> "The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve."

The specification requires that only a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.

The water loop seal design used by Prairie Island (PI) does not have isolation valves; however, it is acceptable since the water acts as a barrier preventing a direct air path from the containment atmosphere to the outside atmosphere.

DJMolzahn:kvh

N-RBV-18B

2.2 Discharge when
current press. [illegible]
1.5 psi.

TS 3.6.C
If press > 2 PSI correct
within 8 hrs or shutdown

DCR 2167 - Design Description

This DCR will provide a new refueling containment cableway.

Penetration 43N, an 18 inch diameter penetration, will be modified for use as a cable pass through. A 150 lb. flange will be welded to the inner nozzle. A blind flange with the testable double gasket seal will be installed and tested to satisfy containment integrity.

A 18 inch diameter pipe will be welded to the outer nozzle, extended through the 24 inch diameter Shield Building sleeve, into the 633' 6" level of the Aux. Building. A blind flange will be installed during normal operation. A hypalon boot Shield Building penetration seal will be used to minimize Shield Building leakage. A 1" diameter vent valve will be installed on the 18 inch diameter pipe to vent into the Shield Building annulus.

A 18" diameter loop seal will be constructed of schedule 10s stainless steel pipe and fittings. The loop seal will block a direct air path from containment to the atmosphere, to provide for refueling containment integrity during fuel movement. It will seal containment for a pressure range of ± 20 inches water gage pressure.

During normal operations, the new blind flange is installed and the double gasket arrangement leak tested. The outer blind flange is installed and the annulus vent valve is open to vent any leakage into the annulus. This is exactly similar to penetration 42N as it is installed and described in the FSAR.

During refueling, the blind flanges are taken off, the loop seal assembly is attached to the Auxiliary Building flange connection, and the 1 inch Shield Building annulus vent valve is closed. Cables are pulled through the loop seal into containment. When integrity is required, the loop seal is filled with water to the indicated level.

The status of the loop seal can be monitored by comparing the water level in the leg open to the Auxiliary Building with the water level in the loop seal leg connected to containment. A 3/4" liquid level gage is installed to enable the Aux Operator to check the level in the loop seal leg connected to containment.

A drain connection with shutoff valve and a overflow connection will be provided on the loop seal assembly.

A spare containment penetration (Pen. #43N) will be opened and constructed to be of identical design to the ILRT pressurization penetration #42N as shown in USAR Fig. 11.1-4 (Dwg. M350, Flow Diagram-Reactor Plant Miscellaneous Vents, Drains And Sump Pump Piping), except that the penetration piping will be the same size (18 inch) as the penetration #43N. The appropriate sections of the USAR, and in particular the penetration Table 5.2-2, will be updated to add this new active containment penetration.

A loop seal, for cable pass through, will be constructed to meet the requirements of Technical Specification 3.8.a.1, which states "During refueling operations: The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve." The loop seal will satisfy the requirement for a closed isolation valve.

Attachment 1 to this SER illustrates the loop seal arrangement for penetration #43N, which will provide a maximum seal of 20 inches, either positive or negative pressure in containment relative to the fuel handling area, which will be essentially at atmospheric pressure. The water level in the open end of the loop seal will provide capability for periodic monitoring of the differential pressure between the containment and the fuel handling area. 1

During normal refueling operation, with the containment vent supply and vent exhaust systems in proper balance, the pressure in containment should be essentially atmospheric. However, in the event of component/system malfunction or inadvertent actuation, the pressure in containment could decrease or increase. The safety evaluation for the minimum seal height required for potential decrease or increase in the containment pressure is as follows:

1. Decrease in Containment Pressure

   A decrease in containment pressure could be as a result of inadvertent actuation of all post accident containment pressure suppression systems, i.e., both trains of the containment spray system and all four containment fan coils, with the containment vent isolation in progress. USAR Section 5.4.3 discusses this analysis as a basis for the containment vacuum relief system sizing in order to prevent the containment vessel from exceeding the maximum permissible external/internal pressure differential of 0.8 psi. This analysis is also based on no heat energy being added to the containment, i.e., the reactor at cold/refueling shutdown. With only one train of the vacuum relief system operating, this event will produce a maximum pressure differential between the shield building and the containment of 0.556 psi (USAR Figure 5.4-4). Since during refueling operation the shield building is open to the fuel handling area, the same differential pressure is applicable to the loop seal. This differential of 0.556 psi translates to 15.4" WC (0.556 psi x 2.31 ft/psi x 12 in./ft = 15.4 inches). Since this analysis is based on the containment temperature being initially at 120°F, during refueling operation with the initial temperature of approximately 70°F, this transient will be less severe.

A decrease in containment pressure could also be experienced in the event of inadvertent isolation of the containment vent supply penetration when the containment vent exhaust fan could draw the containment pressure down to the maximum capability of the fan. The containment vent exhaust fan curve, Attachment 2 to this SER, shows that the maximum pressure capability of the fan is 24" WC. The vent exhaust fan is interlocked to trip if either of the two vent supply penetration isolation valves close. However, should the interlock fail, one train of the vacuum relief system will prevent the fan from drawing more than negative 18" WC for the fan blade setting of 5. From Attachment 2, at fan pressure of 18" WC, the fan flow will be less than 12,500 scfm; from the vacuum relief system sizing analysis dated February 1970, one train of the system will pass 14,600 scfm at a differential pressure of 18" WC. Even at a fan blade setting of 4, the differential for the loop seal will be less than 18" WC.

2. Increase in Containment Pressure

Inadvertent isolation of containment vent exhaust penetration will result in containment pressure increase up to the maximum capability of the vent supply fan. Since the supply fan is a low pressure unit with a design static pressure of 3.5" WC, the maximum capability will probably be less than 10" WC.

The most likely event to occur is ventilation isolation upon high radiation as detected by R-11 and/or R-12. This could result in containment pressure increase due to rise in temperature of the containment environment or due to instrument/station air leak in the containment. Either of these, however, is a slow process which will allow time for operator action to reduce temperature or isolate the air leak. A calculation was performed for a pressure increase of 16" WC, which shows the following:

(1) With no air leak and initial containment temperature of 70°F, a 21°F temperature rise will produce a pressure increase of 16" WC.

(2) With constant temperature in containment, addition of 51,800 cubic feet of air at standard conditions will produce a pressure increase of 16" WC. A complete severance of the largest size air line, which is the 2" station air line, will discharge 3,600 scfm, which will increase the containment pressure to 16" WC in 15 minutes. A 1" diameter break will discharge 1,800 scfm, with an increase in containment pressure to 16" WC in 30 minutes. These flow rates are based on conservative calculations which neglect pressure drop in the piping system and which assumes sufficient compressor capacity to sustain these flow rates. (These flow rates are much higher than the combined capacity of all five air compressors).

Based on the above, a 18" minimum seal height should be sufficient to provide containment integrity for various abnormal occurrences. The normal water level (with the cables in place), as shown in Attachment 1, will provide for a maximum differential of 12" without loss of water from the loop seal. For higher differentials, some of the water will be lost either via the overflow connection in the fuel handling area or via the penetration into the containment. 1

Based on the Safety Evaluation for DCR-1811, the loop seal is classified as a QA Type 3 component (see Attachment 3). The calculations for DCR 1811 show that for a fuel handling accident inside containment and a simultaneous failure of the loop seal, exposure at the site boundary will be less than the criteria for 1

Wisconsin Public Service Corporation DCR No. 2167
Kewaunee Nuclear Power Plant Sheet No. 5 of 5
Design Change Record Sheet Subject SER

QA Type 1, less than 0.25 rem whole body or 3.0 rem thyroid dose over a two hour exposure. A review of the calculations show that the site boundary dose is <u>not dependent</u> on the size of the vent opening in the containment boundary; and therefore, the calculated dose for the 10" DCR-1811 Loop Seal is also applicable to the 18" Loop Seal.

The loop seal, however, was seismically analyzed and supported so as to limit the containment penetration stress levels from exceeding the code allowable stresses. The stress analysis, based on a cable weight of 60 pounds per linear foot, show the stress levels to be well within the code allowable stresses, with the highest stress in the 18" schedule 10S loop seal, which is only about 25% of the code allowable stress. 1

Rev. 0- RP Bergus 2/4/88
Rev. 1. RP Bergus 2/19/88

REV. 1. 2-18-88

FLUOR DANIEL

CALCULATIONS and SKETCHES

DATE	2-04-88
CONT. NO.	834856
BY	S.K.F.
CHK'D	
SHEET NO.	1 OF 1

WISCONSIN PUBLIC SERV. CORP

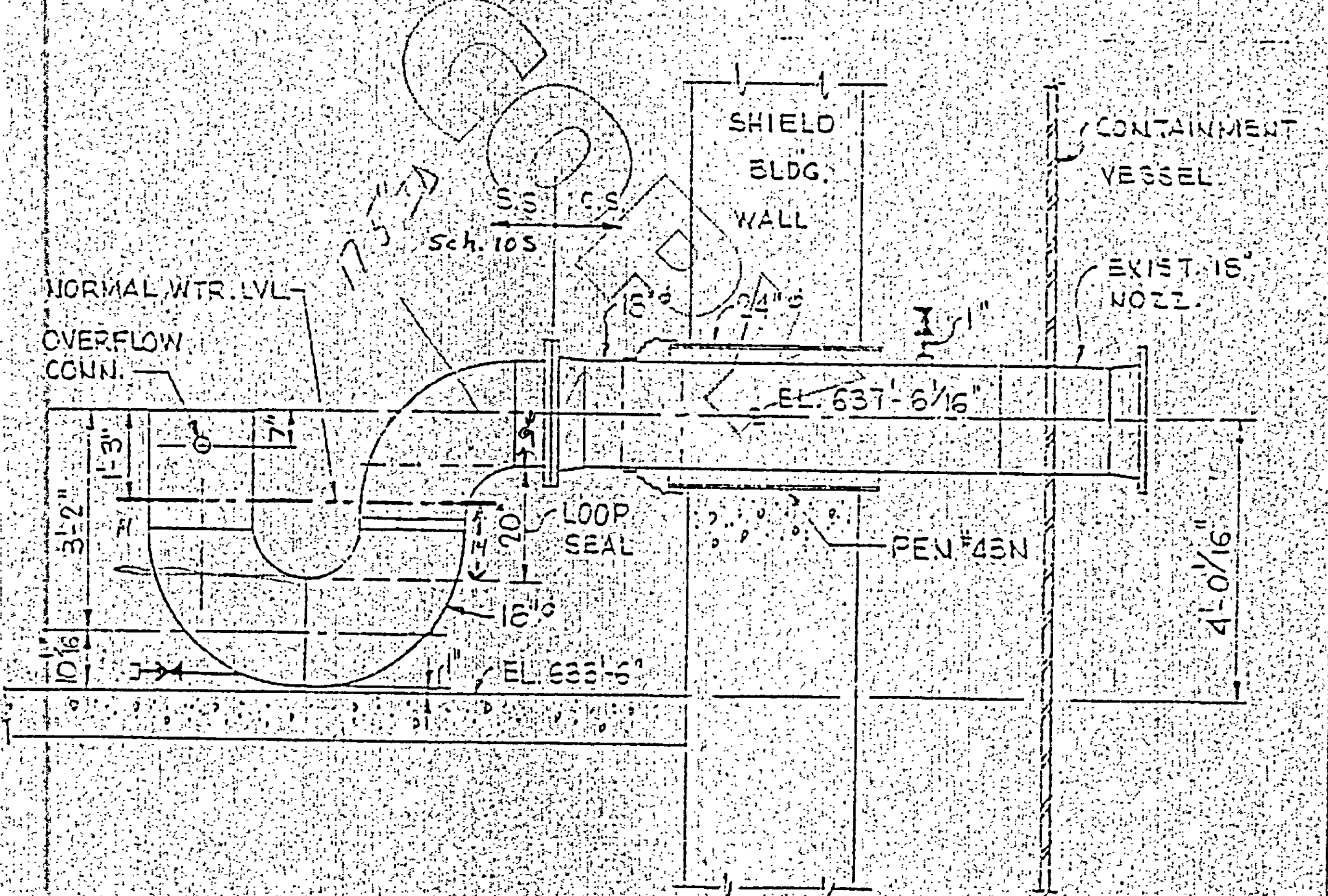
KEWAUNEE NUCLEAR POWER PLANT

REFUELING CONTAINMENT LOOP SEAL PEN. #43N

DCR #2167

DCR #2167 SER

ATTACHMENT 1



ELEVATION VIEW

SCALE: 3/8" = 1'-0"

FORM E-050 CPD 12/86
PRINTED IN U.S.A.

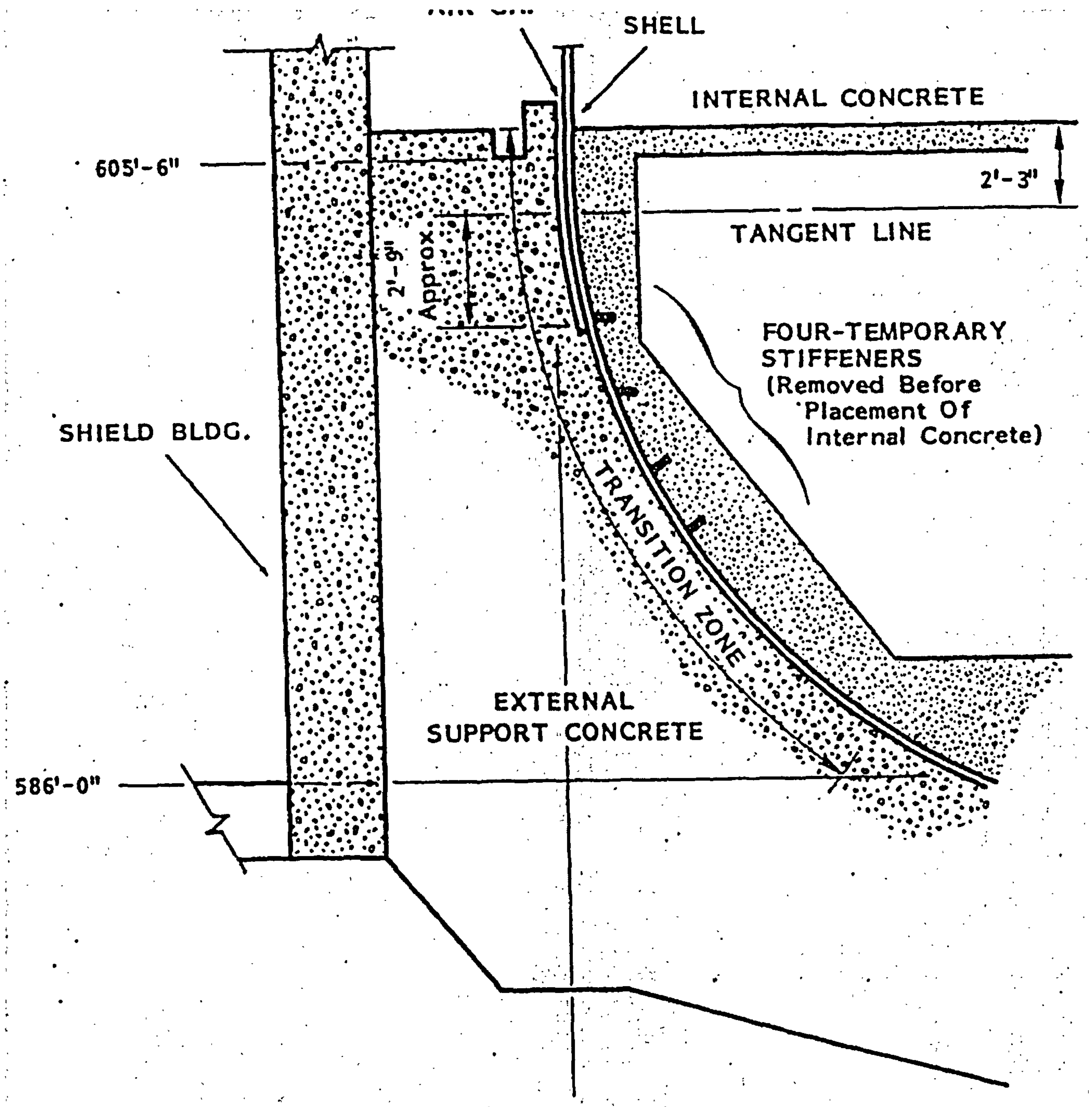


FIGURE 5.2-1

SHELL TRANSITION ZONE AND INTERNAL STIFFENERS

GE-84	ANNUNCIATOR NUMBER:	H2 PANEL 3-1

SER PT:
None

COMPUTER PT:
None

INSTRUMENT:
PS-16968

HYDROGEN
PRESSURE
HIGH OR LOW

SETPOINT:
High > 70 psig
Low < 50 psig

RECOMMENDED ACTION:

1. VERIFY proper operation of SW-2602/CV-31068, Gen H_2 Temp CV.

2. GO TO A-EG-43A.

COMMENTS:

1. SW-2602 fails open on loss of air.

2. Changes in hydrogen pressure will cause changes in stator temperatures; as pressure decreases temperature will increase.

REFERENCES:
FLOW: XK-101-235, M-216 OTHER: E-449
LOGIC: E-3639

NUCLEAR SAFETY RELATED	YES ☐ NO ☒	PORC REVIEW REQUIRED	YES ☐ NO ☒	SRO APPROVAL OF TEMPORARY CHANGES REQUIRED	YES ☒ NO ☐

REVIEWED BY [signature] DATE APR 05 2001

APPROVED BY [signature] DATE APR 05 2001

REV DATE
4/5/01
REV D

NUREG-0737
Supplement No. 1

Clarification of TMI Action Plan Requirements

Requirements for Emergency Response Capability

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

EMERGENCY RESPONSE CAPABILITY

INTRODUCTION

This supplement was prepared as a result of a review by the Committee to Review Generic Requirements (CRGR). The supplement represents the staff's attempt to distill the fundamental requirements for nuclear plant Emergency Response Capability from the wide range of guidance documents that the NRC has issued. It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be implemented as written; rather, they should be regarded as useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document. It is also not intended that either the guidance documents or the fundamental requirements are to be considered binding legal requirements at this time. As indicated below, however, the fundamental requirements will be translated into binding legal requirements in the manner specified.

These requirements are a further delineation of the general guidance issued previously by the Commission in its regulations, orders and policy statements on emergency planning and TMI issues. It is intended that these requirements would be applicable to licensees of operating nuclear power plants. For applicants for a construction permit (CP) or manufacturing license (ML), the requirements described in this document must be supplemented with the specific provisions in the rule specifying licensing requirements for pending CP and ML applications. Thus, compliance with requirements in this document may not be sufficient to meet the related requirements in 10 CFR 50.34(f) and Appendix E. In this regard, it is expected that the staff would review CP and ML applications against the guidance in the current Standard Review Plan (which includes the provisions of NUREG-0718) and this might lead to more detailed requirements than prescribed in this document in order to satisfy the requirements of 50.34(f) and Appendix E.

Based on discussions with licensees, the staff has learned that many of the Commission approved schedules for emergency response facilities probably will not be met. In recognition of this fact and the difficulty of implementing generic deadlines, plant-specific schedules will be established which take into account the unique status of each plant. The following sequence for developing implementation schedules will be used.

The requirements for emergency response capabilities and facilities are being transmitted to licensees by this supplement and are being promulgated to NRC staff. The letter which forwards this supplement requests that licensees submit a proposed schedule for completing actions to comply with the requirements.

Each licensee's proposed schedule will then be reviewed by the assigned NRC Project Manager, who will discuss the subject with the licensee and mutually agree on schedules and completion dates. The implementation dates will then be formalized into an enforceable document.

The requirements in this document do not alter previously issued guidance, which remains in effect. This document does attempt to place that guidance in perspective by identifying the elements that the NRC staff believes to be essential to upgrade emergency response capabilities. The proposal to formalize implementation dates in an enforceable document reflects the level of importance which the NRC staff attributes to these requirements. The Commission does not believe that existing guidance should be imposed in this manner, but rather that it be used as guidance to be considered in upgrading emergency response capabilities. This indicates the distinction which the staff believes should be made between the requirements and guidance.

The following sections describe the requirements, their interrelationhips, and NRC actions to improve management of emergency response regulations. Reference documents are cited with a description of content as it relates to specific initiatives.

The requirements set forth in this document have been reviewed by the Commission and, at a meeting held July 16, 1982, were approved by the Commission as appropriately clarifying and providing greater detail with respect to related TMI Action Plan requirements contained in NUREG-0737 for all operating license applicants. These requirements are, therefore, to be accorded the status of approved NUREG-0737 items as set forth in the Commission's "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses" (45 FR 85236), December 24, 1980). In this connection, the provisions for scheduling set forth herein supersede any schedules with respect to such items contained in NUREG-0737. Accordingly, the requirements should be used by the staff and by adjudicatory boards as appropriate clarifications and interpretation of the related NUREG-0737 items.

The requirements set forth in this document are believed to be consistent with the requirements regarding related items for construction permits and manufacturing licenses contained in 10 CFR 50.34(f) and 10 CFR Part 50, Appendix E. Accordingly, no changes to these regulations are required.

2. USE OF EXISTING DOCUMENTATION

The following NUREG documents are intended to be used as sources of guidance and information, and the Regulatory Guides are to be considered as guidance or as an acceptable approach to meeting formal requirements. The items by virtue of their inclusion in these documents shall not be misconstrued as requirements to be levied on licensees or as inflexible criteria to be used by NRC staff reviewers.

NUREG Report		Titles
0696	-	Functional Criteria for Emergency Response Facilities
0700	-	Guidelines for Control Room Design Reviews
0799	-	Draft Criteria for Preparation of Emergency Operating Procedures (to be superseded by NUREG-0899)
0801	-	Evaluation Criteria for Detailed Control Room Design Reviews
0814	-	Methodology for Evaluation of Emergency Response Facilities
0818	-	Emergency Action Levels for Light Water Reactors
0835	-	Human Factors Acceptance Criteria for SPDS
0899	-	Guidelines for the Preparation of Emergency Operating Procedures: Resolution of Comments on NUREG-0799

Regulatory Guides		Titles
1.23 (Rev. 1)	-	Meteorological Measurement Program for Nuclear Power Plants
1.97 (Rev. 2)	-	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident
1.101 (Rev. 2)	-	Emergency Planning for Nuclear Power Plants
1.47	-	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

3. COORDINATION AND INTEGRATION OF INITIATIVES

3.1 The design of the Safety Parameter Display System (SPDS), design of instrument displays based on Regulatory Guide 1.97 guidance, control room design review, development of function oriented emergency operating procedures, and operating staff training should be integrated with respect to the overall enhancement of operator ability to comprehend plant conditions and cope with emergencies. Assessment of information needs and display formats and locations should be performed by individual licensees. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS may obviate the need for large-scale control room modifications. Installation of the SPDS should not be delayed by slower progress on other initiatives, and should not be contingent on completion of the control room design review. Nor should other initiatives, such as upgraded emergency operating procedures, be impacted by delays in SPDS procurement. While the NRC does not plan to impose additional requirements on licensees regarding SPDS, the NRC will work with the industry to assure the development of appropriate industry standards for SPDS systems.

3.2 Implementation of part or all of Regulatory Guide 1.97 (Rev. 2) represents a control room improvement. The implementation of control room improvements is not contingent on implementing Technical Support Center (TSC) and Emergency Operations Facility (EOF) requirements.

3.3 The Technical Support Center (TSC) and Emergency Operations Facility (EOF) are dependent on control room improvements in terms of communication and instrumentation needs among the TSC, EOF, and control room. TSC and EOF facilities are not necessarily dependent on each other. The Operational Support Center (OSC) is independent of TSC and EOF.

3.4 The three groups of initiatives--SPDS, control room improvements, and emergency response facilities (TSC, EOF, OSC)-- have the following interrelationships:

a. The SPDS is an improvement because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS could obviate the need for extensive modifications to control rooms.

b. New instrumentation that may be added to the control room should be considered a requirement for inclusion in the design of the TSC and EOF only to the extent that such instrumentation is essential to the performance of TSC and EOF functions.

c. The SPDS and control room improvements are essential elements in operator training programs and the upgraded plant-specific emergency operating procedures.

d. Acquisition, processing, and management of data for SPDS, control room improvements, and emergency response facilities should be coordinated.

3.5 Specific implementation plans and reasonable, achievable schedules for improvements that will satisfy the requirements will be established by agreement between the NRC Project Manager and each individual licensee. The NRC office responsible for implementing each requirement will develop procedures identifying the following.

a. The respective roles of NRR, IE, and Regional Offices in managing implementation, checking licensee rate of progress, and verifying compliance, including the extent to which NRC review and inspection is necessary during implementation.

b. Procedural methods and enforcement measures that could be used to ensure NRC staff and licensee attention to meeting mutually agreed upon schedules without significant delays and extensions.

3.6 The NRC Project Manager for each nuclear power plant is assigned program management responsibility for NRC staff actions associated with implementing emergency response initiatives. The NRC Project Manager is the principal contact for the licensee regarding these initiatives.

3.7 The NRC will make allowances for work already done by licensees in a good-faith effort to meet requirements as they understand them. For each case in which a licensee would have to remove or rip out emergency response facilities or equipment that was installed in good faith to meet previous guidance in order to meet the basic requirements described in this document, the Director of the Office of Nuclear Reactor Regulation or Inspection and Enforcement will review the circumstances and determine whether removal is necessary or existing facilities or equipment represent an acceptable alternative. Any regulatory position that would require the removal or major modification of existing emergency response facilities or equipment requires the specific approval of the responsible Office Director.

3.8 The NRC recognizes that acceptable alternative methods of phasing and integrating emergency response activities may be developed. Each licensee needs flexibility in integrating these activities, taking into account the varying degree to which the licensee has implemented past requirements and guidance. An example of a way in which these activities could be integrated is discussed below. Other methods of integration proposed by licensees would be reviewed considering licensees' progress on each initiative.

a. SPDS

(1) Review the functions of the nuclear power plant operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk, (b) could cause operators to make cognitive errors in diagnosing them, and (c) are not included in routine operator training programs.

(2) Combine the results of this review with accepted human factors principles to select parameters, data display, and functions to be incorporated in the SPDS.

(3) Design, build, and install the SPDS in the control room and train its users.

b. To be done in parallel without delaying SPDS, complete emergency operating procedure technical guidelines that will be used to develop plant-specific emergency operating procedures.

c. Using these EOP technical guidelines, the SPDS design, and accepted human factors principles, conduct a review of the control room design. Apply the results of this review to:

(1) Verify SPDS parameter selection, data display, and functions.

(2) Develop plant-specific EOPs.

(3) Design control room modifications that correct conditions adverse to safety (reduce significant contributions to risk), and add additional instrumentation that may be necessary to implement Regulatory Guide 1.97.

(4) Train and qualify plant operating staff regarding upgraded EOPs and modifications.

d. Verify, prior to finalization of designs for modifications and of procedures and training, that the functions of control room operators in emergencies can be accomplished (i.e., that the individual initiatives have been integrated sufficiently to meet the needs of control room operators and provide adequate emergency response capabilities).

e. Implement EOPs and install control room modifications coincident with scheduled outages as necessary, and train operators in advance of these changes as they are phased into operation.

. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

4.1 Requirements

a. The SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

b. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

c. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) provides the operators with the information necessary for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, 4 in Appendix A; 10 CFR Part 100; single-failure requirements). The SPDS need not meet requirements of the single-failure criteria and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. Procedures which describe the timely and correct safety status assessment when the SPDS is and is not available, will be developed by the licensee in parallel with the SPDS. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.

d. There is a wide range of useful information that can be provided by various systems. This information is reflected in such staff documents as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97

Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information that should be provided for a particular plant shall be based on engineering judgement of individual plant licensees, taking into account the importance of prompt implementation.

e. The SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily preceived and comprehended by SPDS users.

f. The minimum information to be provided shall be sufficient to provide information to plant operators about:

(i) Reactivity control

(ii) Reactor core cooling and heat removal from the primary system

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions

The specific paramenters to be displayed shall be determined by the licensee.

4.2 Documentation and NRC Review

a. The licensee shall prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Such analysis, along with the specific implementation plan for SPDS shall be reviewed as described below.

b. The licensee's proposed implementation of an SPDS system shall be reviewed in accordance with the licensee's technical specifications to determined whether the changes involve an unreviewed safety question or change of technical specifications. If they do, the shall be processed in the normal fashion with prior NRC review. If the changes do not involve an unreviewed safety question or a change in the technical specifications, the licensee may implement such changes without prior approval by NRC or may request a pre-implementation review and approval. If the changes are to be implemented without prior NRC approval, the licensee's analysis shall be submitted to NRC promptly on completion of review by the licensee's offsite safety review committee. Based on the results of NRC review, the Director of IE or the Director of NRR may request or direct the licensee to cease implementation if a serious safety question is posed by the licensee's proposed system, or if the licensee's analysis is seriously inadequate.

4.3 Integration

Prompt implementation of an SPDS is a design goal and of primary importance. The schedule for implementing SPDS should not be impacted by schedules for the control room design review and development of function-oriented emergency operating procedures. For this reason, licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to the other initiatives. If reasonable, this schedule will be accepted by NRC.

4.4 Reference Documents

NUREG-0660	--	Need for SPDS identified
NUREG-0737	--	Specified SPDS
NUREG-0696	--	Functional Criteria for SPDS
NUREG-0835	--	Specific acceptance criteria keyed to NUREG-0696
Reg. Guide 1.97 (Rev. 2)	--	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

5. DETAILED CONTROL ROOM DESIGN REVIEW

5.1 Requirements

a. The objective of the control room design review is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (from NUREG-0660, Item I.D.1). As a complement to improvements of plant operating staff capabilities in response to transients and other abnormal conditions that will result from implementation of the SPDS and from upgraded emergency operating procedures, this design review will identify any modifications of control room configurations that would contribute to a significant reduction of risk and enhancement in the safety of operation. Decisions to modify the control room would include consideration of long-term risk reduction and any potential temporary decline in safety after modifications resulting from the need to relearn maintenance and operating procedures. This should be carefully reviewed by persons competent in human factors engineering and risk analysis.

b. Conduct a control room design review to identify human engineering discrepancies. The review shall consist of:

(i) The establishment of a qualified multidisciplinary review team and a review program incorporating accepted human engineering principles.

(ii) The use of function and task analysis (that had been used as the basis for developing emergency operating procedures Technical Guidelines and plant specific emergency operating procedures) to identify control room operator tasks and information and control requirements during emergency operations. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters.

(iii) A comparison of the display and control requirements with a control room inventory to identify missing displays and controls.

(iv) A control room survey to identify deviations from accepted human factors principles. This survey will include, among other things, an assessment of the control room layout, the usefulness of audible and visual alarm systmes, the information recording and recall capability, and the control room environment.

(ii) An audit of the justification for those human engineering discrepancies of safety significance to be left uncorrected or only partially corrected.

The audit will consist of a review of the licensee's record of the control room reviews, discussions with the licensee review team, and usually a control room visit. Within a month after this onsite audit, NRC will issue its safety evaluation report (SER).

e. For control rooms for which NRC does not perform a pre-implementation onsite audit, NRC will conduct a review and issue its SER within ~~two~~ months after receipt of the licensee's summary report. The review shall be similar to that conducted for pre-implementation plants under paragraph 4 above, except that it does not include a specific audit. The SER shall indicate whether, based on the review carried out, changes in the licensee's modification plan are needed to assure operational safety. Flexibility is considered in the control room review, because certain control board discrepancies can be overcome by techniques not involving control board changes. These techniques could include improved procedures, improved training, or the SPDS.

f. The following approach will be used for OL review. For OL applications with SSER dates prior to June 1983, licensing may be based on either a Preliminary Design Assessment or a Control Room Design Review (CRDR) at the applicant's option. However, applicants who choose the Preliminary Design Assessment option are required to perform a CRDR after licensing. For applications with SSER dated after June 1983, Control Room Design Review will be required prior to licensing.

g. After the staff has issued an SER and licensees have addressed any open issues, they may begin their upgrade according to an approved schedule that has been negotiated with the staff.

5.3 Reference Documents

NUREG-0585	--	States that licensees should conduct review.
NUREG-0660 (Rev. 1)	--	States that NRR will require reviews for operating reactors and operating licensee applicants.
NUREG-0700	--	Final guidelines for CRDR.
NUREG-0737	--	States that requirement was issued June, 1980, final guidance not yet issued.
NUREG-0801	--	Staff evaluation criteria.

6. REGULATORY GUIDE 1.97 - APPLICATION TO EMERGENCY RESPONSE FACILITIES

6.1 Requirements

a. Functional Statement

Regulatory Guide 1.97 provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

b. Control Room

Provide measurements and indication of Type A, B, C, D, E variables listed in Regulatory Guide 1.97 (Rev. 2). Individual licensees may take exceptions based on plant-specific design features. BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements. It is acceptable to rely on currently installed equipment if it will measure over the range indicated in Regulatory Guide 1.97 (Rev. 2), even if the equipment is presently not environmentally qualified. Eventually, all the equipment required to monitor the course of an accident would be environmentally qualified in accordance with the pending Commission rule on environmental qualification.

Provide reliable indication of the meteorological variables (wind direction, wind speed, and atmospheric stability) specified in Regulatory Guide 1.97 (Rev. 2) for site meteorology. No changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity (up to about 10 miles) of the plant site. Information on meteorological conditions for the region in which the site is located shall be available via communication with the National Weather Service. These requirements supersede the clarification of NUREG-0737, Item III.A.2.2.

c. Technical Support Center (TSC)

The Type A, B, C, D and E variables that are essential for performance of TSC functions shall be available in the TSC.

(i) BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements.

(ii) The indicators and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

d. Emergency Operations Facility (EOF)

(i) Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be available in the EOF.

(ii) The EOF data indications and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

6.2 Documentation and NRC Review

NRC review is not a prerequisite for implementation. Staff review will be in the form of an audit that will include a review of the licensee's method of implementing Regulatory Guide 1.97 (Rev. 2) guidance and the licensee's supporting technical justification of any proposed alternatives.

The licensee shall submit a report describing how it meets these requirements. The submittal should include documentation which may be in the form of a table that includes the following information for each Type A, B, C, D, E variable shown in Regulatory Guide 1.97 (Rev. 2).

(a) instrument range

(b) environmental qualification (as stipulated in guide or state criteria)

(c) seismic qualification (as stipulated in guide or state criteria)

(d) quality assurance (as stipulated in guide or state criteria)

(e) redundance and sensor(s) location(s)

(f) power supply (e.g., Class 1E, non-Class 1E, battery backed)

(g) location of display (e.g., control room board, SPDS, chemical laboratory)

(h) schedule (for installation or upgrade)

Deviations from the guidance in Regulatory Guide 1.97 (Rev. 2) should be explicitly shown, and supporting justification or alternatives should be presented.

UPGRADE EMERGENCY OPERATING PROCEDURES (EOFs)

7.1 Requirements

a. The use of human factored, function oriented, emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors, without the need to diagnose specific events.

b. In accordance with NUREG-0737, Item I.C.1, reanalyze transients and accidents and prepare Technical Guidelines. These analyses will identify operator tasks, and information and control needs. The analyses also serve as the basis for integrating upgraded emergency operating procedures and the control room design review and verifying the SPDS design.

c. Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide.

d. Provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs.

e. Implement upgraded EOPs.

7.2 Documentation and NRC Review

a. Submit Technical Guidelines to NRC for review. NRC will perform a pre-implementation review of the Technical Guidelines. Within two months of receipt of the Technical Guidelines, NRC will advise the licensees of their acceptability.

b. Each licensee shall submit to NRC a procedures generation package at least three months prior to the date it plans to begin formal operator training on the upgraded procedures. NRC approval of the submittal is not necessary prior to upgrading and implementing the EOPs. The procedures generation package shall include:

(i) Plant-Specific Technical Guidelines — plant-specific guidelines for plants not using generic technical guidelines. For plants using generic technical guidelines, a description of the planned method for developing plant specific EOPs from the generic guidelines, including plant specific information.

(ii) A Writer's Guide that details the specific methods to be used by the licensee in preparing EOPs based on the Technical Guidelines.

(iii) A description of the program for validation of EOPs.

(iv) A brief description of the taining program for the upgraded EOPs.

c. All procedures generation packages will be reviewed by the staff. On an audit basis for selected facilities, upgraded EOPs will be reviewed. The details and extent of this review will be based on the quality of the procedures generation packages submitted to NRC. A sampling of upgraded EOPs will be reviewed for technical adequacy in conjunction with the NRC Reactor Inspection Program.

7.3 Reference Documents

NUREG-0600,
Item I.C.1, I.C.8, I.C.9

NUREG-0799 -- (Superseded by NUREG-0899)

EMERGENCY RESPONSE FACILITIES

8.1 Regulations

10 CFR 50.47(b)(6) (for Operating License applicants) -- Requirement for prompt communications among principal response organizations and to emergency personnel and to the public.

10 CFR 50.47(b)(8) -- Requirement for emergency facilities and equipment to support emergency response.

10 CFR 50.47(b)(9) -- Requirement that adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.54(q) (for Operating Reactors) -- Same requirement as 10 CFR 50.47(b) plus 10 CFR 50, Appendix E.

10 CFR 50, Appendix E, Paragraph IV.E
Requirement for:

"1. Equipment at the site for personnel monitoring;"

"2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;"

"3. Facilities and supplies at the site for decontamination of onsite individuals;"

"4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;"

"5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on site;"

"6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;"

"7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;"

"8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;"

"9. At least one onsite and one offsite communications system; each system shall have a backup power source."

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

"a. Provision for communications with contiguous State/local governments within the plume exposure pathway (emergency planning zone) EPZ. Such communications shall be tested monthly."

"b. Provisions for communication with Federal emergency response organizations. Such communication systems shall be tested annually."

"c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually."

"d. Provisions for communication by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly."

Within this section on emergency response facilities, the Technical Support Center (TSC), Operational Support Center (OSC) and Emergency Operations Facility (EOF) are addressed separately in terms of their functional statements and recommended requirements. The subsections on Documentation and NRC Review and Reference Documents that follow the EOF discussion apply to this entire section on emergency response facilities.

8.2 Technical Support Center (TSC)

8.2.1 Requirements

a. The TSC is the onsite technical support center for emergency response. When activated, the TSC is staffed by predesignated technical, engineering, senior management, and other licensee personnel, and five pre-designated NRC personnel. During periods of activation, the TSC will operate uninterrupted to provide plant management and technical support to plant operations personnel, and to relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations. The TSC will perform EOF functions for the Alert Emergency class and for the Site Area Emergency class and General Emergency class until the EOF is functional.

The TSC will be:

b. Located within the site protected area so as to facilitate necessary interaction with control room, OSC, EOF and other personnel involved with the emergency.

c. Sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation in the center.

d. Structurally built in accordance with the Uniform Building Code.

e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

f. Provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

g. Provided with reliable voice and data communications with the control room and EOF and reliable voice communications with the OSC, NRC Operations Centers and state and local operations centers.

h. Capable of reliable data collection, storage, analysis, display and communication sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

(i) the variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2) that are essential for performance of TSC functions; and

(ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and National Weather Service data available by voice communication for the region in which the plant is located.

Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

Provided with accurate, complete and current plant records (drawings, schematic diagrams, etc.) essential for evaluation of the plant under accident conditions.

Staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and be fully operational within approximately 1 hour after activation.

Designed taking into account good human factors engineering principles.

Operational Support Center (OSC)

8.3.1 Requirements

a. When activated, the OSC will be the onsite area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by control room, TSC and EOF personnel.

The OSC will be:

b. Located onsite to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

c. Capable of reliable voice communications with the control room, TSC and EOF.

Emergency Operations Facility (EOF)

8.4.1 Requirements

a. The EOF is a licensee controlled and operated facility. The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment, development of recommendations for public protective actions, and coordination of emergency response activities with Federal, State and local agencies.

When the EOF is activated, it will be staffed by predesignated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radio-active releases from the plant and to determine dose projections.

The EOF will be:

b. Located and provided with radiation protection features as described in Table 1 (previous guidance approved by the Commission) and with appropriate radiological monitoring systems.

c. Sufficient to accommodate and support Federal, State, local and licensee predesignated personnel, equipment and documentation in the EOF.

d. Structurally built in accordance with the Uniform Building Code.

e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

f. Provided with reliable voice and data communications facilities to the TSC and control room, and reliable voice communication facilities to OSC and to NRC, State and local emergency operations centers.

g. Capable of reliable collection, storage, analysis, display and communication of information on containment conditions, radiological releases and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF functions shall be available in the EOF:

 i) variables from the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2), and

 ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and regional data available via communication from the National Weather Service.

h. Provided with up to date plant records (drawings, schematic diagrams, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

i. Staffed using Table 2 (previous guidance approved by the Commission) as a goal. Reasonable exceptions to goals for the number of additional staff personnel and response times for their arrival should be justified and will be considered by NRC staff.

j. Provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

k. Designed taking into account good human factors engineering principles.

8.4.2 Documentation and NRC Review

The conceptual design for emergency response facilities (TSC, OSC, and EOF) have been submitted to NRC for review. In many cases, the lack of detail in these submittals has precluded an NRC decision of acceptability. Some designs have been disapproved because they clearly did not meet the intent of the applicable regulations. NRC does not intend to approve each design prior to implementation, but rather has provided in this document those requirements which should be satisfied. These requirements provided a degree of flexibility within which licensees can exercise management prerogatives in designing and building emergency response facilities (ERF) that satisfy specific needs of each licensee. The foremost consideration regarding ERFs is that they provide adequate

adequate capabilities of licensees to respond to [illegible]. NUREG guidance on ERFs has been intended to [illegible] issues which the Commission believes should be [illegible] achieving improved capabilities.

Licensees should assure that the design of ERFs [illegible] these requirements. Exemptions from or alternative [illegible] of implementing these requirements should be discussed [illegible] NRC staff and in some cases could require Commission [illegible]. Licensees should continue work on ERFs to complete [illegible] according to schedules that will be negotiated on a plant-specific basis. NRC will conduct appraisals of completed facilities to verify that these requirements have been satisfied [illegible] that ERFs are capable of performing their intended functions. Licensees need not document their actions on each specific [illegible] contained in NUREG-0696 or 0814.

8.4.3 Reference Documents (Emergency Response Facilities)

10 CFR 50.47(b) -- Requirements for emergency facilities and equipment for OLs.

10 CFR 50.54(q) and Appendix E, Paragraph IV.E -- Requirements for emergency facilities and equipment for ORs.

NUREG-0660 -- Description of and implementation schedule for TSC, OSC and EOF.

Eisenhut letter to power reactor licensees 9/13/79 -- Request for commitment to meet requirements

Denton letter to power reactor licensees 10/30/79 -- Clarification of requirements.

NUREG-0654 -- Radiological Emergency Response Plans

NUREG-0696 -- Functional criteria for emergency response facilities.

NUREG-0737 -- Guidance on meteorological monitoring and dose assessment.

Eisenhut letter to power reactor license 2/18/81 -- Commission approved guidance on location, habitability and staff for emergency facilities. Request and deadline for submittal of conceptual design of facilities.

NUREG-0814 (Draft Report for Comment) -- Methodology for evaluation of emergency response facilities.

NUREG-0818 (Draft Report for Comment) -- Emergency Action Levels

Reg. Guide 1.97 (Rev. 2) -- Guidance for variables to be used in selected emergency response facilities.

COMJA-80-37, January 21, 1981 -- Commission approval guidance on EOF location and habitability.

Secretary memorandum S81-19, February 19, 1981 -- Commission approval of NUREG-0696 as general guidance only.

TABLE

EMERGENCY OPERATIONS FACILITY

Option 1 Two Facilities	Option 2 One Facility
Close-in Primary: Reduce Habitability* o within 10 miles o protection factor = 5 o ventilation isolation with HEPA (no charcoal)	o At or Beyond 10 miles. o No special protection factor. o If beyond 20 miles, specific approval required by the Commission, and some provision for NRC site team closer to site. o Strongly recommended location be coordinated with offsite authorities.
Backup EOF o between 10-20 miles o no separate, dedicated facility o arrangements for portable backup equipment o strongly recommended location be coordinated with offsite authorities o continuity of dose projection and decision making capability	

For both Options:

- located outside security boundary
- space for about 10 NRC employees
- none designated for severe phenomena, e.g., earthquakes

labitability requirements are only for the part of the EOF in which dose assessments communications and decision making take place.

f a utility has begun construction of a new building for an EOF that is located with 5 miles, that new acility is acceptable (with less than protection factor of 5 and ventilation isolation and HEPA) provided hat a backup EOF similar to "B" in Option 1 is provided.

TABLE 2

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions On Shift*	30 min.	60 min.
Plant Operations and Assessment of Operational Aspects		Shift supervisor (SRO)	1	--	--
		Shift foreman (SRO)	1	--	--
		Control-room operators	2	--	--
		Auxiliary operators	2		
Emergency Direction and Control (Emergency Coordinator)***		Shift technical advisor, shift supervisor, or designated facility manager	1**	--	--
Notification/ Communication****	Notify licensee, state local, and federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency operations facility (EOF) director	Senior manager	--	--	1
	Offsite dose assessment	Senior health physics (HP) expertise	--	1	--
	Offsite surveys		--	2	2
	Onsite (out-of-plant)		--	1	1
	Inplant surveys	HP technicians	1	1	1
	Chemistry/radio-chemistry	Rad/chem technicians	1	--	1

NOTE: Source of this table is NUREG-0654, "Functional Criteria for Emergency Response Facilities."

TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions: On Shift*	30 min.	60 min.
Plant System Engineering, Repair and Corrective Actions	Technical support	Shift technical advisory	1	--	--
		Core/thermal hydraulics	--	1	--
		Electrical	--	--	1
		Mechanical	--	--	1
	Repair and corrective actions	Mechanical maintenance/	1**	--	1
		Radwaste operator			1
		Electrical maintenance/	1**	1	1
		instrument and control		1	--
		(I&C) technician	--	1	--
Protective Actions (In-Plant)	Radiation protection: a. Access control b. HP Coverage for repair, corrective actions, search and rescue first-aid, & firefighting c. Personnel monitoring d. Dosimetry	HP technicians	2**	2	2
Firefighting	--	--	Fire brigade per technical specification	Local support	
Rescue Operations and First-Aid	--	--	2**	Local support	

TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions On Shift*	30 min.	60 min.
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security personnel	All per security plan		
		Total	10	11	15

*For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control-room operator, and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

**May be provided by shift personnel assigned other functions.

***Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Directo of minute-to-minute facility operations remains with senior manager in technical support center or control room.

****May be performed by engineering aide to shift supervisor.

Emergency Action Level Technical Bases Document

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ACRONYMS

AC	Alternating Current
ATWS	Anticipated Transient Without Scram
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CMT	Containment
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
DC	Direct Current
DOT	Department of Transportation
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
ESF	Engineered Safeguards Feature
GE	General Emergency
HPSI	High Pressure Safety Injection
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IGLD	International Great Lakes Datum
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LFL	Lower Flammability Limit
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MAT	Main Auxiliary Transformer

MSIV	Main Steam Isolation Valve
mR	milliRem
Mw	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Rem
RAT	Reserve Auxiliary Transformer
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RVLIS	Reactor Vessel Level Indicating System
SAE	Site Area Emergency
SG	Steam Generator
SI	Safety Injection
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
SW	Service Water
TAT	Tertiary Auxiliary Transformer
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel
TSC	Technical Support Center
UE	Unusual Event
USAR	Updated Final Safety Analysis Report
WOG	Westinghouse Owners Group

1. PURPOSE

This document provides the detailed set of Emergency Action Levels (EALs) applicable to the Kewaunee Nuclear Plant (KNP) and the associated Technical Bases using the EAL development methodology found in NEI 99-01 Revision 4 [Ref. 2.1]. Personnel responsible for implementation of EPIP-AD-02 "Emergency Class Determination" [Ref. 2.2], and the Emergency Action Level Matrix [Ref. 2.3] may use this document as a technical reference and an aid in EAL interpretation.

The primary tool for determining the emergency classification level is the Emergency Action Level Matrix. The user of the Emergency Action Level Matrix may (but is not required to) consult the EAL Technical Basis Document in order to obtain additional information concerning the EALs under classification consideration.

2. REFERENCES

2.1 NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels, January 2003

2.2 KNPP Technical Specifications, Section 1.0 Definitions, Amendments 162, 172 and 176.

3. DISCUSSION

3.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the KNPP Emergency Plan.

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG 0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revision 4 represents the most recent NRC endorsed methodology per RG 1.101 Rev 4, "Emergency Planning and Preparedness for Nuclear Power Reactors." Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Addressing initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Using NEI 99-01 Rev. 4, KNPP conducted an EAL implementation upgrade project that produced the EALs discussed herein. While the upgraded EALs are site-specific, an objective of the project was to ensure to the extent possible EAL conformity and consistency between the NMC plant sites.

3.2 Key Definitions in EAL Methodology

The following definitions apply to the generic EAL methodology:

EMERGENCY CLASS: One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping of normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time sensitive onsite and off site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Section 3.3 provides further discussion of the emergency classes.

INITIATING CONDITION (IC): One of a predetermined subset of nuclear power plant conditions when either the potential exists for a radiological emergency, or such an emergency has occurred.

- An IC is an emergency condition which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency.

- It can be a continuous, measurable function that is outside technical specifications, such as elevated RCS temperature or falling reactor coolant level (a symptom).

- It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

EMERGENCY ACTION LEVEL (EAL): A pre determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

- There are times when an EAL will be a threshold point on a measurable continuous function, such as a primary system coolant leak that has exceeded technical specifications.

- At other times, the EAL and the IC will coincide, both identified by a discrete event that places the plant in a particular emergency class.

3.3 Recognition Categories

ICs and EALs are grouped in one of several categories. This classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

- R - Abnormal Rad Levels/Radiological Effluent

- C - Cold Shutdown./ Refueling System Malfunction

- F - Fission Product Barrier Degradation

- H - Hazards

- S - System Malfunction

Some recognition categories are further divided into one or more subcategories depending on the types and number of plant conditions that dictate emergency classifications. An EAL may or may not exist for each subcategory at all four classification levels. Similarly, more than one EAL may exist for a subcategory in a given emergency classification when appropriate (i.e., no EAL at the General Emergency level but three EALs at the Unusual Event level).

3.4 Emergency Class Descriptions

There are three considerations related to the emergency classes. These are:

- The potential impact on radiological safety, either as now known or as can be reasonably projected.

- How far the plant is beyond its predefined design, safety and operating envelopes.

- Whether or not conditions that threaten health are expected to be confined to within the site boundary.

The ICs deal explicitly with radiological safety affect by escalating from levels corresponding to releases within regulatory limits to releases beyond EPA Protective Action Guideline (PAG) plume exposure levels.

UNUSUAL EVENT: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

- Potential degradation of the level of safety of the plant is indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO) allowable action statement time for achieving required mode change.

- Precursors of more serious events may be included because precursors represent a potential degradation in the level of safety of the plant.

- Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or offsite response (e.g., dose consequences of less than 10 millirem).

ALERT: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

SITE AREA EMERGENCY: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline (PAG) exposure levels beyond the site boundary.

- The discriminator (threshold) between Site Area Emergency and General Emergency is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary.

- This threshold, in addition to dynamic dose assessment considerations discussed in the EAL guidelines, clearly addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency.

GENERAL EMERGENCY: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

- The bottom line for the General Emergency is whether evacuation or sheltering of the general public is indicated based on EPA PAGs and, therefore, should be interpreted to include radionuclide release regardless of cause.

- To better assure timely notification, EALs in this category are primarily expressed in terms of plant function status, with secondary reliance on dose projection. In terms of fission product barriers, loss of two barriers with loss or potential loss of the third barrier constitutes a General Emergency.

3.5 Operating Mode Applicability

Technical Specifications [Ref. 2.4] provides definitions for the following operating modes:

1 Operating (OP)

Reactivity Δk/k is LESS THAN Technical Specification minimum required (0.25%) and EQUAL TO OR GREATER than 2% fission power.

2 Hot Standby (HSB)

Reactivity Δk/k is LESS THAN Technical Specification minimum required (0.25%) and LESS THAN 2% fission power.

3 Hot Shutdown (HSD)

Reactivity Δk/k as specified in the Core Operating Limits Report with coolant temperature (Tavg) GREATER THAN OR EQUAL TO 540°F.

4 Intermediate Shutdown (ISD)

Reactivity Δk/k as specified in the Core Operating Limits Report with coolant temperature (Tavg) LESS THAN 540°F and GREATER THAN 200°F.

5 Cold Shutdown (CSD)

Reactivity Δk/k GREATER THAN OR EQUAL TO Technical Specification minimum required (-1%) with coolant temperature (Tavg) LESS THAN OR EQUAL TO 200°F.

6 Refueling (REF)

Reactivity Δk/k GREATER THAN OR EQUAL TO Technical Specification minimum required for refueling operations (-5%) and coolant temperature (Tavg) LESS THAN OR EQUAL TO 140°F.

In addition to the Technical Specification operating modes, NEI 99-01 [Ref. 1] defines the following additional mode:

D Defueled

All reactor fuel removed from Reactor Vessel (full core off load during refueling or extended outage)

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

Recognition categories are associated with the operating modes listed in the following matrix:

	Recognition Category				
Mode	R	C	F	H	S
Operations	X		X	X	X
Hot Standby	X		X	X	X
Hot Shutdown	X		X	X	X
Intermediate Shutdown	X		X	X	X
Cold Shutdown	X	X		X	
Refueling	X	X		X	
Defueled	X	X		X	

3.6 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss of or potential loss of one or more of the three fission product barriers. "Loss" and "potential loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials and "potential loss" means imminent loss of the barrier.

The primary fission product barriers are:

- Fuel Cladding (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

- Reactor Coolant System (RCS): The reactor vessel shell, vessel head, vessel nozzles and penetrations and all primary systems directly connected to the reactor vessel up to the first containment isolation valve comprise the RCS barrier.

- Containment (CMT): The vapor containment structure and all isolation valves required to maintain containment integrity under accident conditions comprise the Containment barrier.

3.7 Emergency Classification Based on Fission Product Barrier Degradation

The following criteria are the bases for event classification related to fission product barrier loss or challenge:

- Unusual Event:

  Any loss or any potential loss of Containment

- Alert:

  Any loss or any potential loss of either Fuel Cladding or RCS

- Site Area Emergency:

  Loss or potential loss of any two barriers

- General Emergency:

  Loss of any two barriers and loss or potential loss of third barrier

3.8 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the Critical Safety Function Status Trees (CSFSTs). While the symptoms that drive operator actions specified in the CSFSTs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. Where these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the CSFSTs, classification of emergencies using these EALs is not dependent upon Emergency Operating Procedures (EOP) entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

3.9 Symptom Based vs. Event Based Approach

To the extent possible, the EALs are symptom based. That is, the action level is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. But, a purely symptom based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

Category R - Abnormal Rad Levels/Radiological Effluent and Category F - Fission Product Barrier Degradation are primarily symptom-based. The symptoms are indicative of actual or potential degradation of either fission product barriers or personnel safety.

Other categories tend to be event-based. For example, System Malfunctions are abnormal and emergency events associated with vital plant system failures, while Hazards are those non-plant system related events that have affected or may affect plant safety.

3.10 Treatment of Emergency Class Upgrading

The emergency class is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency.

3.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Emergency Director classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined. In other situations, further analyses (e.g., coolant sampling) may be necessary.

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

- An ATWS represents a failure of a front line Reactor Protection System (RPS) designed to protect the health and safety of the public.
- The affect of an earthquake on plant equipment and structures may not be readily apparent until investigations are conducted.

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Rev. 1, Section 3 should be applied.

3.12 Imminent EAL Thresholds

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes. Explicit EALs, specifying use of Emergency Director judgment, are given in the Hazards and Fission Product Barrier Degradation categories.

4. TECHNICAL BASES INFORMATION

4.1 Recognition Category Organization

The technical bases of the EALs are provided under Recognition Categories R, C, F, H and S of this document. A table summarizing the Initiating Conditions introduces each category. The tables provide an overview of how the ICs are related under each emergency class. ICs within each category are listed according to classification (as applicable) in the following order: Unusual Event, Alert, Site Area Emergency, and General Emergency.

For Recognition Category F, Table F-0 defines the emergency classifications associated with barrier loss and potential loss. Table F-1 lists the thresholds associated with the loss and potential loss of each fission product barrier. The presentation method shown for Table F-1 was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Basis discussion of the thresholds immediately follows Table F-1.

4.2 Initiating Condition Structure

ICs in Recognition Categories R, C, H and S are structured in the following manner:

- Recognition Category Title
- IC Identifier:
  - o First character identifies the category by letter (R, C, H and S)
  - o Second character identifies the emergency classification level (U for Unusual Event, A for Alert, S for Site Area Emergency, and G for General Emergency)
  - o Third character is the numerical sequence as given in Revision 4 of NEI 99-01 [Ref. 1] (e.g., SA2). Due to document revisions, certain NEI ICs have been deleted, leaving gaps in the numerical sequence.
- Emergency Class: Unusual Event, Alert, Site Area Emergency, or General Emergency
- IC Description
- Operating Mode Applicability: Refers to the operating mode during which the IC/EAL is applicable

- Emergency Action Level(s): EALs are the conditions applicable to the criteria of the IC and are used to determine the need to classify an event/condition. If more than one EAL is applicable to an IC, emergency classification is required when any EAL within the IC reaches the EAL threshold. To clarify this intent, ICs with multiple EALs include a parenthetical phrase in the EAL title line, indicating that each constitutes an emergency classification. For example, the phrase "(RA1.1 or RA1.2)" indicates that either EAL is a Notification of Unusual Event.

- Basis: Provides information that explains the IC and EAL(s). Plant source document references are provided as needed to substantiate site-specific information included in the EALs and bases.

4.3 EAL Identification

The EAL identifier is the IC identifier followed by a period and sequence number (e.g., RU1.1, RU1.2, etc.). If only one EAL is assigned to an IC, the EAL is given the number one.

The primary purpose of the EAL identifier is to uniquely distinguish each classifiable condition. Secondary purposes are to assist location of an EAL within the EAL classification scheme and to announce the emergency classification level.

5. DEFINITIONS

In the ICs and EALs, selected words are in uppercase print. These words are defined terms. Definitions are provided below.

AFFECTING SAFE SHUTDOWN: event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not "AFFECTING SAFE SHUTDOWN."

> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is "AFFECTING SAFE SHUTDOWN."

BOMB: an explosive device suspected of having sufficient force to damage plant systems or structures.

CIVIL DISTURBANCE: a group of unexpected or unauthorized individuals violently protesting station operations or activities at the site.

CONFINEMENT BOUNDARY: the barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE: defined by N-CCI-56A, "Open Containment Boundary Tracking".

EXPLOSION: a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION: an attempt to cause an action at the station by threat of force.

FAULTED: a steam generator, the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

HOSTAGE: a person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: one or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

INTRUSION / INTRUDER: person(s) present in a specified area without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by a HOSTILE FORCE.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

PROTECTED AREA: boundary within the security isolation zone.

RUPTURED: In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

SABOTAGE: deliberate damage, misalignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may NOT meet the definition of SABOTAGE until this determination is made by security supervision.

SIGNIFICANT TRANSIENT: an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

STRIKE ACTION: a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONs.

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: An indication, report, or condition is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator operability, the condition existence, or the report accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE: damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA: Area within the PROTECTED AREA, which contains equipment, systems, components, or material; the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

6. EMERGENCY ACTION LEVEL CATEGORIES

R - Abnormal Rad Levels/Radiological Effluent

C - Cold Shutdown / Refueling System Malfunction

F - Fission Product Barrier Degradation

H - Hazards

S - System Malfunction

Table R-0

Recognition Category R

Abnormal Rad Levels / Radiological Effluent

INITIATING CONDITION MATRIX

UE		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
RU1	**Any UNPLANNED Release of Gaseous or Liquid Radio-activity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual for 60 Minutes or Longer.** *Op. Modes: All*	RA1	**Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual for 15 Minutes or Longer.** *Op. Modes: All*	RS1	**Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release.** *Op. Modes: All*	RG1	**Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.** *Op. Modes: All*
RU2	**Unexpected Increase in Plant Radiation.** *Op. Modes: All*	RA3	**Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown** *Op. Modes: All*				
		RA2	**Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.** *Op. Modes: All*				

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1

Initiating Condition -- UNUSUAL EVENT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual for 60 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels: (RU1.1 or RU1.2 or RU1.3)

RU1.1. VALID reading on any effluent monitor that is GREATER THAN two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

Auxiliary Building	Action Value
R-13 Aux. Bldg. Vent Exhaust	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	2.62E+05 cpm
Reactor Building	
R-12 Containment Gas	4.41E+05 cpm
R-21 Containment Vent	4.40E+05 cpm
Liquid Radwaste	
R-18 Waste Disposal System Liquid	2 X Calculated ODCM Setpoint

RU1.2. VALID reading on one or more of the following radiation monitors that is GREATER THAN the reading shown for 60 minutes or longer.

Liquid Radwaste	Action Value
R-16 Containment FCU SW Return	3.38E+05 cpm
R-19 S/G Blowdown Liquid	2.58E+06 cpm
R-20 Aux Bldg SW Return	1.03E+05 cpm

RU1.3. *Confirmed sample analyses for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times the ODCM limit.*

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. KNPP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual

(ODCM) [Ref. 2, 3]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in ICs RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold for this IC.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RU1.1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ODCM limit and releases are not terminated within 60 minutes. The "UE" values are two times the monitor high alarm setpoints or ODCM release limits. The setpoints are established to ensure the ODCM release limits are not exceeded [Ref. 2, 3].These alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit. Each liquid discharge permit includes a value for R-18, calculated in accordance with the ODCM that will vary based on the discharge flow rate. Therefore 2 X Calculated ODCM Setpoint was used as the threshold. Escalation will be based on radiation readings increasing per the following:

Normal Effluent Release Monitor Classification Thresholds				
Monitor	GE	SAE	Alert	UE
<u>Auxiliary Building</u>				
01-05 Aux. Bldg. SPING Lo Range	—	—	—	—
01-07 Aux. Bldg. SPING Mid Range	1.00E+05 cpm	1.00E+04 cpm	—	—
01-09 Aux. Bldg. SPING Hi Range	1.00E+02 cpm	1.00E+01 cpm	—	—
R-13 Aux. Bldg. Vent Exhaust	—	—	2.61E+07 cpm	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	—	—	2.62E+07 cpm	2.62E+05 cpm
<u>Reactor Building</u>				
02-05 Rx Bldg. Vent SPING Lo Range	—	—	—	—
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+04 cpm	2.00E+03 cpm	—	—
02-09 Rx Bldg. Vent SPING Hi Range	2.00E+01 cpm	—	—	—
R-12 Containment Gas	—	—	4.41E+07 cpm	4.41E+05 cpm
R-21 Containment Vent	—	—	4.40E+07 cpm	4.40E+05 cpm
<u>Liquid Radwaste</u>				
R-18 Waste Disposal System Liquid	N/A	N/A	200 X Calculated ODCM Setpoint	2 X Calculated ODCM Setpoint

RU1.2 is intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology. The "UE" values are two times the monitor high alarm setpoints or ODCM release limits. The setpoints are established to ensure the ODCM release limits are not exceeded [Ref. 2, 3]. Escalation will be based on radiation readings increasing per the following:

Abnormal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
<u>Main Steam Line (PORV)</u>				
R-31 'A' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-32 'A' Steamline High Range	1.77E+00 R/hr	—	—	—
R-33 'B' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-34 'B' Steamline High Range	1.77E+00 R/hr	—	—	—
<u>Main Steam Line (SG Safety)</u>				
R-31 'A' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-32 'A' Steamline High Range	—	—	—	—
R-33 'B' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-34 'B' Steamline High Range	—	—	—	—
<u>Liquid Radwaste</u>				
R-16 Containment Fcu SW Return	N/A	N/A	3.38E+07 cpm	3.38E+05 cpm
R-19 S/G Blowdown Liquid	N/A	N/A	2.58E+08 cpm	2.58E+06 cpm
R-20 Aux Bldg SW Return	N/A	N/A	1.03E+07 cpm	1.03E+05 cpm

RU1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

RU1.1 and RU1.2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.

KNPP Basis Reference(s):

1. USAR Section 11.2.3 Radiation Monitoring System, Rev. 18
2. KNPP ODCM Section 2.0 Gaseous Effluents, Rev. 8
3. KNPP ODCM Section 1.2 Liquid Effluent Monitor Setpoint Determination, Rev. 8
4. C11620, Evaluation of Radiological Effluent Monitor Response Action Levels, Rev. 0

RU2

Initiating Condition -- UNUSUAL EVENT

Unexpected Rise in Plant Radiation.

Operating Mode Applicability: All

Emergency Action Levels: (RU2.1 or RU2.2)

RU2.1. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by Spent Fuel Pool low water level alarm setpoint (3 ft 4 in. below floor, SER 159/160) OR visual observation

AND

Any UNPLANNED VALID Direct Area Radiation Monitor reading rises as indicated by:

- R-2 Containment Area ALERT Alarm
- R-5 Fuel Handling Area ALERT Alarm
- R-10 New Fuel Pit Area ALERT Alarm

RU2.2. Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels.
*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

This IC addresses increased radiation levels as a result of water level decreases above the Reactor Vessel flange or events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via RU2.1 is appropriate given their potential for increased doses to plant staff. Classification as an UE is warranted as a precursor to a more serious event. Indications include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. If available, security video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the reading on an area radiation monitor located on the refueling bridge may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss. For refueling events where the water level drops below the Reactor Vessel flange classification would be via CU2. This event escalates to an Alert per IC RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in Operating through Intermediate Shutdown operating modes.

The Spent Fuel Pool (SFP) low level alarm is actuated by LA-16640-02 (SER 159) and LA-16641-02 (SER 160) at 3 ft 4 in. below floor level. The North (A) and South (B) Spent Fuel Pools are located in the Auxiliary Building refueling area. The pools can be isolated from each other by a removable gate, which is normally removed. The top of each pool is at 649 ft 6 in. el. and the bottom is at 608 ft el. Fuel occupies the bottom 14 ft. [Ref. 3].

RU2.2 addresses UNPLANNED increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. This event escalates to an Alert per IC RA3 if the increase in dose rates impedes personnel access necessary for safe operation.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

KNPP Basis Reference(s):

1. Control Room Alarm Response Procedure 47055-N Spent Fuel Pool Abnormal Beta Window Box 05-N5, Rev. C
2. Operating Procedure A-SFP-21 Abnormal Spent Fuel Pool Cooling and Cleanup System Operation, Rev. T
3. KNPP System Description 21 Spent Fuel Pool Cooling and Cleanup System (SFP), Rev. 1
4. USAR Section 11.2.3 Radiation Monitoring System, Rev. 18
5. Control Room Alarm Response Procedure 47011-B Radiation Indication High Beta Window Box 01-B1, Rev. D
6. E-2021 Integrated Logic Diagram Radiation Monitoring, Rev. X

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA1

Initiating Condition -- ALERT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual for 15 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels: (RA1.1 or RA1.2 or RA1.3)

RA1.1. VALID reading on any effluent monitor GREATER THAN 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

Auxiliary Building	Action Value
R-13 Aux. Bldg. Vent Exhaust	2.61E+07 cpm
R-14 Aux. Bldg. Vent Exhaust	2.62E+07 cpm
Reactor Building	
R-12 Containment Gas	4.41E+07 cpm
R-21 Containment Vent	4.40E+07 cpm
Liquid Radwaste	
R-18 Waste Disposal System Liquid	200 X Calculated ODCM Setpoint

RA1.2. VALID reading on one or more of the following radiation monitors GREATER THAN the reading shown for 15 minutes or longer:

Liquid Radwaste	Action Value
R-16 Containment FCU SW Return	3.38E+07 cpm
R-19 S/G Blowdown Liquid	2.58E+08 cpm
R-20 Aux Bldg SW Return	1.03E+07 cpm

RA1.3. Confirmed sample analyses for gaseous or liquid release indicate concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ODCM limit.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. KNPP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual

(ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in ICs RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.
RA1.1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. The "Alert" values shown for each monitor are two hundred times the alarm setpoints or calculated ODCM release limits as specified in Reference 4. The setpoints are established to ensure the ODCM release limits are not exceeded [Ref. 2, 3]. The alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit. Each liquid discharge permit includes a value for R-18, calculated in accordance with the ODCM, that will vary based on the discharge flow rate, therefore "200 X Calculated ODCM Setpoint" was used as the threshold. Escalation will be based on radiation readings increasing per the following:

Normal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Auxiliary Building				
01-05 Aux. Bldg. SPING Lo Range	—	—	—	—
01-07 Aux. Bldg. SPING Mid Range	1.00E+05 cpm	1.00E+04 cpm	—	—
01-09 Aux. Bldg. SPING Hi Range	1.00E+02 cpm	1.00E+01 cpm	—	—
R-13 Aux. Bldg. Vent Exhaust	—	—	2.61E+07 cpm	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	—	—	2.62E+07 cpm	2.62E+05 cpm
Reactor Building				
02-05 Rx Bldg. Vent SPING Lo Range	—	—	—	—
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+04 cpm	2.00E+03 cpm	—	—
02-09 Rx Bldg. Vent SPING Hi Range	2.00E+01 cpm	—	—	—
R-12 Containment Gas	—	—	4.41E+07 cpm	4.41E+05 cpm
R-21 Containment Vent	—	—	4.40E+07 cpm	4.40E+05 cpm
Liquid Radwaste				
R-18 Waste Disposal System Liquid	N/A	N/A	200 X Calculated ODCM Setpoint	2 X Calculated ODCM Setpoint

RA1.2 addresses effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared) [Ref. 1]. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default

source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology. The "Alert" values for each monitor are two hundred times the alarm setpoints or calculated ODCM release limits as specified in Reference 4. The setpoints are established to ensure the ODCM release limits are not exceeded [Ref. 2, 3]. Escalation will be on based radiation readings increasing per the following:

Abnormal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Main Steam Line (PORV)				
R-31 'A' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-32 'A' Steamline High Range	1.77E+00 R/hr	—	—	—
R-33 'B' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-34 'B' Steamline High Range	1.77E+00 R/hr	—	—	—
Main Steam Line (SG Safety)				
R-31 'A' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-32 'A' Steamline High Range	—	—	—	—
R-33 'B' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-34 'B' Steamline High Range	—	—	—	—
Liquid Radwaste				
R-16 Containment Fcu SW Return	N/A	N/A	3.38E+07 cpm	3.38E+05 cpm
R-19 S/G Blowdown Liquid	N/A	N/A	2.58E+08 cpm	2.58E+06 cpm
R-20 Aux Bldg SW Return	N/A	N/A	1.03E+07 cpm	1.03E+05 cpm

RA1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

RA1.1 and RA1.2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.

Due to the uncertainty associated with meteorology, emergency implementing procedures call for the timely performance of dose assessments using actual (real-time) meteorology in the event of a gaseous radioactivity release of this magnitude. The results of these assessments should be compared to the ICs RS1 and RG1 to determine if the event classification should be escalated.

KNPP Basis Reference(s):

1. USAR Section 11.2.3 Radiation Monitoring System, Rev. 18
2. KNPP ODCM Section 2.0 Gaseous Effluents, Rev. 8
3. KNPP ODCM Section 1.2 Liquid Effluent Monitor Setpoint Determination, Rev. 8
4. C11620, Evaluation of Radiological Effluent Monitor Response Action Levels, Rev. 0

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA2

Initiating Condition-- ALERT

Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

Emergency Action Levels: (RA2.1 or RA2.2)

RA2.1. A VALID radiation indication high alarm or reading on one or more of the following radiation monitors resulting from damage to irradiated fuel or loss of water level:

- R-2 Containment Area
- R-5 Fuel Handling Area
- R-13 or R-14 Aux Bldg Vent Exhaust
- R-11 or R-12 Containment Particulate / Gas Ventilation
- R-21 Containment Vent

RA2.2. Water level LESS THAN 50% Wide Range Refueling Water Level OR GREATER THAN 14 feet below top of Spent Fuel Pool that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant. These events escalate from IC RU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in IC EU1.

RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3] of fuel uncovery and/or fuel damage. *Increased readings on ventilation monitors may be indication of a radioactivity release from the* fuel, confirming that damage has occurred. Increased background at the monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, *"KR-85 Hazards from Decayed Fuel"* was considered in establishing radiation monitor EAL thresholds and there is no impact on this EAL.

In RA2.2, indications include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Wide Range Refueling Water Level is measured by L9053A for channel A and L9054A for channel B. If available, security video cameras may allow remote observation. The top of each pool is at 649 ft 6 in. el. and the bottom is at 608 ft el. Fuel occupies the bottom 14 ft. [Ref. 4]. Declaration may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

Escalation, if appropriate, would occur via IC RS1 or RG1 or Emergency Director judgment.

KNPP Basis Reference(s):

1. E-2021 Integrated Logic Diagram Radiation Monitoring, Rev. X
2. Control Room Alarm Response Procedure 47055-N Spent Fuel Pool Abnormal Beta Window Box 05-N5, Rev. C
3. Operating Procedure A-SFP-21 Abnormal Spent Fuel Pool Cooling and Cleanup System Operation, Rev. T
4. KNPP System Description 21, Spent Fuel Pool Cooling and Cleanup System (SFP), Rev. 1
5. Manipulator Crane drawing XK-113557-5, Rev. D
6. N-RHR-34C RHR Operation at a Reduced Inventory Condition, Rev. N
7. C11619 Determination of Cavity Level EAL RA2.2, Rev. 0

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA3

Initiating Condition -- ALERT

Release of Radioactive Material or Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability: All

Emergency Action Levels: (RA3.1 or RA3.2)

RA3.1. VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (Rad monitor R-1)
OR
Central Alarm Station (Rad monitor R-1)
OR
Secondary Alarm Station (by survey)

RA3.2. Any VALID radiation monitor reading GREATER THAN 6 R/hr in areas requiring infrequent access to maintain plant safety functions.

- Auxiliary Building
- Safeguards Alley
- Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel)
- Screenhouse/Forebay
- Relay Room
- Safeguard Battery Room

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mR/hr in the control room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, an SE or GE may be indicated by the fission product barrier matrix ICs.

This IC is not meant to apply to increases in the containment radiation monitors, as these are events which are addressed in the fission product barrier matrix ICs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.)

For RA3.1 areas requiring continuous occupancy include the Control Room and the central alarm station (CAS). The CAS has no installed radiation monitoring capability [Ref. 3].The value of

15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, *"Clarification of TMI Action Plan Requirements"* [Ref. 1, 2], provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

For RA3.2 areas requiring infrequent access, the basis of the 6 R/hr value is as follows:

> The KNPP annual administrative personnel exposure limit is 2 Rem/Year. Assuming an emergency worker is at his administrative limit, any emergency worker needing access to a plant area for the safe shutdown of the plant could receive up to an additional 3 Rem without exceeding the legal 10CFR20 annual exposure limit of 5 Rem [Ref. 4] and thus the need for emergency exposure authorization. Assuming that an activity required to be performed in the plant would, on average, require a 30 minute stay time in that area, an area exposure rate of 6 R/hr would not unduly impede access to areas necessary for safe plant shutdown.

As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. RA3.2 provides the list of safe shutdown areas requiring infrequent access. The listed areas contain functions and systems required for the safe shutdown of the plant. KNPP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode [Ref 5].

In-plant radiation surveys and Area Radiation Monitor (ARM) readings are methods available to assess this EAL. Radiation monitors are not specified in the EAL wording because portable monitoring devices may be used to determine area accessibility. It would then be possible to erroneously exclude information gained from portable monitor surveys when interpreting the EAL.

KNPP Basis Reference(s):

1. GDC 19, January 1, 2004
2. NUREG-0737, "Clarification of TMI Action Plan Requirements", Section III.D.3
3. E-2021 Integrated Logic Diagram Radiation Monitoring, Rev. X
4. EPIP-AD-11, Emergency Radiation Controls, Rev. T
5. KNPP Fire Protection Program Plan Section 5.19, Rev. 5

RS1

Initiating Condition -- SITE AREA EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

Emergency Action Levels: (RS1.1 or RS1.2 or RS1.3)

Note: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

RS1.1. VALID reading on any monitors listed that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Auxiliary Building	Action Value
01-07 Aux. Bldg. SPING Mid Range	1.00E+04 cpm
01-09 Aux. Bldg. SPING Hi Range	1.00E+01 cpm
Reactor Building	
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+03 cpm
Main Steam Line (PORV)	
R-31 'A' Steamline Lo Range	1.77E+02 mR/hr
R-33 'B' Steamline Lo Range	1.77E+02 mR/hr
Main Steam Line (SG Safety)	
R-31 'A' Steamline Lo Range	8.30E+01 mR/hr
R-33 'B' Steamline Lo Range	8.30E+01 mR/hr

RS1.2. Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary.

RS1.3. Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour, at or beyond the site boundary;

OR

Analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation, at or beyond the site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone, e.g., fuel handling accident in spent fuel building.

The TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. The monitor list in RS1.1 includes monitors on all potential release pathways [Ref. 1, 3, 4].

The "SAE" effluent monitor readings are derived from Reference 2.

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

Escalation will be on based radiation readings increasing per the following:

Normal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Auxiliary Building				
01-05 Aux. Bldg. SPING Lo Range	—	—	—	—
01-07 Aux. Bldg. SPING Mid Range	1.00E+05 cpm	1.00E+04 cpm	—	—
01-09 Aux. Bldg. SPING Hi Range	1.00E+02 cpm	1.00E+01 cpm	—	—
R-13 Aux. Bldg. Vent Exhaust	—	—	2.61E+07 cpm	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	—	—	2.62E+07 cpm	2.62E+05 cpm
Reactor Building				
02-05 Rx Bldg. Vent SPING Lo Range	—	—	—	—
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+04 cpm	2.00E+03 cpm	—	—
02-09 Rx Bldg. Vent SPING Hi Range	2.00E+01 cpm	—	—	—
R-12 Containment Gas	—	—	4.41E+07 cpm	4.41E+05 cpm
R-21 Containment Vent	—	—	4.40E+07 cpm	4.40E+05 cpm
Liquid Radwaste				
R-18 Waste Disposal System Liquid	N/A	N/A	200 X Calculated ODCM Setpoint	2 X Calculated ODCM Setpoint

Abnormal Effluent Release Monitor Classification Thresholds				
Monitor	GE	SAE	Alert	UE
<u>Main Steam Line (PORV)</u>				
R-31 'A' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-32 'A' Steamline High Range	1.77E+00 R/hr	—	—	—
R-33 'B' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-34 'B' Steamline High Range	1.77E+00 R/hr	—	—	—
<u>Main Steam Line (SG Safety)</u>				
R-31 'A' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-32 'A' Steamline High Range	—	—	—	—
R-33 'B' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-34 'B' Steamline High Range	—	—	—	—
<u>Liquid Radwaste</u>				
R-16 Containment Fcu SW Return	N/A	N/A	3.38E+07 cpm	3.38E+05 cpm
R-19 S/G Blowdown Liquid	N/A	N/A	2.58E+08 cpm	2.58E+06 cpm
R-20 Aux Bldg SW Return	N/A	N/A	1.03E+07 cpm	1.03E+05 cpm

KNPP Basis Reference(s):

1. USAR Section 11.2.3 Radiation Monitoring System, Rev. 18
2. C11620, Evaluation of Radiological Effluent Monitor Response Action Levels, Rev. 0
3. EPIP-RET-02B Gaseous Effluent Release Path, Radioactivity, and Release Rate Determination, Rev. T
4. ODCM Section 2.0 Gaseous Effluents, Rev. 8

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RG1

Initiating Condition -- GENERAL EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

Emergency Action Levels: (RG1.1 or RG1.2 or RG1.3)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RG1.1. VALID reading on any monitors listed that exceeds or expected to exceed the reading shown for 15 minutes or longer:

<u>Auxiliary Building</u>	<u>Action Value</u>
01-07 Aux. Bldg. SPING Mid Range	1.00E+05 cpm
01-09 Aux. Bldg. SPING Hi Range	1.00E+02 cpm
<u>Reactor Building</u>	
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+04 cpm
02-09 Rx Bldg. Vent SPING Hi Range	2.00E+01 cpm
<u>Main Steam Line (PORV)</u>	
R-31 'A' Steamline Lo Range	1.77E+03 mR/hr
R-32 'A' Steamline High Range	1.77E+00 R/hr
R-33 'B' Steamline Lo Range	1.77E+03 mR/hr
R-34 'B' Steamline High Range	1.77E+00 R/hr
<u>Main Steam Line (SG Safety)</u>	
R-31 'A' Steamline Lo Range	8.30E+02 mR/hr
R-33 'B' Steamline Lo Range	8.30E+02 mR/hr

RG1.2. Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary.

RG1.3. Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour, at or beyond site boundary.

OR

Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.
The monitor list in RG1.1 includes monitors on all potential release pathways [Ref. 1, 3, 4].

The "GE" effluent monitor readings are derived from Reference 2.

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

Radiation Monitor readings for all classification levels:

Normal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Auxiliary Building				
01-05 Aux. Bldg. SPING Lo Range	—	—	—	—
01-07 Aux. Bldg. SPING Mid Range	1.00E+05 cpm	1.00E+04 cpm	—	—
01-09 Aux. Bldg. SPING Hi Range	1.00E+02 cpm	1.00E+01 cpm	—	—
R-13 Aux. Bldg. Vent Exhaust	—	—	2.61E+07 cpm	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	—	—	2.62E+07 cpm	2.62E+05 cpm
Reactor Building				
02-05 Rx Bldg. Vent SPING Lo Range	—	—	—	—
02-07 Rx Bldg. Vent SPING Mid Range	2.00E+04 cpm	2.00E+03 cpm	—	—
02-09 Rx Bldg. Vent SPING Hi Range	2.00E+01 cpm	—	—	—
R-12 Containment Gas	—	—	4.41E+07 cpm	4.41E+05 cpm
R-21 Containment Vent	—	—	4.40E+07 cpm	4.40E+05 cpm
Liquid Radwaste				
R-18 Waste Disposal System Liquid	N/A	N/A	200 X Calculated ODCM Setpoint	2 X Calculated ODCM Setpoint

Abnormal Effluent Release Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Main Steam Line (PORV)				
R-31 'A' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-32 'A' Steamline High Range	1.77E+00 R/hr	—	—	—
R-33 'B' Steamline Lo Range	1.77E+03 mR/hr	1.77E+02 mR/hr	—	—
R-34 'B' Steamline High Range	1.77E+00 R/hr	—	—	—
Main Steam Line (SG Safety)				
R-31 'A' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-32 'A' Steamline High Range	—	—	—	—
R-33 'B' Steamline Lo Range	8.30E+02 mR/hr	8.30E+01 mR/hr	—	—
R-34 'B' Steamline High Range	—	—	—	—
Liquid Radwaste				
R-16 Containment Fcu SW Return	N/A	N/A	3.38E+07 cpm	3.38E+05 cpm
R-19 S/G Blowdown Liquid	N/A	N/A	2.58E+08 cpm	2.58E+06 cpm
R-20 Aux Bldg SW Return	N/A	N/A	1.03E+07 cpm	1.03E+05 cpm

KNPP Basis Reference(s):

1. USAR Section 11.2.3 Radiation Monitoring System, Rev. 18
2. C11620, Evaluation of Radiological Effluent Monitor Response Action Levels, Rev. 0
3. EPIP-RET-02B Gaseous Effluent Release Path, Radioactivity, and Release Rate Determination, Rev. T
4. ODCM Section 2.0 Gaseous Effluents, Rev. 8

Table C-0
Recognition Category C
Cold Shutdown/Refueling System Malfunction
INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 RCS Leakage. *Op. Mode: Cold Shutdown*	**CA1** Loss of RCS Inventory. *Op. Modes: Cold Shutdown*	**CS1** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability. *Op. Modes: Cold Shutdown*	**CG1** Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel. Op. Modes: Cold Shutdown, Refueling
CU2 UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel *Op. Mode: Refueling*	**CA2** Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Refueling*	**CS2** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. Op. Modes: Refueling	
CU3 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*	**CA3** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU4 UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. *OP. Modes: Cold Shutdown, Refueling*	**CA4** Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Cold Shutdown, Refueling*		
CU5 Fuel Clad Degradation. *Op. Modes: Cold Shutdown, Refueling*			
CU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities. *Op. Modes: Cold Shutdown, Refueling*			
CU7 UNPLANNED Loss of Required DC Power for Greater than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*			

CU8 Inadvertent Criticality.
Op Modes:, Cold Shutdown, Refueling

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CU1

Initiating Condition -- UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CU1.1 or CU1.2)

CU1.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

CU1.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it is considered to be a potential degradation of the level of safety of the plant. Positive indications in the Control Room of Reactor Coolant System (RCS) leakage to the containment are provided by equipment that monitors:

- Charging/Letdown flow mismatch
- Containment air activity
- Containment humidity
- Containment Sump A In-leakage

[Ref. 1, 2]

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation (e.g., Pressurizer level, RCS loop level instrumentation, etc.) or reduced inventory instrumentation such as tygon level indication. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

KNPP Basis Reference(s):

1. Technical Specifications LCO 3.1.d, Amendment No. 165
2. SP-36-82 Reactor Coolant System Leak Rate Check, Rev. AE

SYSTEM MALFUNCTION

CU2

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CU2.1 or CU2.2)

CU2.1. UNPLANNED RCS level lowering below the Reactor Vessel flange (21.5%) for GREATER THAN OR EQUAL TO 15 minutes

CU2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A, Containment Sump C or Liquid Waste Disposal System level rise

AND

Reactor Vessel level cannot be monitored

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange warrants declaration of an Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists. Continued loss of RCS Inventory will result in escalation to the Alert level via either IC CA2 (Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

In the refueling shutdown mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump A, Containment Sump C and Liquid Waste Disposal System level changes [Ref. 1, 2]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. When CONTAINMENT SUMP A LEVEL HIGH is received, the corresponding leakrate within containment is calculated from sump pump run history. Escalation to Alert would be via either CA2 or RCS heatup via CA4.

CU2.1 involves a decrease in RCS level below the top of the Reactor Vessel flange that continues for 15 minutes due to an UNPLANNED event. The level at the Reactor Vessel flange is monitored by:

- Wide Range Refueling Water Level (L9053A for channel A and L9054A for channel B) indication: 21.5%
- RVLIS: 52.8%
- Sightglass/Tygon: 340 in. WC

[Ref. 3]

This EAL is not applicable to decreases in flooded reactor cavity level (covered by RU2.1) until such time as the level decreases to the level of the vessel flange. If Reactor Vessel level continues to decrease and reaches the Bottom ID of the RCS Loop (Refueling Level, 0% RVLIS, 252 in. sightglass), then escalation to CA2 would be appropriate. Note that the Bottom ID of the RCS Loop Setpoint corresponds to the bottom of the Reactor Vessel loop penetration (not the low point of the loop).

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. N-RHR-34C RHR Operation at a Reduced Inventory Condition, Rev. N
3. SP 36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
4. SP 36-082 Reactor Coolant System Leak Rate Check, Rev. AE
5. ES-1.3 Transfer to Containment Sump Recirculation, Rev. W
6. A-MDS-30 Miscellaneous Drains and Sumps (MDS) Abnormal Operation, Rev. N

SYSTEM MALFUNCTION

CU3

Initiating Condition -- **UNUSUAL EVENT**

Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU3.1. Loss of all offsite power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes.

AND

At least one emergency diesel generator is supplying power to Bus 5 or Bus 6.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system is divided into six busses, two of which are Engineered Safety Function (ESF) Busses 5 and 6. The ESF busses supply power to Safety Injection (SI) pumps, Residual Heat Removal (RHR) pumps, containment heat removal equipment, etc.

Offsite power is available from the 345 kVAC and 138 kVAC systems. The 345 kVAC system is connected to the North Appleton line, the Point Beach line, the main transformers, and transformer T-10. The 345 kVAC is the normal supply to the 13.8 kVAC system through transformer T-10, which feeds the Tertiary Auxiliary Transformer (TAT). The TAT normally provides power to ESF bus 5. The TAT is not considered available to power both ESF busses in an emergency situation due to its size. As a contingency, however, it is acceptable to use the TAT to power both ESF busses when guidance for sequencing and monitoring TAT loads is available in the Control Room. The Reserve Auxiliary Transformer (RAT) and Main Auxiliary Transformer (MAT) provide backup sources to bus 5, in that order.

The 138 kVAC system is connected to the Shoto/Mishicot line, the East Krok line and transformer T-10. The 138 kVAC system is the normal supply to the Reserve Auxiliary Transformer (RAT) via the East and West substation busses. (When the 345 kVAC system is unavailable, the 138 kVAC system can supply power to transformer T-10 and the TAT.) The RAT normally provides power to ESF bus 6. The TAT and MAT provide backup sources to bus 6 in that order.

When the main turbine generator is on line, generator output supplies power to the Main Auxiliary Transformer (MAT) and the 4160 VAC busses. When the main turbine generator is off line, the 345 kVAC system can be aligned to backfeed the MAT. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval associated with this EAL. If shutdown plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source.

Following a loss of power, ECA 0.0 provides guidance to restore power to ESF busses. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 kVAC system supplying power to transformer T-10 and the TAT
- 138 kVAC system supplying power to transformer T-10 and the TAT
- 138 kVAC system supplying power to the RAT
- 345 kVAC system supplying power to the MAT on backfeed through the main transformers when the main turbine generator is off line

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. GNP-08.04.01 Shutdown Safety Assessment, Rev. K

SYSTEM MALFUNCTION

CU4

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU4.1 or CU4.2)

CU4.1. An UNPLANNED event results in RCS temperature GREATER THAN 200°F

CU4.2. Loss of all RCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the alert level via CA4 or CA1 will occur if required.

During refueling the level in the Reactor Vessel will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation to the Alert level via CA4.

Unlike the cold shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, CU4.2 would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200°F) [Ref. 1].

Reactor Vessel water level is normally monitored using the following instruments:

- 21158 Refueling Water Level Narrow Range (L9055A)
- 21159 Refueling Water Level B Wide Range (L9054A)
- 24068 Refueling Water Level A Wide Range (L9053A)
- LI-41337 Reactor Cavity Lvl
- Local Rx Vessel Level Sightglass/Tygon (252 in. to 645 in.)
- RVLIS 41622 Train A
- RVLIS 41623 Train B

Refueling Water Level B Wide Range instrument is calibrated to provide indication from the top of active fuel (0% or 200 in. WC) to the refueling floor (68.5% or 645 in. WC). The Reactor Vessel Level Indicating System (RVLIS) is part of the Post Accident Monitoring Instrumentation. RVLIS is provided for verification and long term surveillance of core cooling and indicates from the bottom of the RCS hot leg penetration (0% or 252 in. WC) to above the high point of the Reactor Vessel head (100% or 419 in.). Procedures N-RC-36E, Draining the Reactor Coolant System, and N-RHR-34C, RHR Operation at a Reduced Inventory Condition, provide a cross-reference table of indicated water levels and sightglass readings.

[Ref. 2, 3, 4]

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). N-0-01, Plant Startup from Cold Shutdown Condition to Hot Shutdown Condition, specifies the use of the highest of the wide range, RHR inlet, or Core Exit Thermocouples to monitor RCS temperature in the Cold Shutdown or Refueling Mode.

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

KNPP Basis Reference(s):

1. Technical Specifications, Modes Definition for Cold Shutdown, Amendment No. 172
2. N-RC-36E Draining the Reactor Coolant System, Rev. AE
3. N-RHR-34C RHR Operation at a Reduced Inventory Condition, Rev. N
4. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
5. A-RHR-34 Abnormal Residual Heat Removal System Operation, Rev. Y
6. N-0-01 Plant Startup from Cold Shutdown Condition to Hot Shutdown Condition, Rev. Z
7. USAR Figure 7.7-1, Plan-Vertical Panels and Consoles, Rev. 18

SYSTEM MALFUNCTION

CU5

Initiating Condition -- UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU5.1or CU5.2)

CU5.1. RCS Letdown Line (R-9) radiation monitor GREATER THAN 2000 mR/hr indicating fuel clad degradation.

CU5.2. Coolant sample activity GREATER THAN ANY of the following indicating fuel clad degradation:

- 1.0 μCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval
- 60 μCi/gram dose equivalent Iodine-131.
- 91/Ē μCi/cc gross radioactivity

Basis:

This IC is included as a UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. CU5.1 addresses RCS Letdown Line (R-9) radiation monitor readings that provide indication of fuel clad integrity [Ref. 4 & 5]. CU5.2 addresses coolant samples exceeding coolant technical specifications for iodine spike [Ref. 1].

2000 mR/hr was calculated using the following:

0.01% fuel cladding defect equals 7.2E+1 mR/hr increase on R-9 [Ref. 4]
0.2745% fuel cladding defect equals 1.0 μCi/gram dose equivalent Iodine-131 [Ref. 5].

Therefore 1976.4 mR/hr increase on R-9 is equal to 1.0 μCi/gram dose equivalent Iodine-131

R-9 background is equivalent to 56 mR/hr [Ref. 4], which is added to the calculated dose rate above.
With the addition of background R-9 will read 2032.4 mR/hr (rounded to 2000 mR/hr) equal to 1.0 μCi/gram dose equivalent Iodine-131.

Although the Technical Specification is applicable when average reactor coolant temperature is GREATER THAN 500°F, it is appropriate that this EAL be applicable in cold shutdown and refueling modes, as it indicates a potential degradation in the level of safety of the plant.

KNPP Basis Reference(s):

1. Technical Specifications LCO 3.1.c.1.A, Amendment No. 167
2. E-2021 Integrated Logic Diagram Radiation Monitoring, Rev. X
3. A-RC-36A High Reactor Coolant Activity, Rev. J
4. USAR Section 9, Rev. 16
5. CN-CRA-99-28 Rev. 1

CU6

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU6.1 or CU6.2)

CU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Intraplant Paging (Gai-tronics) • Sound powered phones • PBX telephone system • Personal communications system (PCS phones) • Portable radio communications system

CU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• PBX telephone system • NRC FTS System (including ENS and HPN) • Dial select phones

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

Table C-1 onsite communications loss encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system and radios / walkie talkies). Due to its limited capability, the emergency gai-tronics is not listed in Table C-1.

Table C-2 offsite communications loss encompasses the loss of all means of communications with offsite authorities. This includes the NRC FTS System (including Emergency Notification System - ENS and Health Physics Network – HPN), commercial telephone lines, telecopy transmissions, and dedicated phone systems.

KNPP Basis Reference(s):

1. N-COM-44-CL Communications Systems CL, Rev. K

SYSTEM MALFUNCTION

CU7

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of Required DC Power for GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU7.1 UNPLANNED Loss of Vital DC power based on LESS THAN 105 VDC on Train A AND Train B Safeguards DC Distribution System.

AND

Failure to restore power to at least one required Train of the Safeguards DC Distribution System within 15 minutes from the time of loss.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

UNPLANNED is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per EAL CA4 "Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel."

LESS THAN 105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment [Ref. 1, 2]. The loss of a safeguards DC train consists of a combination of loss of power to specified DC distribution panels. These panels include: BRA (BRB)-102, and BRA (BRB)-104. In all cases, BRA (BRB)-102 panel indicating less than 105 VDC constitutes a loss of the associated DC distribution train. However, a loss of power to the BRA (BRB) -104 panel, which does not have voltage indication, also constitutes a loss of the associated DC distribution train.

125 VDC safeguard main distribution cabinets (BRA-102 and BRB-102) supply two safeguard sub-distribution cabinets (BRA-104 and BRB-104) and provide for connection of safeguard batteries (BRA-101 and BRB-101) to their associated battery chargers (BRA-108 and BRB-108). The combination of low voltages on the specified distribution cabinets results in a total loss of vital 125 VDC power. The 125 VDC safeguards system powers circuit breaker control, Control Room alarms, Control Room controls, diesel generator controls, and the Reactor Protection System. It is also the standby power source to the AC inverters. BRA-102 and BRB-102 voltage is displayed on Control Room indicators 4494001 and 4494002, respectively. Undervoltage is alarmed on Control

Room Sequence of Event Recorder (SER) points 490011196 and 490011200 and annunciators 447101A and 47101B, respectively.

Each of the 125 VDC batteries has been sized to carry the expected shutdown loads following a reactor trip and a loss of all AC power for a period of eight hours without battery terminal voltage falling below 105 VDC. This voltage value therefore incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads. The nominal battery cell voltage is 2.20 VDC. Low battery terminal voltage activates Control Room SER point 49001832 and annunciator 47105A. The batteries are located in Battery Rooms A and B on the Turbine Building Mezzanine Floor (606 ft el.).

KNPP Basis Reference(s):

1. USAR 8.2.2, Rev. 18
2. USAR 8.2.3, Rev. 18
3. Technical Specifications 3.7, Amendment No. 122
4. A-EDC-38, Abnormal DC Supply and Distribution System, Rev. Z
5. Plant Drawing 237127A-E233, Rev. AQ

SYSTEM MALFUNCTION

CU8

Initiating Condition -- UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability:

Cold Shutdown
Refueling

Emergency Action Level:

CU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion IC SU8.

This condition can be identified using startup rate meters. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate meters (NI-31D/32D - Source Range Startup Rate).

Escalation would be by Emergency Director Judgment.

KNPP Basis Reference(s):

1. N-0-02 Plant Startup from Hot Shutdown to 35% Power, Rev. AN

SYSTEM MALFUNCTION

CA1

Initiating Condition -- ALERT

Loss of RCS Inventory.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CA1.1 or CA1.2)

CA1.1. Loss of RCS inventory as indicated by one or more of the following:
- Wide Range Refueling Water Level LESS THAN 8%
- RVLIS at 0%
- Sightglass water level LESS THAN 252 in

CA1.2. Loss of RCS inventory as indicated by unexplained level rise in any of the following:
- Containment Sump A
- Containment Sump C
- Liquid Waste Disposal System

AND

RCS level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. The 8% Refueling Level (0% RVLIS, 252 in. sightglass) threshold corresponds to the bottom inside diameter of the RCS hot leg [Ref. 2]. This condition will result in a minimum classification of Alert. The Bottom ID of the RCS hot leg Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The Bottom ID of the RCS hot leg Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

The elevation of the bottom of the RCS hot leg can be monitored by:
- Wide Range Refueling Water Level (L9053A for channel A and L9054A for channel B) indication: 7.95% rounded to 8% for readability
- RVLIS: 0%
- Sightglass/Tygon: 252 in. WC

Reactor Vessel water level is normally monitored using the following instruments:

- 21158 Refueling Water Level Narrow Range (L9055A)
- 21159 Refueling Water Level B Wide Range (L9054A)
- 24068 Refueling Water Level A Wide Range (L9053A)
- LI-41337 Reactor Cavity Lvl
- Local Rx Vessel Level Sightglass/Tygon (252 in. to 645 in.)
- RVLIS 41622 Train A
- RVLIS 41623 [Ref 2]

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the cold shutdown mode, if the RCS is pressurized, then the refueling water level indication (including sightglass / tygon) will not be in service. In this case, RVLIS will serve as the means for declaration of this EAL. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump A, Containment Sump C and Liquid Waste Disposal System level changes [Ref. 1, 5]. Each time annunciator CONTAINMENT SUMP A LEVEL HIGH is received, the corresponding leakrate within containment is calculated from sump pump run history. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage [Ref. 1, 2,]. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

If Reactor Vessel level continues to decrease then escalation to Site Area Emergency will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
3. SP-36-082 Reactor Coolant System Leak Rate Check, Rev. AE
4. ES-1.3 Transfer to Containment Sump Recirculation, Rev. W
5. N-RHR-34C RHR Operation at a Reduced Inventory Condition, Rev. N

SYSTEM MALFUNCTION

CA2

Initiating Condition -- ALERT

Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CA2.1 or CA2.2)

CA2.1. Loss of RCS inventory as indicated by Wide Range Refueling Water Level LESS THAN 8% (0% RVLIS, 252 in. sightglass)

CA2.2. Loss of Reactor Vessel inventory as indicated by unexplained level rise in any of the following:

- Containment Sump A
- Containment Sump C
- Liquid Waste Disposal System

AND

Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes

Basis:

These example EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. The 8% Refueling Level (0% RVLIS, 252 in. sightglass) threshold corresponds to the bottom inside diameter of the RCS loop [Ref. 2]. This condition will result in a minimum classification of Alert. The Bottom ID of the RCS hot leg Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems may occur. The Bottom ID of the RCS hot leg Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

The elevation of the bottom of the RCS hot leg can be monitored by:

- Wide Range Refueling Water Level (L9053A for channel A and L9054A for channel B) indication: 7.95% rounded to 8% for readability
- RVLIS: 0%
- Sightglass/Tygon: 252 in. WC

Reactor Vessel water level is normally monitored using the following instruments:

- 21158 Refueling Water Level Narrow Range (L9055A)
- 21159 Refueling Water Level B Wide Range (L9054A)
- 24068 Refueling Water Level A Wide Range (L9053A)
- LI-41337 Reactor Cavity Lvl
- Local Rx Vessel Level Sightglass/Tygon (252 in. to 645 in.)
- RVLIS 41622 Train A
- RVLIS 41623 Train B

[Ref. 2]

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump A, Containment Sump C and Liquid Waste Disposal System level changes [Ref. 1, 5]. Each time annunciator CONTAINMENT SUMP A LEVEL HIGH is received, the corresponding leakrate within containment is calculated from sump pump run history. [Ref. 1, 2, 3] Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

If Reactor Vessel level continues to decrease then escalation to Site Area Emergency will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
3. SP-36-082 Reactor Coolant System Leak Rate Check, Rev. AE
4. ES-1.3 Transfer to Containment Sump Recirculation, Rev. W
5. N- RHR-34C RHR Operation at a Reduced Inventory Condition, Rev. N

SYSTEM MALFUNCTION

CA3

Initiating Condition -- ALERT

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Cold Shutdown
Refueling
Defueled

Emergency Action Level:

CA3.1. Loss of ALL power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Service Water System. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is by Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This EAL is indicated by the loss of all offsite and onsite AC power to the 4160 VAC ESF busses. Offsite power sources include the 345 kVAC system or 138 kVAC system to the Tertiary Auxiliary Transformer (TAT), the 345 kVAC system or 138 kVAC system to the Reserve Auxiliary Transformer (RAT), and the 345 kVAC system to the Main Auxiliary Transformer (MAT) on backfeed through the main transformers. Note that the time required to effect a backfeed to the MAT is likely longer than the fifteen-minute interval. If shutdown plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source. Onsite power sources consist of Diesel Generator A to Bus 5 and Diesel Generator B to Bus 6. [Ref. 1, 2, 3, 4, 5].

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. USAR Section 8.2.4, Rev. 18
5. GNP-08.04.01 Shutdown Safety Assessment, Rev. K

SYSTEM MALFUNCTION

CA4

Initiating Condition -- ALERT

Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (EAL CA4.1 or CA4.2 or CA4.3)

CA4.1. With CONTAINMENT CLOSURE NOT established
AND
RCS integrity NOT established,
An UNPLANNED event results in RCS temperature GREATER THAN 200°F.

CA4.2. With CONTAINMENT CLOSURE established
AND
RCS integrity NOT established OR Wide Range Refueling Water Level LESS THAN 17.0%,
An UNPLANNED event results in RCS temperature GREATER THAN 200°F for GREATER THAN 20 minutes*.

***NOTE**: If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.

CA4.3. An UNPLANNED event results in RCS temperature GREATER THAN 200°F for GREATER THAN 60 minutes*.
OR
Results in an RCS pressure increase of GREATER THAN 10 psig.

***NOTE** : If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.

Basis:

CA4.1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., reactor head on with studs tensioned, S/G and PRZR man-ways installed, PRZR safety valves installed, no freeze seals or nozzle dams). No delay time is allowed for CA4.1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

CA4.2 addresses the complete loss of functions required for core cooling for GREATER THAN 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced (e.g., mid loop operation). As in CA4.1, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., reactor head on with studs tensioned, S/G and PRZR man-ways installed, PRZR safety valves installed, no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. The Note for CA4.2 indicates that CA4.2 is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20 minute time frame. Wide Range Refueling Water Level is measured by L9053A for channel A and L9054A for channel B.

CA4.3 addresses complete loss of functions required for core cooling for GREATER THAN 60 minutes during refueling and cold shutdown modes when RCS integrity is established. As in CA4.1 and CA4.2, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., reactor head on with studs tensioned, S/G and PRZR man-ways installed, PRZR safety valves installed, no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. RCS Pressure Narrow Range instrument PI-420 and PPCS/SPDS point P0420A are capable of measuring pressure to less than 10 psig. [Ref. 3, 7]. The Note for CA4.3 indicates that CA4.3 is not applicable if actions are successful in restoring the RHR system to operation and RCS temperature is being reduced within the 60 minute time frame assuming that the RCS pressure rise has remained less than the site specific pressure value.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). N-0-01, Plant Startup from Cold Shutdown Condition to Hot Shutdown Condition, specifies the use of the highest of the wide range, RHR inlet, or Core Exit Thermocouples to monitor RCS temperature in the Cold Shutdown or Refueling Mode.

[Ref. 2, 3,]

Escalation to Site Area Emergency would be via CS1 or CS2 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200 degrees F when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

KNPP Basis Reference(s):

1. Technical Specifications, Modes Definition for Cold Shutdown, Amendment No. 172
2. A-RHR-34 Abnormal Residual Heat Removal System Operation, Rev. Y
3. N-0-01 Plant Startup from Cold Shutdown Condition to Hot Shutdown Condition, Rev. Z
4. N-CCI-56A Open Containment Boundary Tracking, Rev. F
5. GNP-08.04.01 Shutdown Safety Assessment, Rev. K
6. N-O-05 Plant Cooldown from Hot Shutdown to Cold Shutdown Condition 1, Rev. AY
7. N-RC-36E Draining the Reactor Coolant System, Rev. AE

SYSTEM MALFUNCTION

CS1

Initiating Condition -- **SITE AREA EMERGENCY**

Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CS1.1 or CS1.2)

CS1.1. With CONTAINMENT CLOSURE NOT established:

a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level LESS THAN 7%

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained level rise in any of the following:

- Containment Sump A
- Containment Sump C
- Liquid Waste Disposal System

CS1.2. With CONTAINMENT CLOSURE established:

a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level EQUAL TO 0%

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either:

- Unexplained Containment Sump A, Containment Sump C, OR Liquid Waste Disposal System level rise
- Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a Reactor Vessel breach, pressure boundary leakage, or continued boiling in the Reactor Vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

In the cold shutdown mode, normal RCS level and reactor vessel level indication systems (RVLIS) will normally be available. If the RCS is pressurized, then the Wide Range Refueling Water Level indication will not be in service. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. <u>RVLIS indication is considered lost if leakage reduces RCS level below its indicating range</u>. Each time annunciator CONTAINMENT SUMP A LEVEL HIGH is received, the corresponding leakrate within containment is calculated from sump pump run history. [Ref. 1, 5] Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

When Reactor Vessel water level drops to 616 ft 4 in. el., the level associated without CONTAINMENT CLOSURE established, level is six inches below the bottom of the RCS hot leg vessel penetration. This level can be monitored by Wide Range Refueling Water Level (L9053A for channel A and L9054A for channel B) indication at 7.1% (rounded to 7% for readability). The following indications are off scale low and as such are not available:

- RVLIS: <0%
- Sightglass/Tygon level equal to 246 in. WC.

When Reactor Vessel water level drops to 612 ft 4 in. el., the level associated with CONTAINMENT CLOSURE established, core uncovery is about to occur. Wide Range Refueling Water Level indication of 0% is approximately the top of active fuel.

[Ref. 1, 2]

The 30-minute duration allowed when CONTAINMENT CLOSURE is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. Effluent release is not expected with closure established.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC AG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
3. N-CCI-56A Open Containment Boundary Tracking, Rev. F
4. GNP-08.04.01 Shutdown Safety Assessment, Rev. K
5. SP-36-082 Reactor Coolant System Leak Rate Check, Rev. AE
6. ES-1.3 Transfer to Containment Sump Recirculation, Rev. W
7. N-0-02 Plant Startup from Hot Shutdown to 35% Power, Rev. AN
8. A-MDS-30 Miscellaneous Drains and Sumps (MDS) Abnormal Operation, Rev. N

SYSTEM MALFUNCTION

CS2

Initiating Condition -- SITE AREA EMERGENCY

Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CS2.1 or CS2.2)

CS2.1. With CONTAINMENT CLOSURE NOT established:

a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level LESS THAN 7%

OR

b. Reactor Vessel level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following:
- Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr
- Erratic Source Range Monitor Indication

CS2.2. With CONTAINMENT CLOSURE established

a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level EQUAL TO 0%

OR

b. Reactor Vessel level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following:
- Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr
- Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to an Reactor Vessel breach or continued boiling in the Reactor Vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the

Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

When Reactor Vessel water level drops to 616 ft 4 in. el., the level associated without CONTAINMENT CLOSURE established, level is six inches below the bottom of the RCS hot leg vessel penetration. This level can be monitored by Wide Range Refueling Water Level (L9053A for channel A and L9054A for channel B) indication at 7.1% (rounded to 7% for readability). The following indications are off scale low and as such are not available:

- RVLIS: <0%
- Sightglass/Tygon level equal to 246 in. WC.

When Reactor Vessel water level drops to 612 ft 4 in. el., the level associated with CONTAINMENT CLOSURE established, core uncovery is about to occur. Wide Range Refueling Water Level indication of 0% is approximately the top of active fuel.

[Ref. 1, 2]

In Refuel mode at the levels of interest, RVLIS is unavailable but alternate means of level indication (refueling level) are installed to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in an unplanned alarm on the Containment Area Monitor (R-2). R-2 is used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale. Therefore the alarm setpoint of R-2 was selected to indicate a rise in containment radiation resulting from the conditions of this EAL [Ref. 8].

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (SRM) N-31B and N-32B can be used as a tool for making such determinations. SRM count rate can also be indicated in the Control Room by the audible SRM count rate monitor.

Effluent release is not expected with CONTAINMENT CLOSURE established.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC AG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
3. N-CCI-56A Open Containment Boundary Tracking, Rev. F
4. GNP-08.04.01 Shutdown Safety Assessment, Rev. K
5. SP-36-082 Reactor Coolant System Leak Rate Check, Rev. AE
6. ES-1.3 Transfer to Containment Sump Recirculation, Rev. W
7. N-0-02 Plant Startup from Hot Shutdown to 35% Power, Rev. AN
8. C11622, Determination of R-2 Reading with Loss of Inventory, Rev. 0

SYSTEM MALFUNCTION

CG1

Initiating Condition -- GENERAL EMERGENCY

Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CG1.1. Loss of Reactor Vessel inventory as indicated by unexplained level rise in Containment Sump A, Containment Sump C OR Liquid Waste Disposal System

AND

Reactor Vessel Level (a or b):

a. EQUAL TO 0% Wide Range Refueling Water Level for GREATER THAN 30 minutes

OR

b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by one or more of the following:
   - Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr
   - Erratic Source Range Monitor Indication

AND

Indication of CONTAINMENT challenged as indicated by one or more of the following:

- GREATER THAN OR EQUAL TO 6% hydrogen in containment
- CONTAINMENT CLOSURE NOT established
- CONTAINMENT pressure above:
  - 46 psig IF Containment Integrity or Reduced Inventory Containment Integrity is established

    OR

  - 46 psig IF Refueling Containment Integrity is established with no loop seal penetrations installed at Penetration 42N or 43N.

    OR

  - 0.6 psig IF Refueling Containment Integrity is established with loop seal penetration installed at either Penetration 42N or 43N.

Basis:

In the cold shutdown mode, normal RCS level and Reactor Vessel level instrumentation systems will normally be available. If the RCS is pressurized, then the Wide Range Refueling Water Level indication will not be in service. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. RVLIS indication is considered lost if leakage reduces RCS level below its indicating range.

In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Wide Range Refueling Water Level is measured by L9053A for channel A and L9054A for channel B.

Containment Sump A, Containment Sump C or Liquid Waste Disposal System level changes may be indicative of a loss of RCS inventory. Containment Sump A receives all liquid waste from floor and equipment drains inside containment including that from Containment Sump C. Each time annunciator CONTAINMENT SUMP A LEVEL HIGH is received, the corresponding leakrate within containment is calculated from sump pump run history. [Ref. 1, 8] Sump level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. [Ref. 19]

This EAL represents the inability to restore and maintain Reactor Vessel level to above the top of active fuel.. Fuel damage is probable if Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the Reactor Vessel level. When Reactor Vessel water level drops to 612 ft 4 in. el., core uncovery is about to occur. Wide Range Refueling Water Level indication of 0% is approximately the top of active fuel. [Ref. 2]

If all means of level monitoring are not available, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in an unplanned alarm on the Containment Area Monitor (R-2). R-2 is used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale. Therefore the alarm setpoint of R-2 was selected to indicate a rise in containment radiation resulting from the conditions of this EAL [Ref. 3].

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (N-31 and N-32) can be used as a tool for making such determinations.

The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE.

CONTAINMENT CLOSURE is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE should not be confused with Refueling Containment Integrity as described in N-FH-53-CLA or CLB [Ref 6, 7]. Reduced Inventory Containment Integrity is described in N-CCI-56A–CLA or CLB [Ref 9, 10]. Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, escalation to GE would not occur.

The pressure at which CONTAINMENT is considered challenged is based on the condition of the CONTAINMENT. When the Unit is in the cold shutdown mode and the CONTAINMENT is fully intact, Containment is considered challenged at CONTAINMENT design pressure of 46 psig. Refueling CONTAINMENT Integrity establishes normal CONTAINMENT isolation except that penetrations 42N and 43N may have loop seal penetrations installed. When a loop seal penetration is installed, CONTAINMENT is considered challenged when CONTAINMENT pressure exceeds 0.6 psig. If fiber optic penetration is installed with no loop seal penetration installed, CONTAINMENT is considered challenged at full CONTAINMENT design pressure of 46 psig. [Ref. 20 and 21].

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. When hydrogen and oxygen concentrations reach or exceed the deflagration limits (equal to or greater than 6% hydrogen), loss of the containment barrier is possible [Ref. 13, 15, 16]. Containment hydrogen concentration can be obtained from PPCS/SPDS point X8001A and X8002A, or Control Room meters 41615 and 41616.

KNPP Basis Reference(s):

1. N-RC-36E Draining the Reactor Coolant System, Rev. AE
2. SP-36-196A Refueling Water Level Indication System Transmitter Calibration, Rev. G
3. C11622, Determination of R-2 Reading with Loss of Inventory, Rev. 0
4. N-0-02 Plant Startup from Hot Shutdown to 35% Power, Rev. AN
5. N-RHR-34C-CL Requirements for Entering Reduced Inventory Checklist, Rev. H
6. N-FH-53-CLA Refueling Containment Integrity CL, S/G Secondary Side Intact, Rev. G
7. N-FH-53-CLB Refueling Containment Integrity CL, S/G Secondary Side Open, Rev. G
8. N-CCI-56A Open Containment Boundary Tracking, Rev. F
9. N-CCI-56A-CLA Reduced Inventory Cntmt Integrity Checklist – SG Secondary Side Intact, Rev. K
10. N-CCI-56A-CLB Reduced Inventory Cntmt Integrity Checklist – SG Secondary Side Open, Rev. J
11. GNP-08.04.01 Shutdown Safety Assessment, Rev. K
12. EPIP-TSC-07 RV Head Venting time Calculation, Rev. J
13. M-403 Reactor Building Vent System Post-LOCA Hydrogen Control, Rev. Y
14. Technical Specifications Table 3.5.6, Amendment No. 105

15. FR-C.1 Response to Inadequate Core Cooling, Rev. N
16. N-RBV-18C POST-LOCA Hydrogen Control, Rev. K
17. F-0.5 Containment, Rev. F
18. USAR Section 5.2.1, Rev. 16
19. SP-36-082 Reactor Coolant System Leak Rate Check, Rev. AE
20. DCR1811, Refueling Containment Loop Seal
21. DCR 2167, New Refueling Containment Cableway

Table F-0

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

GENERAL EMERGENCY		SITE AREA EMERGENCY		ALERT	
FG1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier *Op. Modes: Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown*	FS1	Loss or Potential Loss of ANY Two Barriers *Op. Modes: Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown*	FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS *Op. Modes: Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown*

NOTES

1. The logic used for these initiating conditions reflects the following considerations:

   - The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table F-1 states that imminent (i.e., within 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

TABLE F-1

KNPP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Core-Cooling Red	Core Cooling-Orange OR Heat Sink-Red	Not Applicable	RCS Integrity-Red OR Heat Sink-Red	Not Applicable	Containment-Red
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
Coolant Activity GREATER THAN 300 μCi/gm I-131 equivalent	Not Applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling • LESS THAN 20°F if the reactor is critical • LESS THAN 30°F if the reactor is sub-critical	Unisolable leak GREATER THAN 60 gpm the capacity of one charging pump in the normal charging mode	Rapid unexplained decrease following initial rise OR Containment pressure or sump level response not consistent with LOCA conditions	46 PSIG and rising OR Hydrogen concentration GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 23 psig with LESS THAN one full train of depressurization equipment operating
OR				**OR**	

TABLE F-1

KNPP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
<u>3. Core Exit Thermocouple Readings</u>				<u>3. Core Exit Thermocouple Reading</u>	
GREATER THAN OR EQUAL TO 1200°F	GREATER THAN OR EQUAL TO 700°F			Not applicable	Core exit thermocouples GREATER THAN OR EQUAL TO 1200°F and restoration procedures not effective within 15 minutes; OR Core exit thermocouples GREATER THAN OR EQUAL TO 700°F with RCPs NOT running <u>AND</u> restoration procedures not effective within 15 minutes OR RVLIS void fraction rising with at least one RCP running and RCS subcooling LESS THAN 30°F [65°F] and restoration procedures not effective within 15 minutes

TABLE F-1
KNPP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with Primary-to-Secondary Leakage**	
Not Applicable	RVLIS void fraction rising AND At least one RCP running AND RCS subcooling LESS THAN 30°F [65°F]	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate GREATER THAN 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				OR	
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment isolation valve(s) not closed AND Downstream pathway to the environment exists, after containment isolation	Not Applicable
OR		**OR**		**OR**	

TABLE F-1

KNPP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory In Containment**	
Containment rad monitor (R-40/41) reading GREATER THAN 1000 R/hr	Not Applicable	Containment rad monitor (R-40/41) reading GREATER THAN 30 R/hr	Not Applicable	Not Applicable	Containment rad monitor (R-40/41) reading GREATER THAN 4000 R/hr
OR		**OR**		**OR**	
6. Emergency Director Judgment		**5. Emergency Director Judgment**		**7. Emergency Director Judgment**	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

Basis Information For Table F-1
KNPP Emergency Action Level
Fission Product Barrier Reference Table

FUEL CLAD BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6)

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. Core Cooling-ORANGE path is entered if:

- RCS subcooling based on CETs is equal to or less than 30°F [65°F] and
- No RCPs are running, and
- Core Exit Thermocouples (CETs) are reading between 700°F and 1200°F

OR

- RCS subcooling based on CETs is equal to or less than 30°F [65°F], and
- At least one RCP is running, and
- RVLIS Void Fraction is Rising

[Ref. 1, 2]

Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items (Core Cooling – ORANGE or Heat Sink – RED) indicate potential loss of the Fuel Clad Barrier. Heat Sink-Red path is entered if narrow range level in both S/Gs is less than 4% [15%] and total feedwater flow to S/Gs is less than 200 gpm.

[Ref. 4, 5]

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier. Core Cooling-RED path is entered if Core Exit Thermocouples (CETs) are equal to or greater than 1200°F.

CSFST setpoints enclosed in brackets (e.g., [65°F], etc.) are used under adverse containment conditions. Adverse containment condition thresholds apply when containment pressure is greater than 4 psig or containment radiation exceeds 10E+05 R/hr.

2. **Primary Coolant Activity Level**

This value is 300 μCi/gm I_{131} equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no equivalent "Potential Loss" EAL for this item.

3. **Core Exit Thermocouple Readings**

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked).

The "Loss" EAL 1200 degrees F reading should correspond to significant superheating of the coolant. This value corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier EAL #1 which is 1200 degrees F. [Ref. 1, 6]

The "Potential Loss" EAL 700 degrees F reading should correspond to loss of subcooling. This value corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier EAL #1 which is 700 degrees F. [Ref.1, 2]

4. **Reactor Vessel Water Level**

There is no "Loss" EAL corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" EALs.

The "Potential Loss" EAL is indicative of core uncovery but, when the reactor is at pressure and temperature, RVLIS should not be used for a quantitative value (i.e., top of active fuel). Functional restoration procedure FR-C.2 specifies monitoring of RVLIS void fraction trend and RCS subcooling instead of the water level corresponding to the top of active fuel.

The "Potential Loss" EAL is therefore defined by the Core Cooling - ORANGE path. The trend in RVLIS RCS void fraction is used to check the effectiveness of safety injection in restoring RCS inventory. If void fraction percent is decreasing and RCS subcooling based on Core Exit Thermocouples (CETs) is greater than 30°F [65°F], safety injection has been successful in restoring RCS inventory and core cooling. In the event that RCS void fraction is increasing and subcooling requirements are not met, core cooling continues to be degraded and some fuel cladding damage may occur. Setpoints enclosed in brackets are used under adverse containment conditions. Adverse containment condition thresholds apply when containment pressure is greater than 4 psig or containment radiation exceeds 10E+05 R/hr. [Ref. 7]

5. **Containment Radiation Monitoring**

The 1000 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere. [Ref. 8, 9, 10] Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors R-40 and R-41.

There is no "Potential Loss" EAL associated with this item.

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

RCS BARRIER EALs: (1 or 2 or 3 or 4 or 5)

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including isolation valves.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

RCS Integrity-Red path is entered if:

- Temperature decrease in both RCS cold legs is equal to or greater than 100°F in the last 60 minutes, and

- Any RCS cold leg temperatures are equal to or less than 274°F.

The combination of these two conditions indicates the RCS barrier is under extreme challenge and should be considered a Potential Loss of the RCS barrier. [Ref. 11, 12]

Heat Sink-Red path is entered if:

- Narrow range level in both S/Gs is less than 4% [15%]

- Total feedwater flow to S/Gs is less than 200 gpm.

The combination of these two conditions indicates the heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and should be considered a Potential Loss of the RCS barrier. [Ref. 4, 5]

Critical Safety Function Status Tree (CSFST) setpoints enclosed in brackets are used under adverse containment conditions. Adverse containment condition thresholds apply when containment pressure is greater than 4 psig or containment radiation exceeds 10E+05 R/hr.

There is no "Loss" EAL associated with this item.

2. RCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak. Loss of subcooling is defined by:

- LESS THAN 20°F if the reactor is critical
- LESS THAN 30°F if the reactor is sub-critical

Core exit thermocouples LESS THAN 20°F is the subcooling margin threshold while critical. This is based on the minimum subcooling allowed for normal operation defined in Operating Procedure A-RC-36-D. [Ref. 23]

Core exit thermocouples LESS THAN 30°F is the subcooling margin threshold while subcritical. This is the level specified in Critical Safety Function Status Trees. IPEOPs define this value as a loss of RCS subcooling. [Ref. 1]

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one variable-speed, positive displacement charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak. 60 gpm is the design flow rate for each charging pump. [Ref. 13]

3. SG Tube Rupture

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL #4 and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PRZR pressure less than 1815 psig
- S/G pressure less than 500 psig
- Containment pressure greater than 4 psig

Per IPEOP E-0, Reactor Trip or Safety Injection, the Operators are directed to perform a manual Safety Injection actuation if PRZR level is less than 5% or RCS subcooling based on Core Exit Thermocouples (CETs) is less than 30°F.

By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Emergency per Containment Barrier "Loss" EAL #4. [Ref. 13, 14]

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The 30 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. [Ref. 8, 9, 10] This reading is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors R-40 and R-41.

There is no "Potential Loss" EAL associated with this item.

5. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

CONTAINMENT BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7)

The Containment Barrier includes the Shield Building and Containment and its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown lines +outside the Containment up to and including the isolation valve(s).

1. Critical Safety Function Status

There is no "Loss" EAL associated with this item.

RED path indicates an extreme challenge to the safety function. Containment-Red path is entered if containment pressure is equal to or greater than 46 psig. This pressure is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Emergency and General Emergency representing a potential loss of the third barrier. [Ref. 15, 16, 17]

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity. USAR Section 14.3.4.2 describes containment pressure response for a bounding LOCA. [Ref. 17]

Containment pressure and sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity.

The 46 PSIG for potential loss of containment is based on the containment design pressure. [Ref. 15, 16, 17]

If hydrogen concentration reaches or exceeds 6% in an oxygen rich environment, an explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred As described above, this EAL is primarily a discriminator between Site Emergency and General Emergency representing a potential loss of the third barrier. [Ref. 6, 18]

The third potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system (but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (23 psig) at which the equipment was supposed to have actuated. One internal containment spray pump and two containment fan cooler units comprise one train of depressurization equipment. This equipment will provide 100% of the required cooling capacity during post-accident conditions. Each internal containment spray system consists of a spray pump, spray header, nozzles, valves, piping, instruments, and controls to ensure an operable flow path capable of taking suction from the RWST upon an ESF actuation signal. Each containment fan cooler unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path. [Ref. 15, 16, 19, 20, 21]

3. Core Exit Thermocouples

There is no "Loss" EAL associated with this item.

In this EAL, the function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

RVLIS void fraction increasing and RCS subcooling less than or equal to 30°F [65°F] is indicative of core uncovery. When the reactor is at pressure and temperature, RVLIS should not be used for a quantitative value (i.e., top of active fuel). Function restoration procedure FR-C.2 specifies monitoring of RVLIS void fraction trend and RCS subcooling instead of the water level corresponding to the top of active fuel. This is defined by the Core Cooling - ORANGE path. The trend in RVLIS RCS void fraction is used to check the effectiveness of safety injection in restoring RCS inventory. If void fraction percent is decreasing and RCS subcooling based on Core Exit Thermocouples (CETs) is greater than 30°F [65°F], safety injection has been successful in restoring RCS inventory and core cooling. In the event that RCS void fraction is increasing and subcooling requirements are not met, core cooling continues to be degraded and some fuel cladding damage may occur. Setpoints enclosed in brackets are used under adverse containment conditions. Adverse containment condition thresholds apply when containment pressure is greater than 4 psig or containment radiation exceeds 10E+05 R/hr.

The conditions in this potential loss EAL represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, the Operating Crew will be directed to go to Severe Accident Management Guidelines (SACRG-1). [Ref. 1, 6, 7]

There is no "Loss" EAL associated with this item.

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Emergency. A faulted S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. Confirmation should be based on diagnostic activities consistent with E-0, Reactor Trip or Safety Injection. [Ref. 14]

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

A pressure boundary leakage of 10 gpm is used as the threshold in IC SU5.1, RCS Leakage, and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal charging capacity threshold in RCS Barrier "Potential Loss" EAL #2 (RCS Leak Rate) or not resulting in ECCS actuation in EAL #3 (SG Tube Rupture), this EAL results in a UE. For larger breaks, RCS barrier EALs #2 and #3 would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with RCS barrier "Loss" EAL #3 and would result in a Site Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

5. Containment Isolation Valve Status After Containment Isolation

This EAL is intended to address incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier "direct" in defining the release path clarifies that release paths through interfacing liquid systems is not applicable to this EAL. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no "Potential Loss" EAL associated with this item.

6. Significant Radioactive Inventory in Containment

There is no "Loss" EAL associated with this item.

The4000 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. [Ref. 8, 9, 10] A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors R-40 and R-41.

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

KNPP Basis Reference(s):

1. F-0.2 Core Cooling, Rev. F
2. FR-C.2 Response to Degraded Core Cooling, Rev. M
3. E-0 QRF Quick Reference Foldout, Section E-0, Rev. H
4. F-0.3 Heat Sink, Rev. E
5. FR-H.1 Response to Loss of Secondary Heat Sink, Rev. T
6. FR-C.1 Response to Inadequate Core Cooling, Rev. N
7. BKG FR-C.2 Response to Degraded Core Cooling, Rev. B
8. EPIP-TSC-09A Core Damage Assessment, Rev. K
9. C11617, Determination of Containment Radiation Monitor EALs, Rev 0
10. F-0.4 Integrity, Rev. E
11. FR-P-1 Response to Imminent Pressurized Thermal Shock, Rev. P
12. USAR Section 9.2.2, Rev. 18
13. E-0 Reactor Trip or Safety Injection, Rev. V
14. F-0.5 Containment, Rev. F
15. FR-Z.1 Response to High Containment Pressure, Rev. L
16. USAR Section 14.3.4.2, Rev. 18
17. N-RBV-18C POST-LOCA Hydrogen Control, Rev. K
18. Annunciator 47021F Containment Spray Activated, Rev. A
19. N-CCI-56A-CLA Reduced Inventory Cntmt Integrity Checklist - SG Secondary Side Intact, Rev. K
20. Technical Specifications LCO 3.3.c, Amendment No. 172
21. EOP Setpoints, Rev. 8/31/90
22. A-RC-36D Reactor Coolant Leak, Rev. AE

TABLE H-0

Recognition Category H

Hazards and Other Conditions Affecting Plant Safety

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
HU1 Natural and Destructive Phenomena Affecting the PROTECTED AREA. *Op. Modes: All*	HA1 Natural and Destructive Phenomena Affecting the Plant VITAL AREA. *Op. Modes: All*		
HU2 FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection. *Op. Modes: All*	HA2 FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*		
HU3 Release of Toxic or Flammable Gases Deemed Detrimental to Operation of the Plant. *Op. Modes: All*	HA3 Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*		
HU4 Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant. *Op. Modes: All*	HA4 Confirmed Security Event in a Plant PROTECTED AREA. *Op. Modes: All*	HS1 Confirmed Security Event in a Plant VITAL AREA. *Op. Modes: All*	HG1 Security Event Resulting in Loss Of Physical Control of the Facility. *Op. Modes: All*
HU5 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE. *Op. Modes: All*	HA6 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. *Op. Modes: All*	HS3 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency. *Op. Modes: All*	HG2 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. *Op. Modes: All*
	HA5 Control Room Evacuation Has Been Initiated. *Op. Modes: All*	HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established. *Op. Modes: All*	

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HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition -- UNUSUAL EVENT

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Level: (HU1.1 or HU1.2 or HU1.3 or HU1.4 or HU1.5 or HU1.6 or HU1.7)

HU1.1. Earthquake felt in plant as indicated by:

Consensus of Control Room operators on duty
AND
Activation of seismic monitor with Trigger light lit in Relay Room on RR159 (SER 330 Seismic Monitor Event)

HU1.2. Report by plant personnel of tornado or high winds GREATER THAN 100 mph striking within PROTECTED AREA boundary.

HU1.3. Vehicle crash into plant structures containing functions and systems required for safe shutdown of the plant within the PROTECTED AREA boundary.

HU1.4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

HU1.5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

HU1.6. Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode:

- Diesel Generator A Room
- Diesel Generator B Room
- Safeguards Alley
- Relay Room
- CRDM Equipment Room
- RHR Pump Pits
- Auxiliary Building Basement
- Screen House

HU1.7. High or low lake level in excess of column "Unusual Event", Lake-Forebay Level Thresholds, Table H-2 for GREATER THAN 15 minutes.

Table H-2 Lake-Forebay Level Thresholds (GREATER THAN 15 min.)							
Unusual Event				Alert			
Level	Number of Running Circulating Water Pumps			Level	Number of Running Circulating Water Pumps		
	0	1	2		0	1	2
High GREATER THAN OR EQUAL TO 586.0 ft	Above bottom of bar #2 on south wall	GREATER THAN OR EQUAL TO 98%*	GREATER THAN OR EQUAL TO 88%*	High GREATER THAN OR EQUAL TO 589.9 ft	Above bottom of bar #3 on south wall	Above bottom of bar #1 on south wall	GREATER THAN OR EQUAL TO 94%*
Low LESS THAN 569.5 ft	LESS THAN 53.1%*	LESS THAN 46.9%*	N/A	Low LESS THAN 568.5 ft	LESS THAN 50.0%*	N/A	N/A

* Computer point L9075A

Basis:

UEs in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

HU1.1. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection is based on instrumentation, or operator assessment [Ref. 1, 2]. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

> An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.

HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind value is based on site-specific FSAR design basis [Ref. 3]. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert. Even though the meteorological towers are outside of the Protected Area, winds in excess of 100 mph detected there can be assumed to be inside of the Protected Area.

HU1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant [Ref. 4]. If the crash is confirmed to affect a plant VITAL AREA, the event may be escalated to Alert.

For HU1.4 only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered [Ref. 4]. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable.

HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.

HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The listed internal flooding areas are those vulnerable areas indicated in the KNPP PRA that, should significant internal flooding occur (such as a Service Water or Circulating Water pipe rupture), could impact areas that contain systems required for safe shutdown of the plant that are not designed to be wetted or submerged [Ref. 5]. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring.

HU1.7 covers high lake (forebay) water level conditions that could be a precursor of more serious events as well as low lake (forebay) water level conditions which may threaten operability of plant cooling systems. Lake water level greater than or equal to 586 ft. International Great Lakes Datum (IGLD) corresponds to the floor elevation of the Service Water Pump Room and access tunnel. Lake water level less than 569.5 ft IGLD corresponds to one foot below the Alert (design) threshold [Ref. 6, 7, 8].

KNPP does not have instrumentation for taking direct readings of the lake level. However the intake forebay level is monitored for this purpose. When no circulating water pumps are operating, the intake forebay level is equal to lake level. However, when the Circulating Water Pumps are operating forebay level is reduced compared to actual lake level due to the hydraulic resistance of the plant intake. KNPP has correlated the intake forebay level with actual lake level when either one or both Circulating Water Pumps are operating, adjusting the EAL thresholds accordingly. In most cases the Circulating Water Pumps will trip (42% indicated forebay level) prior to exceeding the forebay level that corresponds to the low lake level threshold.

The classification should be declared if the threshold is exceeded for greater than 15 minutes. This allows for short duration dynamic effects associated with the KNPP forebay and will avoid unnecessary event declaration due to shifting of Circulating Water Pumps and other dynamic effects in the forebay.

The International Great Lakes Datum (IGLD 1955) is a reference used to represent water levels in the Great Lakes region.

KNPP Basis Reference(s):

1. USAR Table 5.2-1 Allowable Stress Criteria - Reactor Containment Vessel, Rev. 16
2. Alarm Response procedure 47023-K Seismic Trouble Beta Window Box #02-K3, Rev. E
3. USAR Section 5.2.2 Shield Building Design - Wind Load, Rev. 16
4. Drawing A-449 Plan of Plant Area, Fence, Lighting and CCTV Support, Rev. F
5. KNPP PRA Section 7.0 Internal Flooding Analysis Workbook, Rev. 0401
6. USAR Section 2.6 Hydrology, Rev. 18
7. Alarm Response Procedure 47051-N Forebay Level Low Beta Window Box #05-N1, Rev. C
8. KNPP – Safety Evaluation - Review for Kewaunee Proposed EAL Changes (TAC No. MB1860) 8/22/2001

HU2

Initiating Condition -- UNUSUAL EVENT

FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

Emergency Action Level:

HU2.1. FIRE in the PROTECTED AREA not extinguished within 15 minutes of control room notification or verification of a control room alarm

Basis:

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, detection is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). The applicable areas are limited and apply to buildings and areas contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs or other significant buildings or areas.

Escalation to a higher emergency class is by IC HA2, " FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown".

KNPP Basis Reference(s):

1. KNPP Fire Protection Program Plan Section 5.19, Rev. 5
2. Drawing A-449 Plan of Plant Area, Fence, Lighting and CCTV Support, Rev. F

HU3

Initiating Condition – UNUSUAL EVENT

Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability: All

Emergency Action Levels: (HU3.1 or HU3.2)

HU3.1. Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

HU3.2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the existence of uncontrolled releases of toxic or flammable gas that may enter the site boundary and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EALs are intended to not require significant assessment or quantification. The IC assumes an uncontrolled process that has the potential to affect plant operations, or personnel safety.

Escalation of this EAL is via HA3, which involves a quantified release of toxic or flammable gas affecting VITAL AREAs.

KNPP Basis Reference(s):

None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

Initiating Condition – UNUSUAL EVENT

Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.

Operating Mode Applicability: All

Emergency Action Levels: (HU4.1 or HU4.2)

HU4.1. Security Shift Supervisor reports ANY of the following:

- Suspected sabotage device discovered within the plant PROTECTED AREA
- Suspected sabotage device discovered outside the PROTECTED AREA or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A hostage or extortion situation that disrupts NORMAL PLANT OPERATIONS
- Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the PROTECTED AREA
- Malevolent use of a vehicle outside the PROTECTED AREA which disrupts normal plant operations

HU4.2 A credible site specific security threat notification

Basis:

Reference is made to the Security Shift Supervisor because this individual is the designated person on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

HU4.1 is based on the Security And Safeguards Contingency Plan. Security events which do not represent a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Examples of security events that indicate Potential Degradation in the Level of Safety of the Plant are provided below for consideration.

INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE would result in EAL escalation to an ALERT or higher.

The intent of HU4.2 is to ensure that appropriate notifications for the security threat are made in a timely manner. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the Security And Safeguards Contingency Plan.

A credible site specific security threat is a threat of physical attack to the plant that represents a potential degradation of the level of safety to the plant.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the Security And Safeguards Contingency Plan and Emergency Plans.

KNPP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Security And Safeguards Contingency Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition –UNUSUAL EVENT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.

Operating Mode Applicability: All

Emergency Action Level:

HU5.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

KNPP Basis Reference(s):

None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition -- ALERT

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HA1.1 or HA1.2 or HA1.3 or HA1.4 or HA1.5)

HA1.1. Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by activation of seismic monitor with OBE Limit Exceeded light lit in Relay Room on RR159 (SER 331 Seismic Monitor Operational Basis Earthquake)

HA1.2. Tornado or high winds GREATER THAN 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment located in Table H-1 areas or Control Room indication of degraded performance of those systems located within Table H-1 areas.

Table H-1 Safe Shutdown/VITAL Areas
• Shield Building (Reactor Building) • Auxiliary Building • Safeguards Alley • Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel) • Screenhouse/Forebay • Technical Support Center Basement • Control Room • Control Room AC Equipment Room • Relay Room • Safeguards Battery Rooms

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment located in Table H-1 areas or Control Room indication of degraded performance of those systems located within Table H-1 areas.

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any plant areas listed in Table H-1:

HA1.5. Uncontrolled flooding in the following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment:

- Diesel Generator A Room
- Diesel Generator B Room
- Safeguards Alley
- Relay Room
- CRDM Equipment Room
- RHR Pump Pits
- Auxiliary Building basement
- Screen House

HA1.6. High or low lake level in excess of column "Alert", Lake-Forebay Level Thresholds, Table H-2 for GREATER THAN 15 minutes.

Table H-2 Lake-Forebay Level Thresholds (GREATER THAN 15 min.)

Unusual Event				Alert			
Level	Number of Running Circulating Water Pumps			Level	Number of Running Circulating Water Pumps		
	0	1	2		0	1	2
High GREATER THAN OR EQUAL TO 586.0 ft	Above bottom of bar #2 on south wall	GREATER THAN OR EQUAL TO 98%*	GREATER THAN OR EQUAL TO 88%*	High GREATER THAN OR EQUAL TO 589.9 ft	Above bottom of bar #3 on south wall	Above bottom of bar #1 on south wall	GREATER THAN OR EQUAL TO 94%*
Low LESS THAN 569.5 ft	LESS THAN 53.1%*	LESS THAN 46.9%*	N/A	Low LESS THAN 568.5 ft	LESS THAN 50.0%*	N/A	N/A

* Computer point L9075A

Basis:

The EALs in this IC escalate from the UE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occurs on the basis of other ICs (e.g., System Malfunction).

HA1.1 is based on the USAR design basis operating earthquake of 0.06 g horizontal or 0.04 g vertical acceleration. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. [Ref. 1, 2]

HA1.2 is based on the FSAR design basis wind speed of 100 mph [Ref. 3, 4, 5]. Wind loads of this magnitude can cause damage to safety functions.

HA1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

HA1.4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. Table H-1 lists areas that contain systems and components required for the safe shutdown functions of the plant.. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The listed internal flooding areas are those vulnerable areas indicated in the KNPP PRA that should significant internal flooding occur (such as a Service Water or Circulating Water pipe rupture), could impact areas that contain systems required for safe shutdown of the plant that are not designed to be wetted or submerged. [Ref. 6, 7].. HA1.6 covers flooding or seiche. This EAL can be a precursor of more serious events. Lake water level greater than or equal to 588 ft International Great Lakes Datum (IGLD) corresponds to levels approaching design limits which if exceeded threatens operability of safety related equipment. Lake water level less than 568.5 ft IGLD corresponds to design levels (with added conservatism) to ensure Service Water Pumps have adequate NPSH and that vortexing does not occur [Ref. 8, 9].

KNPP does not have instrumentation for taking direct readings of the lake level. However the intake forebay level is monitored for this purpose. When no circulating water pumps are operating, the intake forebay level is equivalent to lake level. However, when the Circulating Water Pumps are operating forebay level is reduced compared to actual lake level due to the hydraulic resistance of the plant intake. KNPP has correlated the intake forebay level with actual lake level when either one or both Circulating Water Pumps are operating, adjusting the EAL thresholds accordingly. In most cases the Circulating Water Pumps will trip (42% indicated forebay level) prior to exceeding the forebay level that corresponds to the low lake level threshold.

The classification should be declared if the threshold is exceeded for greater than 15 minutes. This allows for short duration dynamic effects associated with the KNPP forebay and will avoid unnecessary event declaration due to shifting of Circulating Water Pumps and other dynamic effects in the forebay.

The International Great Lakes Datum (IGLD 1955) is a reference used to represent water levels in the Great Lakes region.

KNPP Basis Reference(s):

1. USAR Table 5.2-1 Allowable Stress Criteria - Reactor Containment Vessel, Rev. 16
2. Alarm Response procedure 47023-K Seismic Trouble Beta Window Box #02-K3, Rev. E
3. USAR Section 5.2.2 Shield Building Design - Wind Load, Rev. 16
4. KNPP Fire Protection Program Plan Section 5.19, Rev. 5
5. Drawing A-449 Plan of Plant Area, Fence, Lighting and CCTV Support, Rev. F
6. E-CW-04 Loss of Circulating Water, Rev. V
7. KNPP PRA Section 7.0 Internal Flooding Analysis Workbook, Rev. 0401
8. USAR Section 2.6 Hydrology, Rev. 18
9. KNPP – Safety Evaluation - Review for Kewaunee Proposed EAL Changes (TAC No. MB1860) 8/22/2001

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition -- ALERT

FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Emergency Action Level:

HA2.1. FIRE or EXPLOSION in any of the following areas (Table H-1):

Table H-1 Safe Shutdown/VITAL Areas
• Shield Building (Reactor Building) • Auxiliary Building • Safeguards Alley • Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel) • Screenhouse/Forebay • Technical Support Center Basement • Control Room • Control Room AC Equipment Room • Relay Room • Safeguards Battery Rooms

AND

Affected safety system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment needed for safe shutdown.

Basis:

These areas contain systems and components required for the safe shutdown functions of the plant. The KNPP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. This will make it easier to determine if the FIRE or EXPLOSION is potentially affecting one or more redundant trains of safety systems [Ref. 1, 2]. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs / EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems. Thus, the designation of a single train was intentional and is appropriate when the FIRE / EXPLOSION is large enough to affect more than one component.

This situation is not the same as removing equipment for maintenance that is covered by a plant's Technical Specifications. Removal of equipment for maintenance is a planned activity controlled in accordance with procedures and, as such, does not constitute a substantial degradation in the level of safety of the plant. A FIRE / EXPLOSION is an UNPLANNED activity and, as such, does constitute a substantial degradation in the level of safety of the plant. In this situation, an Alert classification is warranted.

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform these damage assessments. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

KNPP Basis Reference(s):

1. KNPP Fire Protection Program Plan Section 5.19, Rev. 5
2. Drawing A-449 Plan of Plant Area, Fence, Lighting and CCTV Support, Rev. F

HA3

Initiating Condition -- ALERT

Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Emergency Action Levels: (HA3.1 or HA3.2)

HA3.1. Report or detection of toxic gases within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

Table H-1 Safe Shutdown/VITAL Areas
• Shield Building (Reactor Building)
• Auxiliary Building
• Safeguards Alley
• Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel)
• Screenhouse/Forebay
• Technical Support Center Basement
• Control Room
• Control Room AC Equipment Room
• Relay Room
• Safeguards Battery Rooms

HA3.2. Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1).

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous (in actual contact with or immediately adjacent) to plant Safe Shutdown/VITAL AREAs or other significant buildings or areas [Ref. 1, 2]. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant Safe Shutdown/VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

HA3.1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a Safe Shutdown/VITAL AREA or any area or building contiguous to Safe Shutdown/VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

HA3.2 is met when the flammable gas concentration in a Safe Shutdown/VITAL AREA or any building or area contiguous to a Safe Shutdown/VITAL AREA exceed the LOWER FLAMMABILITY LIMIT. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

KNPP Basis Reference(s):

1. KNPP Fire Protection Program Plan Section 5.19, Rev. 5
2. Drawing A-449 Plan of Plant Area, Fence, Lighting and CCTV Support, Rev. F

HA4

Initiating Condition -- ALERT

Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HA4.1 or HA4.2)

HA4.1. INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE.

HA4.2. Security Shift Supervisor reports any of the following:

- Sabotage device discovered in the plant PROTECTED AREA
- Standoff attack on the protected area by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for > 30 minutes:
  - Credible bomb threats
  - Hostage / Extortion
  - Suspicious Fire or Explosion
  - Significant Security System Hardware Failure
  - Loss of contact with Security Officers

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the UE. A confirmed INTRUSION report is satisfied if physical evidence indicates the presence of a HOSTILE FORCE within the PROTECTED AREA.

The Security And Safeguards Contingency Plan identifies numerous events/conditions that constitute a threat/compromise to station security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered.

INTRUSION into a VITAL AREA by a HOSTILE FORCE will escalate this event to a Site Area Emergency.

Reference is made to Security Shift Supervisor because this individual is the designated person on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

KNPP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Security And Safeguards Contingency Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. Physical Security Plan

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition -- ALERT

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

Emergency Action Level:

HA5.1. Entry into E-O-06, Fire in Alternate Fire Zone for control room evacuation.

Basis:

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. E-O-06, Fire in Alternate Fire Zone, provides specific instructions for evacuating the Control Room/Building and establishing plant control at the Dedicated Shutdown Panel and in alternate locations. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

KNPP Basis Reference(s):

1. E-O-06, Fire in Alternate Fire Zone, Rev. W

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

Initiating Condition -- ALERT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.

Operating Mode Applicability: All

Emergency Action Level:

HA6.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class. Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.

KNPP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents, October 1991
2. EPIP-AD-19 Determining Protective Action Recommendations, Rev. T

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition – SITE AREA EMERGENCY

Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HS1.1 or HS1.2)

HS1.1. INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.

HS1.2. Security Supervision reports ANY of the following:
- A security event that results in the loss of control of ANY VITAL AREAS (other than Control Room)
- Imminent loss of physical control of the facility (remote shutdown capability) due to a security event
- A confirmed sabotage discovered in a VITAL AREA

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a HOSTILE FORCE has progressed from the PROTECTED AREA to a VITAL AREA.

Consideration is given to the following types of events when evaluating an event against the criteria of the site specific Security Contingency Plan: SABOTAGE and HOSTAGE / EXTORTION. The Safeguards Contingency Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered.

Loss of Plant Control would escalate this event to a GENERAL EMERGENCY.

Reference is made to Security Shift Supervisor because this individual is the designated person on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

KNPP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Security And Safeguards Contingency Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. Physical Security Plan

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS2

Initiating Condition – SITE AREA EMERGENCY

Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

Operating Mode Applicability: All

Emergency Action Level:

HS2.1. Control room evacuation has been initiated.

AND

Control of the plant cannot be established per E-O-06, Fire in Alternate Fire Zone within 15 minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The determination of whether or not control is established at the Dedicated Shutdown Panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the time for transfer that the operator has control of the plant from the Dedicated Shutdown Panel.

E-O-06, Fire in Alternate Fire Zone, provides specific instructions for evacuating the Control Room/Building and establishing plant control at the Dedicated Shutdown Panel and in alternate locations.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Director Judgment ICs.

KNPP Basis Reference(s):

1. E-O-06, Fire in Alternate Fire Zone, Rev. W

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition – SITE AREA EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HS3.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency. Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.

KNPP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents, October 1991
2. EPIP-AD-19 Determining Protective Action Recommendations, Rev. T

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition – GENERAL EMERGENCY

Security Event Resulting in Loss Of Physical Control of the Facility.

Operating Mode Applicability: All

Emergency Action Level:

HG1.1. A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of EITHER:

A VITAL AREA (including the Control Room) such that operation of equipment required for safe shutdown is lost

OR

Spent fuel pool cooling systems if imminent fuel damage is likely

Basis:

This IC encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAs (containing vital equipment or controls of vital equipment, including the Control Room) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink). If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

This EAL should also address loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).

Loss of physical control of the control room or remote shutdown capability may prevent the ability to maintain safety functions.

KNPP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Security And Safeguards Contingency Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. Physical Security Plan

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG2

Initiating Condition – GENERAL EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class. Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.

KNPP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents, October 1991
2. EPIP-AD-19 Determining Protective Action Recommendations, Rev. T

Table S-0

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	**SA5** AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	**SS1** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	**SG1** Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*
	SA2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was Successful. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown*	**SS2** Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was NOT Successful. *Op. Modes: Power Operation, Hot Standby*	**SG2** Failure of the Reactor Protection System to Complete an Automatic Reactor Trip and Manual Reactor Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core. *Op. Modes: Power Operation, Hot Standby*
SU2 Inability to Reach Required Shutdown Within Technical Specification Limits. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*		**SS4** Complete Loss of Heat Removal Capability. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	
SU3 UNPLANNED Loss of Most or All Safety System Annunciation or Indication In The Control Room for GREATER THAN 15 Minutes *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	**SA4** UNPLANNED Loss of Most or All Safety System Annunciation or Indication In Control Room With Either (1) a SIGNIFICANT TRANSIENT In Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	**SS6** Inability to Monitor a SIGNIFICANT TRANSIENT In Progress. *Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown*	

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

SS3 Loss of All Vital DC Power.
Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown

SU4 Fuel Clad Degradation.
Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown

SU5 RCS Leakage.
Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown

SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Op. Modes: Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown

SU8 Inadvertent Criticality.
Op Modes: Hot Shutdown, Intermediate Shutdown

SU1

Initiating Condition -- UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SU1.1. Loss of all offsite power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes.

AND

Emergency diesel generators are supplying power to Bus 5 AND Bus 6.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system is divided into six busses, two of which are Engineered Safety Features (ESF) busses 5 and 6. The ESF busses supply power to Safety Injection (SI) pumps, Residual Heat Removal (RHR) pumps, containment heat removal equipment, etc.

Offsite power is available from the 345 kVAC and 138 kVAC systems. The 345 kVAC system is connected to the North Appleton line, the Point Beach line, the main transformers, and transformer T-10. The 345 kVAC is the normal supply to the 13.8 kVAC system through transformer T-10, which feeds the Tertiary Auxiliary Transformer (TAT). The TAT normally provides power to ESF bus 5. The TAT is not considered available to power both ESF busses in an emergency situation due to its size. As a contingency, however, it is acceptable to use the TAT to power both ESF busses when guidance for sequencing and monitoring TAT loads is available in the Control Room. The Reserve Auxiliary Transformer (RAT) and Main Auxiliary Transformer (MAT) provide backup sources to bus 5, in that order.

The 138 kVAC system is connected to the Shoto/Mishicot line, the East Krok line and transformer T-10. The 138 kVAC system is the normal supply to the Reserve Auxiliary Transformer (RAT) via the East and West substation busses. (When the 345 kVAC system is unavailable, the 138 kVAC system can supply power to transformer T-10 and the TAT.) The RAT normally provides power to ESF bus 6. The TAT and MAT provide backup sources to bus 6 in that order.

When the main turbine generator is on line, generator output supplies power to the Main Auxiliary Transformer (MAT) and the 4160 VAC busses. When the main turbine generator is off line, the 345 kVAC system can be aligned to backfeed the MAT. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval associated with this EAL. If off-normal plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source.

Following a loss of power, ECA 0.0 provides guidance to restore power to ESF busses. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 kVAC system supplying power to transformer T-10 and the TAT
- 138 kVAC system supplying power to transformer T-10 and the TAT
- 138 kVAC system supplying power to the RAT
- 345 kVAC system supplying power to the MAT on backfeed through the main transformers when the main turbine generator is off line

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. GNP-08.04.01 Shutdown Safety Assessment, Rev. K

SYSTEM MALFUNCTION

SU2

Initiating Condition -- UNUSUAL EVENT

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SU2.1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the KNPP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

KNPP Basis Reference(s):

1. KNPP Technical Specifications

SYSTEM MALFUNCTION

SU3

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes

Operating Mode Applicability: Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SU3.1. UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., PPCS, SER or SPDS).

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This is addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified panels for this EAL include annunciators and indicators identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This UE will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

KNPP Basis Reference(s):

1. USAR Figure 7.7-1, Rev. 18
2. A-SER-52B Abnormal Sequential Event Recorder, Annunciator, and Status Panel System, Rev. C
3. N-CP-46 Honeywell Plant Process Computer, Rev. S
4. A-CP-46 Abnormal Honeywell Plant Process Computer, Rev. AR
5. N-SER-52 Control Room Sequential Event Recorder, Rev. D

Initiating Condition -- UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Levels: (SU4.1 or SU4.2)

SU4.1. RCS Letdown Line (R-9) radiation monitor GREATER THAN 2000 mR/hr indicating fuel clad degradation.

SU4.2. Coolant sample activity GREATER THAN ANY of the following indicating fuel clad degradation:

- 1.0 μCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval
- 60 μCi/gram dose equivalent Iodine-131
- 91/$\bar{E}$ μCi/cc gross radioactivity

Basis:

This IC is included as a UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. SU5.1 addresses RCS Letdown Line (R-9) radiation monitor readings that provide indication of fuel clad integrity. [Ref. 4 & 5] SU4.2 addresses coolant samples exceeding coolant technical specifications [Ref. 1].

2000 mR/hr was calculated using the following:

0.01% fuel cladding defect equals 7.2E+1 mR/hr increase on R-9 [Ref. 4]
0.2745% fuel cladding defect equals 1.0 μCi/gram dose equivalent Iodine-131 [Ref. 5].

Therefore 1976.4 mR/hr increase on R-9 is equal to 1.0 μCi/gram dose equivalent Iodine-131

R-9 background is equivalent to 56 mR/hr [Ref. 4], which is added to the calculated dose rate above.
With the addition of background R-9 will read 2032.4 mR/hr (rounded to 2000 mR/hr) equal to 1.0 μCi/gram dose equivalent Iodine-131.

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. Though the referenced Technical Specification limits are applicable when average reactor coolant temperature is GREATER THAN 500°F, it is appropriate that the EAL's be applicable in all modes, as they indicate a potential degradation in the level of safety of the plant. The companion IC to SU4 for the Cold Shutdown/Refueling modes is CU5.

KNPP Basis Reference(s):

1. Technical Specifications LCO 3.1.c.1.A, Amendment No. 167
2. E-2021 Integrated Logic Diagram Radiation Monitoring, Rev. X
3. A-RC-36A High Reactor Coolant Activity, Rev. J
4. USAR Section 9, Rev. 16
5. CN-CRA-99-28 Rev. 1

SU5

Initiating Condition -- UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Levels: (SU5.1 or SU5.2)

SU5.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

SU5.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Positive indications in the Control Room of Reactor Coolant System (RCS) leakage to the containment are provided by equipment that monitors:

- Charging/Letdown flow mismatch
- Containment air activity
- Containment atmosphere humidity
- Containment Sump A in leakage

[Ref. 1, 2]

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). SP-36-82 provides instructions for calculating primary system leak rate by water inventory balances for off-normal events and for operations troubleshooting [Ref, 2]. The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

KNPP Basis Reference(s):

1. Technical Specifications LCO 3.1.d, Amendment No. 165
2. SP-36-82 Reactor Coolant System Leak Rate Check, Rev. AE

SYSTEM MALFUNCTION

SU6

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Levels: (SU6.1 or SU6.2)

SU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Intraplant Paging (Gai-tronics)
• Sound powered phones
• PBX telephone system
• Personal communications system (PCS phones)
• Portable radio communications system

SU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• PBX telephone system
• NRC FTS System (including ENS and HPN)
• Dial select phones

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

Table C-1 onsite communications loss encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies). Due to its limited capability, the emergency gai-tronics is not listed in Table C-1.

Table C-2 offsite communications loss encompasses the loss of all means of communications with offsite authorities. This includes the NRC FTS System (including Emergency Notification System - ENS and Health Physics Network – HPN), commercial telephone lines, telecopy transmissions, and dedicated phone systems.

KNPP Basis Reference(s):

1. N-COM-44-CL Communications Systems CL, Rev. K

SU8

Initiating Condition -- UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability: Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements such as shutdown bank withdrawal. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate monitors (NI-31D/32D - Source Range Startup Rate).

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

Note: This EAL is SU8 following SU6. SU7 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

KNPP Basis Reference(s):

1. N-0-02 Plant Startup from Hot Shutdown to 35% Power, Rev. AN

SYSTEM MALFUNCTION

SA2

Initiating Condition -- ALERT

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was Successful.

Operating Mode Applicability: Power Operation
Hot Standby
Hot Shutdown

Emergency Action Level:

SA2.1. Indication(s) exist that a Reactor Protection System setpoint was exceeded
AND
RPS automatic trip did <u>not</u> reduce power to LESS THAN 5%
AND
Any of the following operator actions are successful in reducing power to LESS THAN 5%:

- Use of Manual Reactor Trip push buttons
- De-energizing Busses 33 AND 43

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual trip is any set of actions by the reactor operator(s) at the control room consoles which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button). Failure of manual trip would escalate the event to a Site Area Emergency.

Following a successful reactor trip, nuclear power promptly drops to only a few percent of nominal, and then decays away to a level some 8 decades less. Reactor power levels resulting from radioactive fission product decay are never more than a few percent of nominal power and also lower in time. Heat removal safety systems are sized to remove only decay heat and not significant core power. Reactor power levels at or above 5% (in a core that is supposed to be shutdown) are considered an extreme challenge to the Fuel Cladding barrier and warrant a Critical Safety Function Status Tree (CSFST) Subcriticality-Red path priority. The setpoint has been chosen because it is clearly readable on the power range meters. Reactor power levels in the power range are indicated on Mechanical Control Console "B" nuclear instruments NI-41, 42, 43 and 44.

Following any automatic reactor trip signal, plant procedures prescribe operator insertion of redundant manual trip signals to ensure reactor shutdown is achieved. A successful manual trip is any set of actions by the reactor operator(s) in the Control Room that causes control rods to be rapidly inserted into the core and brings the reactor subcritical. Manual trip includes the procedural direction to deenergize Busses 33 and 43 to ensure rod insertion. Control rod insertion completed from the Rod Drive Equipment Room is not considered a successful manual trip as action is required outside the Control Room. Manual insertion of control rods from the Control Room is not considered rapid insertion that brings the reactor sub-critical.

Note: This EAL is SA2 following SU8. SA1 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

KNPP Basis Reference(s):

1. E-0 Reactor Trip or Safety Injection, Rev. V
2. ES-0.1 Reactor Trip Response, Rev. P
3. F-0.1 Subcriticality, Rev. C
4. Technical Specifications 2.3.a, Amendment No. 162

SYSTEM MALFUNCTION

SA4

Initiating Condition -- ALERT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SA4.1. UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console

AND

Either of the following: (a or b)

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

COMPENSATORY NON-ALARMING INDICATIONS include the plant process computer (PPCS), SPDS, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified panels for this EAL include annunciators and indicators identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

Note: This EAL is SA4 following SA2. SA3 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

KNPP Basis Reference(s):

1. USAR Figure 7.7-1, Rev. 18
2. A-SER-52B Abnormal Sequential Event Recorder, Annunciator, and Status Panel System, Rev. C
3. NEI 99-01, Rev. 4, Section 5.4 Definitions
4. N-CP-46 Honeywell Plant Process Computer, Rev. S
5. A-CP-46 Abnormal Honeywell Plant Process Computer, Rev. AR
6. N-SER-52 Control Room Sequential Event Recorder, Rev. D

SYSTEM MALFUNCTION

SA5

Initiating Condition -- ALERT

AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SA5.1. AC power capability to Bus 5 AND Bus 6 reduced to only one of the following sources for GREATER THAN 15 minutes:

- One emergency diesel generator (A or B)
- TAT
- RAT
- MAT on backfeed

AND

Any additional single failure will result in station blackout.

Basis:

This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power. Offsite power sources include the 345 kVAC system or 138 kVAC system to the Tertiary Auxiliary Transformer (TAT), the 345 kVAC system or 138 kVAC system to the Reserve Auxiliary Transformer (RAT), and the 345 kVAC system to the Main Auxiliary Transformer (MAT) on backfeed through the main transformers. Note that the time required to effect a backfeed to the MAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source. Onsite power sources consist of 1A Diesel Generator to Bus 5 and 1B Diesel Generator to Bus 6. Several combinations of power failures could therefore satisfy this EAL. The subsequent loss of the single remaining power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. USAR Section 8.2.4, Rev. 18
5. GNP-08.04.01 Shutdown Safety Assessment, Rev. K

SYSTEM MALFUNCTION

SS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability: Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SS1.1. Loss of ALL power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Service Water System. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency.

Offsite power sources include the 345 kVAC system or 138 kVAC system to the Tertiary Auxiliary Transformer (TAT), the 345 kVAC or 138 kVAC system to the Reserve Auxiliary Transformer (RAT), and the 345 kVAC system to the Main Auxiliary Transformer (MAT) on backfeed through the main transformers. Note that the time required to effect a backfeed to the MAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source. Onsite power sources consist of 1A Diesel Generator to Bus 5 and 1B Diesel Generator to Bus 6.

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to safety-related 4160 VAC busses. Even though a safety-related 4160 VAC bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable. If this bus was the only energized bus then a Site Area Emergency per SS1 should be declared.

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. USAR Section 8.2.4, Rev. 18
5. GNP-08.04.01 Shutdown Safety Assessment, Rev. K

SYSTEM MALFUNCTION

SS2

Initiating Condition -- SITE AREA EMERGENCY

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was NOT Successful.

Operating Mode Applicability: Power Operation
Hot Standby

Emergency Action Level:

SS2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%. Manual Reactor Trips include use of Manual Reactor Trip push buttons or De-energizing Busses 33 AND 43.

Basis:

Automatic and manual trip are not considered successful if action away from the Control Room was required to trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Automatic or manual reactor trip is considered successful if actions taken (use of Manual Reactor Trip push buttons or De-energizing Busses 33 AND 43) result in reducing reactor power less than 5%. Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. Automatic and manual trips are not considered successful if action away from the Control Room (e.g., Rod Drive Equipment Room) is required to trip the reactor. Manual insertion of control rods from the Control Room is not considered rapid insertion that brings the reactor sub-critical.

KNPP Basis Reference(s):

1. E-0 Reactor Trip or Safety Injection, Rev. V
2. ES-0.1 Reactor Trip Response, Rev. P
3. F-0.1 Subcriticality, Rev. C
4. FR-S.1 Response to Nuclear Power Generation/ATWS, Rev. Q

SYSTEM MALFUNCTION

SS3

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SS3.1. Loss of all vital DC power based on LESS THAN 105 VDC on Train A AND Train B Safeguards DC Distribution Systems for GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The loss of a safeguards DC train consists of a combination of loss of power to specified DC distribution panels. These panels include: BRA (BRB)-102, and BRA (BRB)-104. In all cases, BRA (BRB)-102 panel indicating less than 105 VDC constitutes a loss of the associated DC distribution train. However, a loss of power to the BRA (BRB) -104 panel, which does not have voltage indication, also constitutes a loss of the associated DC distribution train.

125 VDC safeguard main distribution cabinets (BRA-102 and BRB-102) supply two safeguard sub-distribution cabinets (BRA-104 and BRB-104) and provide for connection of safeguard batteries (BRA-101 and BRB-101) to their associated battery chargers (BRA-108 and BRB-108). The combination of low voltages on the specified distribution cabinets results in a total loss of vital 125 VDC power. The 125 VDC safeguards distribution system supplies circuit breaker control power, Control Room alarms, Control Room controls, diesel generator controls, and the Reactor Protection System. The 125 VDC safeguard system is also the standby power source to the AC inverters. BRA-102 and BRB-102 voltage is displayed on Control Room indicators 4494001 and 4494002, respectively. Undervoltage is alarmed on Control Room Sequence of Event Recorder (SER) points 490011196 and 490011200 and annunciators 447101A and 47101B, respectively.

Each of the 125 VDC batteries has been sized to carry the expected shutdown loads following a reactor trip and a loss of all AC power for a period of eight hours without battery terminal voltage falling below 105 VDC. This voltage value therefore incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads. The nominal battery cell voltage is 2.20 VDC. Low battery terminal voltage activates Control Room SER point 49001832 and annunciator 47105A. The batteries are located in Battery Rooms A and B on the Turbine Building Mezzanine Floor (606 ft el.).

KNPP Basis Reference(s):

1. USAR 8.2.2, Rev. 18
2. USAR 8.2.3, Rev. 18
3. Technical Specifications B3.7, 4/23/2001
4. Plant Drawing 237127A-E233, Rev. AQ

SYSTEM MALFUNCTION

SS4

Initiating Condition -- SITE AREA EMERGENCY

Complete Loss of Heat Removal Capability.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SS4.1. Loss of core cooling and heat sink.

Basis:

This EAL addresses complete loss of functions, including Service Water System, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

KNPP Basis Reference(s):

None

SYSTEM MALFUNCTION

SS6

Initiating Condition -- SITE AREA EMERGENCY

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SS6.1. Loss of most or all annunciators associated with safety systems on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console.

AND

SIGNIFICANT TRANSIENT in progress.

AND

COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

AND

Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

COMPENSATORY NON-ALARMING INDICATIONS include the plant process computer (PPCS), SPDS, plant recorders, or plant instrument displays in the control room.

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

Note: This EAL is SS6 following SS4. SS5 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

KNPP Basis Reference(s):

1. USAR Figure 7.7-1, Rev. 18
2. A-SER-52B Abnormal Sequential Event Recorder, Annunciator, and Status Panel System, Rev. C
3. NEI 99-01, Rev. 4, Section 5.4 Definitions
7. N-CP-46 Honeywell Plant Process Computer, Rev. S
8. A-CP-46 Abnormal Honeywell Plant Process Computer, Rev. AR
9. N-SER-52 Control Room Sequential Event Recorder, Rev. D
10. UG-0 User's Guide for Emergency and Abnormal Procedures, Rev. D
11. F-0.1 Subcriticality, Rev. C
12. F-0.2 Core Cooling, Rev. F
13. F-0.3 Heat Sink, Rev. E
14. F-0.4 Integrity, Rev. E
15. F-0.5 Containment, Rev. F
16. F-0.6 Inventory, Rev. F

SYSTEM MALFUNCTION

SG1

Initiating Condition -- GENERAL EMERGENCY

Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation
Hot Standby
Hot Shutdown
Intermediate Shutdown

Emergency Action Level:

SG1.1. Loss of all offsite power to Bus 5 AND Bus 6

AND

Failure of all emergency diesel generators to supply power to Bus 5 AND Bus 6.

AND

Either of the following: (a or b)

a. Restoration of either Bus 5 OR Bus 6 within 4 hours is not likely

OR

b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by a Core Cooling-RED or Core Cooling-ORANGE

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Service Water System. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The 4 hours to restore AC power is based on the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout,". Four hours includes appropriate allowance for offsite emergency response including evacuation of surrounding areas. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

Offsite power sources include the 345 kVAC system or 138 kVAC system to the Tertiary Auxiliary Transformer (TAT), the 345 kVAC system or 138 kVAC system to the Reserve Auxiliary Transformer (RAT), and the 345 kVAC system to the Main Auxiliary Transformer (MAT) on backfeed through the main transformers. Time required to effect a backfeed to the MAT is likely longer than the four hours. If shutddown plant conditions have already established the backfeed, however, its power to the ESF busses may be considered an offsite power source. Onsite power sources consist of Diesel Generator A to Bus 5 and Diesel Generator B to Bus 6.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.
In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table F-1 for more information.)

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

KNPP Basis Reference(s):

1. ECA-0.0 Loss of All AC Power, Rev. AB
2. USAR Figure 8.2-2, Rev. 16
3. USAR Section 8.2.3, Rev. 18
4. USAR Section 8.2.4, Rev. 18
5. GNP-08.04.01 Shutdown Safety Assessment, Rev. K
6. F-0.2 Core Cooling, Rev. F
7. FR-C.2 Response to Degraded Core Cooling, Rev. M
8. E-0 QRF Quick Reference Foldout, Section E-0, Rev. H

SYSTEM MALFUNCTION

SG2

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to Complete an Automatic Reactor Trip and Manual Reactor Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation
Hot Standby

Emergency Action Level:

SG2.1. Indication(s) exist that automatic and manual reactor trip were NOT successful in reducing power to LESS THAN 5%.

AND

Either of the following: (a or b)

a. Indication(s) exists that the core cooling is extremely challenged as indicated by Core Cooling - RED.

OR

b. Indication(s) exists that heat removal is extremely challenged as indicated by Heat Sink - RED.

Basis:

Automatic and manual reactor trips are not considered successful if action away from the reactor control console is required to trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities to reduce reactor power, such as emergency boration, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. This EAL equates to a Core Cooling RED condition and an entry into function restoration procedure FR-C.1.

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. This EAL equates to a Heat Sink RED condition.

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

Automatic or manual reactor trip is considered successful if actions taken (use of Manual Reactor Trip push buttons or De-energizing Busses 33 AND 43) result in reducing reactor power less than 5%. Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. Automatic and manual trips are not considered successful if action away from the Control Room (e.g., Rod Drive Equipment Room) is required to trip the reactor. Manual insertion of control rods from the Control Room is not considered rapid insertion that brings the reactor sub-critical.

KNPP Basis Reference(s):

1. E-0 Reactor Trip or Safety Injection, Rev. V
2. ES-0.1 Reactor Trip Response, Rev. P
3. F-0.1 Subcriticality, Rev. C
4. FR-S.1 Response to Nuclear Power Generation/ATWS, Rev. Q
5. F-0.2 Core Cooling, Rev. F
6. FR-C.1 Response to Inadequate Core Cooling, Rev. N
7. F-0.3 Heat Sink, Rev. E
8. FR-H.1 Response to Loss of Secondary Heat Sink, Rev. T

10CFR 50.54(q) REVIEW FORM

Description of Change:
KNPP Emergency Action Levels (EALs) are being revised to the NEI 99-01 rev 4 EAL scheme.

☒ **Plan Sections/Procedure(s) #:** EPIP-AD-02 **Revision(s) #:** TBD
☐ **Mod #:** __________
☒ **Other:** KNPP Emergency Action Levels and KNPP Emergency Plan

Is the proposed change purely editorial in nature (see definition)? *[If YES, discontinue review process and process the procedure change.]* ☐ **YES** ☒ **NO**
Does the proposed change affect any of the following: [*Check 'yes' or 'no'. Reference the actual standards/elements.]*

50.47	**PARAPHRASED STANDARD**	**YES**	**NO**
(b)(1)	Primary responsibilities of Site/NMC, State, County, or Tribal organizations.	☐	☒
	Responsibilities of supporting organizations.	☐	☒
	Initial staffing or augmentation	☐	☒
(b)(2)	On-shift responsibilities for emergency response.	☐	☒
	Staffing for initial accident response	☐	☒
	Timely augmentation	☐	☒
	Interfaces among onsite and offsite response activities.	☐	☒
(b)(3)	Arrangements for requesting and using assistance resources.	☐	☒
	Accommodations at the EOF for state and county staff.	☐	☒
	Other organizations capable of augmenting response are identified.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(4) RSPS	Emergency classification and action level scheme.	☒	☐
	State/county minimum response based on site information.	☐	☒
	EAL Initiating Condition setpoints, or thresholds.	☒	☐
(b)(5) RSPS	Process for notification of state/county response organizations.	☐	☒
	Notification of emergency personnel.	☐	☒
	Procedure for initial and follow-up messages.	☐	☒
	ANS notification within the 10-mile EPZ	☐	☒
(b)(6)	Provisions for prompt communication among principal response organizations to emergency response personnel and to the public.	☐	☒
(b)(7)	Public information distributed on a periodic basis.	☐	☒
	News media points of contact established.	☐	☒
	Procedures for coordinated dissemination of info to the public.	☐	☒
(b)(8)	Emergency response facilities, equipment, and maintenance.	☐	☒
(b)(9) RSPS	Methods, systems, or equipment for assessing and monitoring actual or potential offsite consequences.	☐	☒
(b)(10) RSPS	Range of protective actions for the Plume EPZ established.	☐	☒
	Guidelines for choice of PARs in place.	☐	☒
	Protective actions for Ingestion Pathway EPZ established.	☐	☒
(b)(11)	Controlling radiological exposure for emergency workers.	☐	☒
(b)(12)	Arrangements for medical service for contaminated injured individuals.	☐	☒
(b)(13)	General plans for recovery and reentry.	☐	☒
(b)(14)	Exercise or drill conduct and corrective action system.	☐	☒
(b)(15)	Radiological emergency response training.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(16)	Responsibilities for plan development, review, and distribution of emergency procedures established.	☐	☒
	EP Staff is properly trained.	☐	☒
EP	Implementation of other federal regulations and requirements or formal commitments related to the Emergency Preparedness Program.	☐	☒
ERDS	The operation, maintenance, or testing requirements of the ERDS.	☐	☒

App. E	PARAPHRASED ELEMENT	YES	NO
IV. A	Organization	☐	☒
IV. B	Assessment actions	☒	☐
IV. C	Activation of emergency response	☒	☐
IV. D	Notification procedures	☐	☒
IV E	Emergency facilities and equipment	☐	☒
IV. F	Training	☐	☒
IV. G	Maintaining emergency preparedness	☐	☒
IV. H	Recovery	☐	☒

<table>
<tr><th rowspan="2">STANDARDS AND/OR ELEMENTS EFFECTED</th><th rowspan="2">DESCRIPTION OF EFFECT</th><th colspan="2">DECREASED EFFECTIVENESS?</th></tr>
<tr><th>YES</th><th>NO</th></tr>
<tr><td rowspan="5">50.47(b)(4), App. E IV.B and IV.C</td><td>Background and Scope:
KNPP is revising from NUREG-0654 EAL scheme to NEI 99-01 rev 4 EAL scheme.</td><td rowspan="5">☒</td><td rowspan="5">☐</td></tr>
<tr><td>Program Requirements:
It is required by the CFR sections that the plant have a standard emergency classification and action level scheme. The scheme is required to assess the plant condition and make Emergency Classification determination.</td></tr>
<tr><td>Change Comparison:
Due to improvements in the NEI 99-01 rev 4 EAL scheme, implementation of the NEI 99-01 rev 4 EAL scheme will enhance KNPP's ability to classification emergency conditions.</td></tr>
<tr><td>Change Assessment:
EAL scheme change requires NRC pre-approval. Without pre-approval the change would be considered a decrease in effectiveness.</td></tr>
<tr><td>Justification:
NEI 99-01 rev 4 was endorsed by the NRC via Reg. Guide 1.101, rev 4 July 2003. Therefore, implementation of the NEI 99-01 rev 4 EAL scheme is appropriate after NRC approval.</td></tr>
</table>

<table>
<tr><th></th><th>YES</th><th>NO</th></tr>
<tr><td>This procedure can be processed without prior NRC approval.</td><td>☐</td><td>☒</td></tr>
<tr><td>This procedure change requires prior NRC approval.</td><td>☒</td><td>☐</td></tr>
<tr><td colspan="3">Document all references used for this review:</td></tr>
</table>

NEI 99-01 rev 4	RIS 2003-18 rev 4		

Prepared By: _[signature]_ Date: 10/14/04
1.0Qualified Preparer

Reviewed By: Paul R. Sunderland Date: 10/14/04
2.0Qualified Reviewer

Reviewed By: _[signature]_ Date: 10/15/04
3.0Regulatory Affairs

Approved By: Jerrie A. Coleman Date: 10/15/04
4.0Manager - EP

Abnormal Rad Levels / Radiolgical Effluent

General Difference for section:
Abnormal Rad Levels / Radiological Effluent section designation was changed from "A" (AU, AA, etc) to "R" (RU, RA, etc.).

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AU1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.	RU1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual for 60 Minutes or Longer.
Mode App.	All		All
Site specific	None		
Difference	• The Offsite Dose Calculation Manual (ODCM) gives the site-specific technical specifications for gaseous and liquid releases. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	RU1.1	VALID reading on any effluent monitor that is GREATER THAN two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. **Auxiliary Building** — **Action Value** R-13 Aux. Bldg. Vent Exhaust 2.61E+05 cpm R-14 Aux. Bldg. Vent Exhaust 2.62E+05 cpm **Reactor Building** R-12 Containment Gas 4.41E+05 cpm R-21 Containment Vent 4.40E+05 cpm **Liquid Radwaste** R-18 Waste Disposal System Liquid 2 X Calculated ODCM Setpoint
Site specific	▪ Monitors listed include ODCM associated effluent monitors at KNPP. Values identified are the results of calculations documented in C11620.		
Difference	• "2 X Calculated ODCM Setpoint" was used due to setpoint being calculated for each discharge permit and is dependent upon flow rate of discharge. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 60 minutes or longer: (site-specific list)	RU1.2	VALID reading on one or more of the following radiation monitors that is GREATER THAN the reading shown for 60 minutes or longer. **Liquid Radwaste** — **Action Value** R-16 Containment FCU SW Return — 3.38E+05 cpm R-19 S/G Blowdown Liquid — 2.58E+06 cpm R-20 Aux Bldg SW Return — 1.03E+05 cpm
Mode App.	All		All
Site specific	▪ Monitors listed include ODCM associated effluent monitors at KNPP. Values identified are the results of calculations documented in C11620.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times (site-specific technical specifications).	RU1.3	Confirmed sample analyses for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times the ODCM limit
Site specific	• The Offsite Dose Calculation Manual (ODCM) gives the site-specific technical specifications for gaseous and liquid releases.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	VALID reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal background sustained for 60 minutes or longer [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	N/A		
Difference	Deleted NEI 99-01 example EAL#4 because plant is not equipped with perimeter radiation monitoring.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
5	VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 60 minutes or longer [for sites having such capacity]	N/A	N/A
Site specific	N/A		
Difference	• Deleted NEI 99-01 example EAL#5 because plant is not equipped with automatic real-time dose assessment capabilities.		
Deviation	None		

RU1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. Added specific information pertaining to the radiation monitors listed and the basis for the values listed.	1. This information was added to explain the monitor values.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
3. Deleted reference to EAL#4	3. KNPP is not equipped with perimeter radiation monitors.
4. Deleted reference to EAL#5	4. KNPP is not equipped with automatic real-time dose assessment capabilities.

Difference	• Specific Rad monitors listed, giving the 2 times ODCM values to support EAL determination. • "2 X Calculated ODCM Setpoint" was used for R-18 due to setpoint being calculated for each discharge permit and is dependent upon flow rate of discharge. • EAL #4 and #5 Basis information is not applicable to KNPP and was deleted.
Deviations	None

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AU2	Unexpected Increase in Plant Radiation	RU2	Unexpected Rise in Plant Radiation
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	a. VALID (site-specific) indication of uncontrolled water level decrease in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water. AND b. Unplanned VALID (site-specific) Direct Area Radiation Monitor reading increases	RU2.1	VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by Spent Fuel Pool low water level alarm setpoint (3 ft 4 in below floor, SER 159/160) <u>OR</u> Visual observation <u>AND</u> Any UNPLANNED VALID Area Radiation Monitor reading rises as indicated by: • R2, Containment Area ALERT Alarm • R5, Fuel Handling Area ALERT Alarm • R10, New Fuel Pit Area ALERT Alarm
Site specific	• "Spent fuel pool low water level alarm setpoint" has been included because it is the only means of remote indication of decreasing water level. • "Visual observation" has been included to address the site-specific indication listed in the NEI 99-01 Rev. 4. basis (pg. 5-A-5). • R2, R5, and R10 radiation monitors are located in the containment and fuel handling areas.		
Difference	• The NEI term "Direct" has been deleted because the monitors used at KNPP to assess this threshold are commonly referred to as Area Radiation Monitors. Three monitors are provided in the EAL although the NEI indicates only one monitor. These monitors are used to classify the event because they are available in the control room to aid in classifying the event occurring in the SFP area and the use of these monitors allows for failure of a single monitor. • The phrase "any" has been added to indicate only one of the Area Radiation Monitors is necessary to meet the condition. • These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Unplanned VALID Direct Area Radiation Monitor readings increases by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.	RU2.2	Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
Site specific	None		
Difference	▪ "Any" has been added to clarify the fact that the threshold is met if there is a rise one or more of the indicated readings ▪ The NEI term "Direct" has been deleted because the monitors used at KNPP to assess this EAL are commonly referred to as Area Radiation Monitors. ▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

RU2 – Basis Justification	
<u>KNPP Specific Additions/Deletions</u>	**<u>Justification</u>**
Added the Spent Fuel Pool (SFP) low level alarm and the KNPP specific elevation information that corresponds to applicable levels in the SFP.	This information was added to clearly identify the instrumentation to be used with respect to this EAL.
Difference	One paragraph was added for clarification of site specific information.
Deviations	None

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AA1	Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer	RA1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that exceeds 200 times the Offsite Dose Calculation Manual for 15 minutes or Longer
Mode App.	All		All
Site specific	None		
Difference	• The Offsite Dose Calculation Manual (ODCM) gives the site-specific technical specifications for gaseous and liquid releases. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer	RA1.1	VALID reading on effluent monitor GREATER THAN 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer. **<u>Auxiliary Building</u>** — **<u>Action Value</u>** R-13 Aux. Bldg. Vent Exhaust — 2.61E+07 cpm R-14 Aux. Bldg. Vent Exhaust — 2.62E+07 cpm **<u>Reactor Building</u>** R-12 Containment Gas — 4.41E+07 cpm R-21 Containment Vent — 4.40E+07 cpm **<u>Liquid Radwaste</u>** R-18 Waste Disposal System Liquid — 2 X Calculated ODCM Setpoint
Site specific	▪ Monitors listed include routine ODCM effluent monitors at KNPP. Values identified are the results of calculations documented in C11620.		
Difference	• "2 X Calculated ODCM Setpoint" was used due to setpoint being calculated for each discharge permit and is dependent upon flow rate of discharge. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 15 minutes or longer: (site-specific list)	RA1.2	VALID reading on one or more of the following radiation monitors GREATER THAN the reading shown for 15 minutes or longer. **Liquid Radwaste** — **Action Value** R-16 Containment FCU SW Return — 3.38E+07 cpm R-19 S/G Blowdown Liquid — 2.58E+08 cpm R-20 Aux Bldg SW Return — 1.03E+07 cpm
Site specific	▪ Monitors listed include non-routine ODCM effluent monitors at KNPP. Values identified are the results of calculations documented in C11620.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times (site-specific technical specifications)	RA1.3	Confirmed sample analyses for gaseous or liquid release indicate concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ODCM limit
Site specific	• The ODCM (Offsite Dose Calculation Manual) gives the site-specific technical specifications for gaseous and liquid releases.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	VALID reading on perimeter radiation monitoring system greater than 10.0 mR/hr above normal background sustained for 15 minutes or longer [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	N/A		
Difference	• Deleted NEI 99-01 Example EAL #4 because the plant is not equipped with perimeter radiation monitoring.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
5	VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 15 minutes or longer [for sites having such capability]	N/A	N/A
Site specific	N/A		
Difference	• Deleted NEI 99-01 Example EAL #5 because the plant is not equipped with automatic real-time dose assessment capability.		
Deviation	None		

RA1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. Added specific information pertaining to the radiation monitors listed and the basis for the values listed.		1. This information was added to explain the monitor values.
2. Deleted reference to EAL#4		2. KNPP is not equipped with perimeter radiation monitors.
3. Deleted reference to EAL#5		3. KNPP is not equipped with automatic real-time dose assessment capabilities.
4. Development information contained in the NEI Basis was deleted.		4. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	• Information was added to explain the basis for the monitor values. • "2 X Calculated ODCM Setpoint" was used for R-18 due to setpoint being calculated for each discharge permit and is dependent upon flow rate of discharge. • EAL #4 and #5 Basis information is not applicable to KNPP and was deleted.	
Deviations	None	

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel	RA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	A VALID (site-specific) alarm or reading on one or more of the following radiation monitors: (site-specific monitors) Refuel Floor Area Radiation Monitor Fuel Handling Building Ventilation Monitor Refueling Bridge Area Radiation Monitor	RA2.1	A VALID radiation indication high alarm or reading on one or more of the following radiation monitor resulting from damage to irradiated fuel or loss of water level: • R-2, Containment Area • R-5, Fuel Handling Area • R-13 or R-14, Aux Bldg Vent Exhaust • R-11 or R-12, Containment Particulate/Gas Ventilation • R-21, Containment Vent
Site specific	• The listed radiation monitors represent the site-specific equivalents of Refuel Floor Area Radiation Monitor, Auxiliary Building Ventilation Monitor, Containment Ventilation Monitors, and Spent Fuel Pool Area Radiation Monitor. • "Radiation Indication High" is the appropriate terminology for KNPP.		
Difference	• Added the wording: "resulting from damage to irradiated fuel or loss of water level" to be consistent with NEI basis. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Water level less than (site-specific) feet for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering	RA2.2	Water level LESS THAN 50% Wide Range Refueling Water Level <u>OR</u> GREATER THAN 14 ft. below top of Spent Fuel Pool that will result in irradiated fuel uncovering.
Site specific	• Added site specific indications of 50% Wide Range Refueling Water Level and 14 ft. below top of Spent Fuel Pool. The values inserted are the maximum height the fuel could be during refueling. This would be bounding and conservative for all conditions. Since there is no remote level indication for corresponding Spent Fuel Pool Level, KNPP would rely on visual observation for the Spent Fuel Pool (a installed ruler on side).		
Difference	▪ The Fuel Transfer Canal was removed because there is no level indication and when fuel is located in the canal it is lined-up to the cavity and fuel pool. ▪ Replaced "and" with "<u>OR</u>" since either level lowering can result in irradiated fuel uncovery. ▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	▪ None		

RA2 – Basis Justification	
KNP Specific Additions/Deletions	<u>**Justification**</u>
Added the following statement - The top of each pool is at 649 ft 6 in. el. and the bottom is at 608 ft el. Fuel occupies the bottom 14 ft. [Ref. 4].	This statement was added to provide clear reference to site specific information.
Difference	Site specific information was added for clarity.
Deviations	None

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AA3	Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown	RA3	Release of Radioactive Material or Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	VALID (site-specific) radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: (Site-specific) list	RA3.1	VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (Rad Monitor R1) **OR** Central Alarm Station (Rad Monitor R1) **OR** Secondary Alarm Station (by survey)
Site specific	▪ Control Room, Central Alarm Station and Secondary Alarm Station are site specific areas requiring continuous occupancy to maintain plant safety functions. ▪ "by survey" has been added because automatic monitoring is not available in these areas.		
Difference	None		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	VALID (site-specific) radiation monitor readings GREATER THAN <site-specific> values in areas requiring infrequent access to maintain plant safety functions. (Site-specific) list	RA3.2	Any VALID radiation monitor readings GREATER THAN 6 R/hr in areas requiring infrequent access to maintain plant safety functions • Auxiliary Building • Safeguards Alley • Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel) • Screenhouse/Forebay • Relay Room • Safeguard Battery Room
Site specific	▪ 6 R/hr is the site specific value which based on the administrative dose limit for access to these areas. ▪ Site-specific list of areas requiring infrequent access in order to maintain plant safety functions.		
Difference	▪ "Any" has been added to clarify the fact that the threshold is met if there is a rise one or more of the indicated readings. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

RA3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. Added site specific information for areas requiring continuous occupancy.	1. This information was added to clarify the applicable areas.
2. Added a paragraph explaining the basis of the dose rate (6 Rem/hr) used for areas requiring infrequent access.	2. The basis of this number and specifically the estimated time that it was based on is important in understanding the intent of this EAL.
3. A paragraph explaining that in-plant radiation surveys and/or Area Radiation Monitor (ARM) readings should be used to identify VALID unplanned dose rate radiation monitor readings.	3. This was added to ensure clarity as to the meaning of the term - VALID unplanned dose rate radiation monitor readings
4. General KNPP plant specific information was added or replaced non-specific NEI information.	4. This information was added for explanation and clarification of site specifics.
5. Development information contained in the NEI Basis was deleted.	5. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Clarification and explanation of the EAL basis was added.
Deviations	None

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE or 500 mR Thyroid CDE for the Actual or Projected Duration of the Release	RS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release
Mode App.	All		All
Site specific	None		
Difference	• "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL #2 instead of EAL #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer: (site-specific list)	RS1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on any monitors listed that exceeds or is expected to exceed the reading shown for 15 minutes or longer. <u>Auxiliary Building</u> <u>Action Value</u> 01-07 Aux. Bldg. SPING Mid Range 1.00E+04 cpm 01-09 Aux. Bldg. SPING Hi Range 1.00E+01 cpm <u>Reactor Building</u> 02-07 Rx Bldg. Vent SPING Mid Range 2.00E+03 cpm <u>Main Steam Line (PORV)</u> R-31 'A' Steamline Lo Range 1.77E+02 mR/hr R-33 'B' Steamline Lo Range 1.77E+02 mR/hr <u>Main Steam Line (SG Safety)</u> R-31 'A' Steamline Lo Range 8.30E+01 mR/hr R-33 'B' Steamline Lo Range 8.30E+01 mR/hr
Site specific	• Monitors listed include all effluent monitors at KNPP. Values identified are the results of calculations documented in C11620.		
Difference	▪ The phrase "one or more" has been replaced by "any." The use of the term "any" is equivalent to "one or more", decreases EAL user reading burden and, thereby, increases the potential for timely and accurate emergency classifications ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE or 500 mR thyroid CDE at or beyond the site boundary	RS1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem Thyroid CDE at or beyond the site boundary.
Site specific	None		
Difference	• "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 100 mR/hr. [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	N/A		
Difference	Deleted NEI 99-01 Example EAL #3 because the plant is not equipped with perimeter radiation monitoring.		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at or beyond the site boundary	RS1.3	Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour, at or beyond the site boundary. **<u>OR</u>** Analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation, at or beyond the site boundary
Site specific	None		
Difference	▪ "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office ▪ Added "at or beyond the site boundary" prior to OR for clarity of the logic statement. ▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

RS1 – Basis Justification	
<u>KNPP Specific Additions/Deletions</u>	<u>Justification</u>
1. Added specific information pertaining to the radiation monitors listed and the basis for the values listed.	1. This information was added to explain the monitor values.
2. General KNPP plant specific information was added or replaced non-specific NEI information.	2. This information was added for explanation and clarification of site specifics.
3. Development information contained in the NEI Basis was deleted.	3. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Site specific basis information was added for clarification of radiation monitor values.
Deviations	None.

Abnormal Rad Levels / Radiolgical Effluent

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
AG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE or 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology	RG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology
Mode App.	All	All	
Site specific	None		
Difference	• "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Abnormal Rad Levels / Radiolgical Effluent

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL #2 instead of EAL #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on one or more of the following radiation monitors that exceeds or expected to exceed the reading shown for 15 minutes or longer: (site-specific list)	RG1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on any monitors listed that exceeds or is expected the reading shown for 15 minutes or longer. **Auxiliary Building** **Action Value** 01-07 Aux. Bldg. SPING Mid Range 1.00E+05 cpm 01-09 Aux. Bldg. SPING Hi Range 1.00E+02 cpm **Reactor Building** 02-07 Rx Bldg. Vent SPING Mid Range 2.00E+04 cpm 02-09 Rx Bldg. Vent SPING Hi Range 2.00E+01 cpm **Main Steam Line (PORV)** R-31 'A' Steamline Lo Range 1.77E+03 mR/hr R-32 'A' Steamline High Range 1.77E+00 R/hr R-33 'B' Steamline Lo Range 1.77E+03 mR/hr R-34 'B' Steamline High Range 1.77E+00 R/hr **Main Steam Line (SG Safety)** R-31 'A' Steamline Lo Range 8.30E+02 mR/hr R-33 'B' Steamline Lo Range 8.30E+02 mR/hr
Site	▪ Monitors listed include all effluent monitors at KNPP. Values identified are the results of		

specific	calculations documented in C11620.
Difference	▪ The phrase "one or more" has been replaced by "any." The use of the term "any" is equivalent to "one or more", decreases EAL user reading burden and, thereby, increases the potential for timely and accurate emergency classifications ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL
Deviation	None

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE or 5000 mR thyroid CDE at or beyond the site boundary	RG1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary
Site specific	None		
Difference	• "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 1000 mR/hr. [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	N/A		
Difference	Deleted NEI 99-01 Example EAL #3 because the plant is not equipped with perimeter radiation monitoring		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at or beyond site boundary.	RG1.3	Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour, at or beyond site boundary. **OR** Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary
Site specific	None		
Difference	▪ "mRem" was used instead of "mR" based on a request for use of this nomenclature from the Wisconsin State Emergency Management Office ▪ Added "at or beyond the site boundary" prior to OR for clarity of the logic statement. ▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

RG1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. Added specific information pertaining to the radiation monitors listed and the basis for the values listed.	1. This information was added to explain the monitor values.
2. General KNPP plant specific information was added or replaced non-specific NEI information.	2. This information was added for explanation and clarification of site specifics.
3. Development information contained in the NEI Basis was deleted.	3. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Site specific basis information was added for clarification of radiation monitor values..
Deviations	None.

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU1	RCS Leakage	CU1	RCS Leakage
Mode App.	Cold Shutdown		Cold Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Unidentified or pressure boundary leakage greater than 10 gpm	CU1.1	Unidentified or pressure boundary leakage GREATER THAN 10 gpm
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Identified leakage greater than 25 gpm	CU1.2	Identified leakage GREATER THAN 25 gpm
Site specific	None		
Difference	None		
Deviation	None		

CU1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added: Positive indications in the Control Room of Reactor Coolant System (RCS) leakage to the containment are provided by equipment that monitors: • Charging/Letdown flow mismatch • Containment air activity • Containment atmosphere humidity • Runoff from the air recirculation units and containment floor drains to Containment Sump A In-leakage	Supplemental information was added to provide guidance as to positive indications of RCS leakage.

Difference	Added site specific supplemental information to aid in EAL determination.
Deviations	None

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU2	UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the RPV	CU2	UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel
Mode App.	Refueling		Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	UNPLANNED RCS level decrease below the RPV flange for ≥ 15 minutes	CU2.1	UNPLANNED RCS level lowering below the Reactor Vessel flange (21.5%) for GREATER THAN OR EQUAL TO 15 minutes.
Site specific	▪ 21.5% is the site specific value from the reactor vessel level indicator at the flange per N-RC-36E, Draining the Reactor Coolant System		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	a. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase **AND** b. RPV level cannot be monitored	CU2.2	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A, Containment Sump C or Liquid Waste Disposal System level rise **AND** Reactor Vessel level cannot be monitored
Site specific	▪ "Containment Sump A, Containment Sump C, and Liquid Waste Disposal System" are the site specific indications for loss of reactor vessel level inventory		
Difference	None		
Deviation	None		

CU2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
Added site specific information on sumps, tanks and reactor vessel level indication to the basis where appropriate. Unnecessary NEI EAL development information was deleted.		The site specific information will make EAL determination easier and less time consuming. Also, removed unnecessary information the end user would not need.
Difference	Added site specific information to the basis.	
Deviations	None	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU3	Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes	CU3	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	a. Loss of power to (site-specific) transformers for greater than 15 minutes. **AND** b. At least (site-specific) emergency generators are supplying power to emergency busses	CU3.1	Loss of all offsite power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes **AND** At least one emergency diesel generator is supplying power to Bus 5 or Bus 6
Site specific	▪ "all offsite power to Bus 5 and Bus 6" – has been used in place of "transformers" to focus the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses ▪ Safety related Bus 5 and Bus 6 identifies the site specific ESF busses ▪ "One diesel generator" is the site specific number needed to power one ESF bus.		
Difference	• The NEI example EAL condition "Loss of power to (site-specific) transformers for greater than 15 minutes" has been changed to "Loss of offsite power to Bus 5 and Bus 6 for GREATER THAN 15 min." The KNPP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses. This simplifies the EAL wording and meets the intent of the NEI IC. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

CU3 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. Added site specific system description information on offsite power sources for the purpose of classification under this EAL.		1. This information was added to clarify the conditions under which this EAL would apply.
2. Removed the companion unit paragraph.		2. KNPP is a single unit site.
Difference	Added site specific information on offsite power sources.	
Deviations	None	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU4	UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV	CU4	UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit	CU4.1	An UNPLANNED event results in RCS temperature GREATER THAN 200°F
Site specific	• LESS THAN OR EQUAL to 200 °F" is the Technical Specification cold shutdown temperature limit.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Loss of all RCS temperature and RPV level indication for > 15 minutes	CU4.2	Loss of all RCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.
Site specific	None		
Difference	None		
Deviation	None		

CU4 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. Deleted 30 minute time (and associated information) in discussing escalation to Alert.		1. The 30 minute time statement was incorrect and a single number was not available for KNPP.
2. Added site specific information describing instrumentation on which Reactor Vessel water level is normally monitored.		2. This information was added to clarify indications available to persons making decisions on this EAL classification.
Difference	• Deleted escalation 30 minute statement. • Added site specific information.	
Deviations	N/A	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU5	Fuel Clad Degradation	CU5	Fuel Clad Degradation
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	(Site-specific) radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits	CU5.1	RCS Letdown Line (R-9) radiation monitor GREATER 2000 mR/hr indicating fuel clad degradation
Site specific	• RCS Letdown Line (R-9) is the site specific monitor designated to indicate fuel clad failure. • 2000 mR/hr is equal to the Technical Specification allowable limits		
Difference	• Reworded NEI EAL for readability • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	(Site-specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits	CU5.2	Coolant sample activity GREATER THAN <u>ANY</u> of the following indicating fuel clad degradation: • 1.0 µCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval • 60 µCi/gram dose equivalent Iodine-131. • 91/Ē µCi/cc gross radioactivity

Cold Shutdown / Refueling System Malfunction

Site specific	▪ The following are KNPP Technical Specification Limits for coolant sample activity: • 1.0 µCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval • 60 µCi/gram dose equivalent Iodine-131. • 91/Ē µCi/cc gross radioactivity
Difference	▪ "technical specification allowable limits" – was deleted as it duplicates the site specific tech spec value already listed in the EAL ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL
Deviation	None

CU5 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. Added site-specific information on R-9 setpoint for SU4.1	1. This information was added to clarify the information used to determine the R-9 setpoint
Difference	Added KNPP site specific information
Deviations	N/A

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	CU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of all (site-specific list) onsite communications capability affecting the ability to perform routine operations	CU6.1	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations
Site specific	• Table C-1 lists site specific equipment used for onsite communications		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Loss of all (site-specific list) offsite communications capability	CU6.2	Loss of all Table C-2 offsite communications capability
Site specific	• Table C-2 lists site specific equipment used for offsite communications		
Difference	None		
Deviation	None		

CU6 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Explanation of Tables C1 and C2 was added to the basis.	Tables C1 and C2 were added to clearly delineate between onsite communication equipment and offsite communication equipment.

Difference	Explanation of Tables C1 and C2 were added.
Deviations	None

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU7	UNPLANNED Loss of Required DC Power for Greater than 15 Minutes	CU7	UNPLANNED loss of Required DC power for GREATER THAN 15 minutes
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	a. UNPLANNED Loss of Vital DC power to required DC busses based on (site-specific) bus voltage indications. **AND** b. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.	CU7.1	UNPLANNED Loss of Vital DC power based on LESS THAN 105 VDC on Train A <u>AND</u> Train B Safeguards DC Distribution System. <u>**AND**</u> Failure to restore power to at least one required Train of the Safeguards DC Distribution System within 15 minutes from the time of loss.
Site specific	▪ LESS THAN 105 VDC on Train A <u>AND</u> Train B Safeguards DC Distribution System is the KNPP design voltage and specific DC buses.		
Difference	▪ The design of the KNPP 125v DC Distribution System is such that a loss of different combinations of distribution panels and buses could constitute a loss of DC power to a Train. These combinations that would cause a loss of DC power are covered in the basis for this EAL. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

CU7 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
Added site specific information on safeguards DC train and DC distribution panels. Unnecessary NEI EAL development information was deleted.		This information was added for explanation and clarification of the specific equipment involved when making determinations under this EAL. Also, removed unnecessary information the end user would not need.
Difference	Added site specific information.	
Deviations	None	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CU8	Inadvertent Criticality	CU8	Inadvertent Criticality
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	An UNPLANNED extended positive period observed on nuclear instrumentation	N/A	N/A
Site specific	N/A		
Difference	Not applicable, BWR NEI EAL.		
Deviation	N/A		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	CU8.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation
Site specific	None		
Difference	None		
Deviation	None		

CU8 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
Added the sentence - This condition can be identified using startup rate meters (NI-31D/32D - Source Range Startup Rate).		This was added to provide site specific information to the basis.
Difference	Added site specific information.	
Deviations	None.	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CA1	Loss of RCS Inventory	CA1	Loss of RCS Inventory
Mode App.	Cold Shutdown		Cold Shutdown
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of RCS inventory as indicated by RPV level less than {site-specific level}. (low-low ECCS actuation setpoint) (BWR) (bottom ID of the RCS loop) (PWR)	CA1.1	Loss of RCS inventory as indicated by one or more of the following: • Wide Range Refueling Water Level LESS THAN 8% • RVLIS at 0% • Sightglass water level LESS THAN 252 in
Site specific	▪ 8% Wide Range Refueling Water Level, RVLIS 0% and Sightglass at 252 inches are the site specific indications that water level is at the bottom of the RCS hot leg. per N-RC-36E, Draining the Reactor Coolant System		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	a. Loss of RCS inventory as indicated by unexplained {site-specific} sump and tank level increase **AND** b. RCS level cannot be monitored for > 15 minutes	CA1.2	Loss of RCS inventory as indicated by unexplained level rise in any of the following: • Containment Sump A • Containment Sump C • Liquid Waste Disposal System **AND** RCS level cannot be monitored for GREATER THAN 15 min.
Site specific	▪ "Containment Sump A, Containment Sump C, and Liquid Waste Disposal System" are the site specific indications for loss of reactor vessel level inventory		

Difference	None
Deviation	None

CA1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added site specific information on Reactor water level monitoring indications, corresponding elevations, and applicable sumps and tanks.	This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.
Deviations	None

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CA2	Loss of RPV Inventory with Irradiated Fuel in the RPV	CA2	Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel
Mode App.	Refueling		Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of RPV inventory as indicated by RPV level less than {site-specific level}. (bottom ID of the RCS Loop)	CA2.1	Loss of Reactor Vessel inventory as indicated by Wide Range Refueling Water Level LESS THAN 8% (0% RVLIS, 252 in. sightglass)
Site specific	▪ 8% is the site specific value from the wide range refueling water level indicators (L9053A and L9054A) at the bottom of the RCS hot leg. per N-RC-36E, Draining the Reactor Coolant System		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	a. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase **<u>AND</u>** b. RPV level cannot be monitored for > 15 minutes	CA2.2	Loss of Reactor Vessel inventory as indicated by unexplained level rise in any of the following: • Containment Sump A • Containment Sump C • Liquid Waste Disposal System **<u>AND</u>** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes.
Site specific	▪ "Containment Sump A, Containment Sump C, and Liquid Waste Disposal System" are the site specific indications for loss of reactor vessel level inventory		
Difference	None		
Deviation	None		

CA2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
Added site specific information on Reactor water level monitoring indications, corresponding elevations, and applicable sumps and tanks.		This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.	
Deviations	None	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CA3	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses	CA3	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses
Mode App.	Cold Shutdown, Refueling, Defueled		Cold Shutdown, Refueling, Defueled
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	a. Loss of power to (site-specific) transformers. **AND** b. Failure of (site-specific) emergency generators to supply power to emergency busses. **AND** c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	CA3.1	Loss of ALL power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes
Site specific	▪ "all offsite power to Bus 5 and Bus 6" – has been used in place of "transformers" to focus the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses. Safety related Bus 5 and Bus 6 identifies the site specific ESF buses. ▪ Emergency diesel generators A and B are site specific generators that would supply the ESF buses.		

Difference	▪ The NEI example EAL condition "Loss of power to (site-specific) transformers" has been changed to "Loss of offsite power to Bus 5 and Bus 6." The KNPP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses. This simplifies the EAL wording and concisely meets the intent of the NEI IC. ▪ KNPP EAL was reformatted from three NEI conditions to an encompassing one condition. This was done to simplify the classification and the economy of words. Stating that "ALL power" is lost to Bus 5 and Bus 6 covers the first two NEI conditions (loss of power to the transformers and failure of the emergency diesel generators). "For GREATER THAN 15 minutes" is equivalent to the third NEI condition (time to restore power to at least one emergency bus). ▪ These changes are not deviations because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL
Deviation	None

CA3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added site specific information on KNPP offsite and onsite AC power to the 4160 VAC ESF buses.	This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.
Deviations	None.

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CA4	Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV	CA4	Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	With CONTAINMENT CLOSURE <u>and</u> RCS integrity <u>not</u> established an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit.	CA4.1	With CONTAINMENT CLOSURE NOT established **<u>AND</u>** RCS integrity NOT established An UNPLANNED event results in RCS temperature GREATER THAN 200°F
Site specific	▪ "GREATER THAN 200°F" - represents the Tech Spec cold shutdown temperature limit (Less than or equal to 200°F).		
Difference	▪ Added, "NOT established" for containment closure for human factors considerations. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Cold Shutdown / Refueling System Malfunction

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for greater than 20 minutes[1]. [1]Note: If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.	CA4.2	With CONTAINMENT CLOSURE established **AND** RCS integrity NOT established OR Wide Range Refueling Water Level LESS THAN OR EQUAL TO 17.0% An UNPLANNED event results in RCS temperature GREATER THAN 200°F for GREATER THAN 20 minutes* *NOTE: If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.
Site specific	▪ "GREATER THAN 200°F" - represents the Tech Spec cold shutdown temperature limit (Less than or equal to 200°F). ▪ RHR is the KNPP RCS heat removal system below 200°F.		
Difference	▪ Inserted 17.0% Wide Range Refueling Water Level for the KNPP setpoint for reduced inventory ▪ Inserted note contained in NEI document for human factors consideration. ▪ This change is not a deviation because it dose not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Cold Shutdown / Refueling System Malfunction

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for greater than 60 minutes[1] or results in an RCS pressure increase of greater than {site-specific} psig. [1]Note: If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.	CA4.3	An UNPLANNED event results in RCS temperature GREATER THAN 200°F for GREATER THAN 60 minutes* **OR** Results in an RCS pressure increase of GREATER THAN 10 psig. ***NOTE:** If RHR system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.
Site specific	▪ "GREATER THAN 200°F" - represents the Tech Spec cold shutdown temperature limit (Less than or equal to 200°F). ▪ 10 psig – KNPP pressure instrumentation can read less than 10 psig.		
Difference	▪ Inserted note contained in NEI document for human factors consideration. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

CA4 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added KNPP site specific information on pressure, level and temperature indication. Unnecessary NEI EAL development information was deleted.	This information was added for explanation and clarification of site specifics. Also, removed unnecessary information the end user would not need.
Difference	Added KNPP site specific information.
Deviations	None

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CS1	Loss of RPV Inventory Affecting Core Decay Heat Removal Capability	CS1	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability
Mode App.	Cold Shutdown,		Cold Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	With CONTAINMENT CLOSURE <u>not</u> established: a. RPV inventory as indicated by RPV level less than {site-specific level} (6" below the low-low ECCS actuation setpoint) (BWR) (6" below the bottom ID of the RCS loop) (PWR) **<u>OR</u>** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase	CS1.1	With CONTAINMENT CLOSURE <u>NOT</u> established: a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level LESS THAN 7% **<u>OR</u>** b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained level rise in any of the following: • Containment Sump A • Containment Sump C • Liquid Waste Disposal System
Site specific	▪ Wide Range Refueling Water Level LESS THAN 7% is KNPP setpoint for 6" below ID of Hot Leg ▪ "Containment Sump A, Containment Sump C, and Liquid Waste Disposal System" are the site specific indications for loss of reactor vessel level inventory		
Difference	None		
Deviation	None.		

Cold Shutdown / Refueling System Malfunction

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	With CONTAINMENT CLOSURE established a. RPV inventory as indicated by RPV level less than TOAF **OR** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by either: • Unexplained {site-specific} sump and tank level increase • Erratic Source Range Monitor Indication	CS1.2	With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level EQUAL TO 0% **OR** b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either: • Unexplained Containment Sump A, Containment Sump C, OR Liquid Waste Disposal System level rise • Erratic Source Range Monitor Indication
Site specific	• Wide Range Refueling Water Level EQUAL TO 0% Refueling Level is KNPP's setpoint for top of active fuel. • "Containment Sump A, Containment Sump C, and Liquid Waste Disposal System" are the site specific indications for loss of reactor vessel level inventory		
Difference	None		
Deviation	None		

CS1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added KNPP site specific information and explanations of actions required for raising sump levels and lowering RCS levels. Unnecessary NEI EAL development information was deleted.	This information was added for explanation and clarification of site specifics. Also, removed unnecessary information the end user would not need.
Difference	Added KNPP site specific information.
Deviations	None

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CS2	Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV	CS2	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel
Mode App.	Refueling		Refueling
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	With CONTAINMENT CLOSURE <u>not</u> established: a. RPV inventory as indicated by RPV level less than {site-specific level} (6" below the low-low ECCS actuation setpoint) (BWR) (6" below the bottom ID of the RCS loop) (PWR) **<u>OR</u>** b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications	CS2.1	With CONTAINMENT CLOSURE <u>NOT</u> established: a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level LESS THAN 7% **<u>OR</u>** b. Reactor Vessel level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr • Erratic Source Range Monitor Indication
Site specific	▪ 100 mRem/hr is the alarm setpoint for R-2. ▪ Wide Range Refueling Water Level LESS THAN 7% is KNPP setpoint for 6" below ID of Hot Leg ▪ R-2 is the Containment Area Radiation Monitor.		
Difference	• R-2 is used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale. Therefore the alarm setpoint of R-2 was selected to indicate a rise in containment radiation		

	resulting from the conditions of this EAL. • This change is not a deviation because it dose not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL
Deviation	None

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	With CONTAINMENT CLOSURE established: a. RPV inventory as indicated by RPV level less than TOAF **OR** b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications	CS2.2	With CONTAINMENT CLOSURE established a. Reactor Vessel inventory as indicated by Wide Range Refueling Water Level EQUAL TO 0% **OR** b. Reactor Vessel level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr • Erratic Source Range Monitor Indication
Site specific	▪ 100 mRem/hr is the alarm setpoint for R-2. ▪ R-2 is the Containment Area Radiation Monitor		
Difference	• Wide Range Refueling Water Level EQUAL TO 0% Refueling Level is KNPP's setpoint for top of active fuel (TOAF). • R-2 is used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale. Therefore the alarm setpoint of R-2 was selected to indicate a rise in containment radiation resulting from the conditions of this EAL. • This change is not a deviation because it dose not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

CS2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. Deleted the reference to a 30 minute time limit.		1. The 30 minute time limit is not used in the any of the NEI CS2 EAL statements.
2. Added site specific information on instrumentation and elevations at KNPP. Unnecessary NEI EAL development information was deleted.		2. This information was added for explanation and clarification of site specifics. Also, removed unnecessary information the end user would not need.
Difference	• Deleted reference to a 30 minute time limit and added KNPP site specific information. • Changed the water level effect on containment radiation to R-2 being used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale.	
Deviations	None.	

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
CG1	Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV	CG1	Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged and Irradiated Fuel in the Reactor Vessel
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Site specific	None.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	1. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase 2. RPV Level: a. less than TOAF for > 30 minutes **OR** b. cannot be monitored with Indication of core uncovery for > 30 minutes as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications 3. {Site-specific} indication of CONTAINMENT challenged as indicated by one or more of the following: • Explosive mixture inside containment • Pressure above {site-specific} value • CONTAINMENT CLOSURE	CG1.1	Loss of Reactor Vessel inventory as indicated by unexplained level rise in Containment Sump A, Containment Sump C <u>OR</u> Liquid Waste Disposal System **AND** Reactor Vessel Level (a or b): a. EQUAL TO 0% Wide Range Refueling Water Level for GREATER THAN 30 minutes **OR** b. Cannot be monitored with Indication of core uncovery for GREATER THAN 30 minutes as evidenced by one or more of the following: • Containment Area Radiation Monitor (R-2) reading GREATER THAN 100 mRem/hr • Erratic Source Range Monitor Indication **AND** Indication of CONTAINMENT challenged as indicated by one or more of the following: • GREATER THAN OR EQUAL TO 6%

	<u>not</u> established • Secondary Containment radiation monitors above {site-specific} value (BWR only)		hydrogen in containment • CONTAINMENT CLOSURE <u>NOT</u> established • CONTAINMENT pressure above: • 46 psig <u>IF</u> Containment Integrity or Reduced Inventory Containment Integrity is established <u>OR</u> • 46 psig if Refueling Containment Integrity is established with <u>no</u> loop seal penetrations installed at Penetration 42N or 43N. <u>OR</u> • 0.6 psig if Refueling Containment Integrity is established with loop seal penetration installed at either Penetration 42N or 43N.
Site specific	▪ "Containment Sump A, Containment Sump C and Liquid Waste Disposal System are the site specific indications for loss of reactor vessel level inventory at 0% ▪ "EQUAL TO 0% Wide Range Refueling Water Level" is equal to TOAF ▪ 100 mRem/hr is the alarm setpoint for R-2. ▪ R-2 is the Containment Area Radiation Monitor • Containment pressure ≥ 46 psig <u>IF</u> Containment Integrity or Reduced Inventory Containment Integrity is established • Containment pressure 46 psig if Refueling Containment Integrity is established with no loop seal penetrations installed • Containment pressure 0.6 psig if Refueling Containment Integrity is established with loop seal penetrations installed ▪ Hydrogen concentration in containment ≥ 6% ▪ "Secondary Containment radiation monitors above {site-specific} value (BWR only)" is applicable only to BWRs		
Difference	• Due to NEI Examples EAL 1, 2 & 3 being "and" logic, combined into a single EAL. • R-2 is used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale. Therefore the alarm setpoint of R-2 was selected to indicate a rise in containment radiation resulting from the conditions of this EAL. • These change is not a deviation because it dose not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

CG1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
Deleted the two paragraphs referring to evaluating sump and tank levels against other potential sources of leakage and replaced with KNPP specific information on evaluating sump and tank levels.		This information was added for explanation and clarification of site specifics.
Difference	• Replaced NEI generic wording with KNPP site specific information. • Changed the water level effect on containment radiation to R-2 being used instead of the high range containment monitors because if a small amount of fuel was uncovered, the location of the high range monitors would preclude them reading on scale.	
Deviations	None	

Fission Product Barrier Degradation

General Difference for section:
BWR Fission Product Barrier is not applicable to KNPP and was deleted.

Format change to Emergency Action Levels on the top of the tables, EALs were reordered so from left to right the EALs are listed GE, SAE, Alert and UE. This corresponds to the KNPP wallchart.

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FU1	ANY Loss or ANY Potential Loss of Containment	FU1	ANY Loss or ANY Potential Loss of Containment
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	ANY Loss or ANY Potential Loss of Containment	FU1.1	ANY Loss or ANY Potential Loss of Containment (Table F-1)
Site specific	Table F-1 added, KNPP designation in EALs for Fission Product Barrier Reference Table.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	FA1.1	ANY loss or ANY Potential Loss of EITHER Fuel Clad OR RCS (Table F-1)
Site specific	Table F-1 added, KNPP designation in EALs for Fission Product Barrier Reference Table.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FS1	Loss or Potential Loss of ANY Two Barriers	FS1	Loss or Potential Loss of ANY Two barriers
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	Non		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss or Potential Loss of ANY Two Barriers	FS1.1	Loss or Potential Loss of ANY two barriers (Table F-1)
Site specific	Table F-1 added, KNPP designation in EALs for Fission Product Barrier Reference Table.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FG1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier	FG1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier	FG1.1	Loss of ANY two barriers AND Loss or Potential Loss of third barrier (Table F-1)
Site specific	Table F-1 added, KNPP designation in EALs for Fission Product Barrier Reference Table.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 1	**Critical Safety Function Status** Core-Cooling Red	FC Loss 1	**Critical Safety Function Status** Core-Cooling Red
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 2	**Primary Coolant Activity Level** Coolant Activity GREATER THAN (site-specific) Value	FC Loss 2	**Primary Coolant Activity Level** Coolant Activity GREATER THAN 300 μCi/gm I-131 equivalent
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	300 μCi/gm I-131 equivalent: This value was taken from NEI 99-01 Rev. 4 basis for fuel clad barrier primary coolant activity (pg 5-F-14).		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 3	<u>Core Exit Thermocouple Readings</u> GREATER THAN (site-specific) degree F	FC Loss 3	<u>Core Exit Thermocouple Readings</u> GREATER THAN OR EQUAL TO 1200°F
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	1200°F: This temperature is given in Critical Safety Function Status Trees.		
Difference	None		
Deviation	The NEI phrase "greater than" has been changed to "GREATER THAN OR EQUAL TO" so that the EAL threshold agrees with the level specified in CSF-ST Critical Safety Function Status Trees. Using "greater than or equal to" is conservative deviation. Therefore, this deviation does not decrease the effectiveness of the NEI EAL and does not adversely effects the health/safety of the public.		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 4	<u>Reactor Vessel Water Level</u> Not Applicable	N/A	<u>Reactor Vessel Water Level</u> Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 5	<u>Containment Radiation Monitoring</u> Containment rad monitor reading GREATER THAN (site-specific) R/hr	FC Loss 5	<u>Containment Radiation Monitoring</u> Containment rad monitor (R-40/41) reading GREATER THAN 1000 R/hr
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	1000 R/hr is the site-specific containment rad monitor reading that has been calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere. Refer to Calc. C11617 The high range containment radiation monitors are R-40 and R-41 at KNPP.		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 6	<u>Other (Site-Specific) Indications</u> (Site-specific) as applicable	N/A	No other applicable site-specific indications identified.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC Loss 7	<u>**Emergency Director Judgment**</u> Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier	FC Loss 6	<u>**Emergency Director Judgment**</u> Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 1	**Critical Safety Function Status** Core Cooling-Orange OR Heat Sink-Red	FC P-Loss 1	**Critical Safety Function Status** Core Cooling-ORANGE **OR** Heat Sink-RED
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 2	**Primary Coolant Activity Level** Not Applicable	N/A	**Primary Coolant Activity Level** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 3	<u>Core Exit Thermocouple Readings</u> GREATER THAN (site-specific) degree F	FC P-Loss 3	<u>Core Exit Thermocouple Readings</u> GREATER THAN OR EQUAL TO 700°F
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	700°F - KNPP Core Cooling-Orange path, which is entered if CET readings are equal to or greater than 700°F		
Difference	None		
Deviation	Used "GREATER THAN OR EQUAL TO" wording to be consistent with the CSFST definition of the KNPP Core Cooling-Orange path, which is entered if CET readings are equal to or greater than 700°F. Using "greater than or equal to" is conservative deviation. Therefore, this deviation does not decrease the effectiveness of the NEI EAL and does not adversely effects the health/safety of the public.		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 4	<u>Reactor Vessel Water Level</u> Level LESS than (site-specific) value	FC P-Loss 4	<u>Reactor Vessel Water Level</u> RVLIS void fraction rising <u>AND</u> At least one RCP running <u>AND</u> RCS Subcooling LESS THAN 30 °F [65°F].
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	▪ RVLIS void fraction rising and at least one RCP running and RCS Subcooling less than 30 °F [65°F] indicates a potential loss of the fuel clad barrier. This combination is an indication of inadequate coolant inventory and is used in the Core Cooling-ORANGE path and indicates subcooling has been lost and that some fuel cladding damage may occur. The value in brackets is for adverse containment conditions.		
Difference	• KNPP RVLIS does not extend down to the top of active fuel. It only measures as low as the bottom of the hot legs. The NEI methodology states that the Core Cooling orange path should define the potential loss EAL. This is defined in two ways. If RCS subcooling is less than 30°F with one or more reactor coolant pumps running, then RVLIS void fraction rising defines the Core Cooling orange path. If RCS subcooling is less than 30°F with no reactor coolant pumps running, then core exit thermocouples above 700°F defines the Core Cooling orange path. Since the fuel clad potential loss for core exit thermocouples uses 700°F for its potential loss, the latter path is not used for the reactor vessel water level potential loss.		
Deviation	None		

Fission Product Barrier Degradation

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 5	**Containment Radiation Monitoring** Not Applicable	N/A	**Containment Radiation Monitoring** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 6	**Other (Site-Specific) Indications** (Site-specific) as applicable	N/A	No other applicable site-specific indications identified.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	Not Applicable		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
FC P-Loss 7	**<u>Emergency Director Judgment</u>** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier	FC P-Loss 6	**<u>Emergency Director Judgment</u>** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 1	**Critical Safety Function Status** Not Applicable	N/A	**Critical Safety Function Status** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 2	**RCS Leak Rate** GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling	RCS Loss 2	**RCS Leak Rate** GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling: ▪ LESS THAN 20°F if the reactor is critical ▪ LESS THAN 30°F if the reactor is sub-critical
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	▪ Core exit thermocouples LESS THAN 20°F: The subcooling margin threshold while critical is based on the minimum subcooling allowed for normal operation defined in Operating Procedure A-RC-36-D. ▪ Core exit thermocouples LESS THAN 30°F: The subcooling margin threshold while subcritical is the level specified in Critical Safety Function Status Trees. EOPs define this value as a loss of RCS subcooling.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 3	**SG Tube Rupture** SGTR that results in an ECCS (SI) Actuation	RCS Loss 3	**SG Tube Rupture** SGTR that results in an ECCS (SI) Actuation
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 4	**Containment Radiation Monitoring** Containment rad monitor reading GREATER THAN (site-specific) R/hr	RCS Loss 4	**Containment Radiation Monitoring** Containment rad monitor (R-40/41) reading GREATER THAN 30 R/hr
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	30 R/hr is the site-specific containment rad monitor reading that has been calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere. Refer to Calc. C11617 The high range containment radiation monitors are R-40 and R-41 at KNPP.		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 5	**Other (Site-Specific) Indications** (Site-specific) as applicable	N/A	No other applicable site-specific indications identified.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS Loss 6	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier	RCS Loss 5	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS P-Loss 1	<u>Critical Safety Function Status</u> RCS Integrity-Red OR Heat Sink-Red	RCS P-Loss 1	<u>Critical Safety Function Status</u> RCS Integrity-RED <u>**OR**</u> Heat Sink-RED
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
RCS P-Loss 2	<u>RCS Leak Rate</u> Unisolable leak exceeding the capacity of one charging pump in the normal charging mode	RCS P-Loss 2	<u>RCS Leak Rate</u> Unisolable leak GREATER THAN 60 gpm, the capacity of one charging pump in the normal charging mode.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	60 gpm: This threshold is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System, which is considered as one charging pump discharging to the charging header. The need for a second charging pump would be indicative of a substantial RCS leak.		
Difference	▪ Added "60 gpm" for clarification. ▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
RCS P-Loss 3	**SG Tube Rupture** Not Applicable	N/A	**SG Tube Rupture** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
RCS P-Loss 4	**Containment Radiation Monitoring** Not Applicable	N/A	**Containment Radiation Monitoring** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
RCS P-Loss 5	<u>Other (Site-Specific) Indications</u> (Site-specific) as applicable	N/A	No other applicable site-specific indications identified.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
RCS P-Loss 6	<u>Emergency Director Judgment</u> Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier	RCS P-Loss 5	<u>Emergency Director Judgment</u> Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 1	<u>**Critical Safety Function Status**</u> Not Applicable	N/A	<u>**Critical Safety Function Status**</u> Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 2	<u>**Containment Pressure**</u> Rapid unexplained decrease following initial increase OR Containment pressure or sump level response not consistent with LOCA conditions	CMT Loss 2	<u>**Containment Pressure**</u> Rapid unexplained decrease following initial rise <u>**OR**</u> Containment pressure or sump level response not consistent with LOCA conditions
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 3	<u>Core Exit Thermocouple Readings</u> Not applicable	N/A	<u>Core Exit Thermocouple Readings</u> Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 4	<u>SG Secondary Side Release with Primary-to-Secondary Leakage</u> RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	CMT Loss 4	<u>SG Secondary Side Release with Primary-to-Secondary Leakage</u> RUPTURED S/G is also FAULTED outside of containment <u>**OR**</u> Primary-to-Secondary leakrate GREATER THAN 10 gpm with nonisolable steam release from affected S/G to the environment
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 5	**CNMT Isolation Valves Status After CNMT Isolation** Valve(s) not closed AND downstream pathway to the environment exists	CMT Loss 5	**CNMT Isolation Valves Status After CNMT Isolation** Containment isolation valve(s) not closed **AND** Downstream pathway to the environment exists, after containment isolation
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	Additional wording added for clarity of statement.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 6	**Significant Radioactive Inventory in Containment** Not Applicable	N/A	**Significant Radioactive Inventory in Containment** Not applicable per NEI 99-01 Revision 4.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 7	**Other (Site-Specific) Indications** (Site-specific) as applicable	N/A	No other applicable site-specific indications identified.
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT Loss 8	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	CMT Loss 7	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 1	**Critical Safety Function Status** Containment-Red	CMT P-Loss 1	**Critical Safety Function Status** Containment-Red
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 2	**Containment Pressure** (Site-specific) PSIG and increasing OR Explosive mixture exists OR Pressure greater than containment depressurization actuation setpoint with less than one full train of depressurization equipment operating	CMT P-Loss 2	**Containment Pressure** 46 psig and rising **OR** Hydrogen concentration GREATER THAN OR EQUAL TO 6% **OR** Containment pressure GREATER THAN 23 psig with LESS THAN one full train of depressurization equipment operating
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	▪ 46 psig is the design pressure of containment. ▪ Hydrogen concentration GREATER THAN OR EQUAL TO 6% - Hydrogen concentration is specified in the FR-C.1, Response to Inadequate Core Cooling, for explosive mixture. ▪ GREATER THAN OR EQUAL TO 23 psig - 23 psig is the pressure at which the equipment should have actuated and began performing its function.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 3	<u>Core Exit Thermocouple Readings</u> Core exit thermocouples in excess of 1200 degrees and restoration procedures not effective within 15 minutes; or, core exit thermocouples in excess of 700 degrees with reactor vessel level below top of active fuel and restoration procedures not effective within 15 minutes	CMT P-Loss 3	<u>Core Exit Thermocouple Readings</u> Core exit thermocouples GREATER THAN OR EQUAL TO 1200°F <u>and</u> restoration procedures not effective within 15 minutes <u>OR</u> Core exit thermocouples GREATER THAN OR EQUAL TO 700°F with RCPs <u>NOT</u> running AND restoration procedures not effective within 15 minutes <u>OR</u> RVLIS void fraction rising with at least one RCP running <u>and</u> RCS Subcooling LESS THAN 30 °F [65°F] <u>and</u> restoration procedures not effective within 15 minutes
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	▪ "GREATER THAN OR EQUAL TO 1200°F" - site specific value from Core Cooling CSF Status Tree Red Path ▪ "Core exit thermocouples GREATER THAN OR EQUAL TO 700°F with RCPs <u>NOT</u> running" - site specific value from Core Cooling CSF Status Tree Orange Path ▪ "RVLIS void fraction rising with at least one RCP running <u>and</u> RCS Subcooling less than 30 °F [65°F]" - site specific value from Core Cooling CSF Status Tree Orange Path		
Difference	None		
Deviation	• The NEI phrase "in excess of" has been changed to "GREATER THAN OR EQUAL TO" so that the EAL threshold agrees with the level specified in CSF-ST Critical Safety Function Status Trees. Using "greater than or equal to" is conservative deviation. Therefore, this deviation does not decrease the effectiveness of the NEI EAL and does not adversely effects the health/safety of the public. • KNPP RVLIS does not have the capability to measure the top of active fuel. There are two separate conditions at KNPP for determining reactor water level less than the top of active fuel based on if the RCP(s) are running. These two conditions are covered by two separate orange paths in the Core Cooling CSF Status Tree. ➢ The first condition, Core exit thermocouples GREATER THAN OR EQUAL TO 700°F with RCPs <u>NOT</u> running indicates degraded core cooling per BKG FR-C.2, Response to Degraded Core Cooling. 700°F indicates a superheated condition, which supports the basis for reactor vessel level below the top of the active fuel per EOP Setpoints. ➢ The second condition, RVLIS void fraction rising with at least one RCP running <u>and</u> RCS Subcooling less than 30 °F [65°F], is based on limitations of the KNPP RVLIS indication. The vessel level indication is only valid with the RCPs not running and is only capable of measuring down to the bottom of the RCS hot legs. This EAL uses alternate indication to detect water at top of active fuel and does not change the		

	emergency classification. Therefore, this deviation does not decrease the effectiveness of the NEI EAL and does not adversely effects the health/safety of the public.

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 4	**SG Secondary Side Release With P-to-S Leakage** Not Applicable	N/A	N/A
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 5	**CNMT Isolation Valves Status After CNMT Isolation** Not Applicable	N/A	N/A
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 6	**<u>Significant Radioactive Inventory in Containment</u>** Containment rad monitor reading GREATER THAN (site-specific) R/hr	CMT P-Loss 6	**<u>Significant Radioactive Inventory in Containment</u>** Containment rad monitor (R-40/41) reading GREATER THAN 4000 R/hr
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	▪ 4000 R/hr is the site value for 20% clad damage. Refer to Calc. C11617 ▪ R-40 and R-41 are the KNPP containment high range radiation monitors.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 7	**<u>Other (Site-Specific) Indications</u>** (Site-specific) as applicable	N/A	N/A
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		N/A
Site specific	N/A		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
CMT P-Loss 8	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	CMT P-Loss 7	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Operating, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

Fission Product Barrier Degradation

FPB – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. In general KNPP plant specific information was added or replaced non-specific NEI information. Un-necessary NEI EAL development information was deleted.	1. Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end user would not need.
2. Other (Site-Specific) Indication sections were deleted for Fuel Clad, RCS and Containment.	2. None identified
3. Emergency Director Judgment (Fuel Clad, RCS and Containment) – the following addition guidance has been included in the basis section. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. ▪ <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks. ▪ <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results. ▪ <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.	3. The bulleted items in the bases for ED judgment are an amalgam of bases information from NEI 99-01. • The first bulleted item comes from the notes on Table 5-F-1 as well as sections 3.9 and 3.10 of the NEI document regarding "imminent" barrier loss. • The second from the bases of IC HG1 loss of all AC regarding degraded barrier monitoring capability that appears appropriate here. • The third bulleted item also comes from IC HG1 as well as SG2 (ATWS) regarding the importance of the use of ED judgment to make anticipatory declarations based on FPB monitoring.

<u>Difference</u>	Addition of site specific information and clarifying information.
<u>Deviations</u>	None

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA	HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	(Site-Specific) method indicates felt earthquake	HU1.1	Earthquake felt in plant as indicated by: Consensus of Control Room operators on duty **AND** Activation of seismic monitor with Trigger light lit in Relay Room on RR159 (SER 330 Seismic Monitor Event)
Site specific	▪ "Consensus of Control Room operators on duty" is used to eliminate classification based on a single monitor reading which may be the result of actions in the area other than an actual seismic event. ▪ "Activation of seismic monitor with Trigger light lit in Relay Room on RR159 (SER 330 Seismic Monitor Event)" describes the KNPP site specific seismic monitor. ▪ This EAL reflects the site-specific method of indicating a felt earthquake.		
Difference	• Reworded EAL for readability. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Report by plant personnel of tornado or high winds greater than (site-specific) mph striking within PROTECTED AREA boundary	HU1.2	Report by plant personnel of tornado or high winds GREATER THAN 100 mph striking within PROTECTED AREA boundary.
Site specific	▪ 100 mph winds are the KNPP USAR basis.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	Vehicle crash into plant structures or systems within PROTECTED AREA boundary	HU1.3	Vehicle crash into plant structures containing functions and systems required for safe shutdown of the plant within the PROTECTED AREA boundary.
Site specific	None		
Difference	"containing functions and systems required for safe shutdown of the plant" was added to the EAL to clarify the EAL with information from the NEI basis document.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment	HU1.4	Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
5	Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.	HU1.5	Report of turbine failure resulting in casing penetration or damage to turbine-generator seals.
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
6	Uncontrolled flooding in (site-specific) areas of the plant that has the potential to affect safety related equipment needed for the current operating mode	HU1.6	Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode: • Diesel Generator A Room • Diesel Generator B Room • Safeguards Alley • Relay Room • CRDM Equipment Room • RHR Pump Pits • Auxiliary Building Basement • Screen House
Site specific	▪ List of areas is KNPP areas that could be susceptible to flooding damage to safety related equipment.		
Difference	• The words "the following" were added for readability. This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
7	(Site Specific) occurrences affecting the PROTECTED AREA	HU1.7	High or low lake level in excess of column "Unusual Event", Lake-Forebay Level Thresholds, Table H-2 for GREATER THAN 15 minutes.
Site specific	• Table H-2 contains the KNPP high and low lake water limits for an Unusual Event.		
Difference	• EAL HU1.7 was added to address site specific high and low water level conditions. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

HU1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.	1. Convert NEI basis to KNPP specific basis. Also, removed unnecessary information the end user would not need.
2. Added plant specific HU1.7 to address high and low Lake Michigan water levels.	2. High or low water level conditions may threaten operability of plant cooling systems.
Difference	• Basis was made plant specific. • HU1.7 was added to address abnormal Lake Michigan water levels.
Deviations	None

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection	HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	FIRE in buildings or areas contiguous to any of the following (site-specific) areas not extinguished within 15 minutes of control room notification or verification of a control room alarm: (Site-specific) list	HU2.1	FIRE in the PROTECTED AREA not extinguished within 15 minutes of control room notification or verification of a control room alarm.
Site specific	▪ "PROTECTED AREA" was utilized to denote "buildings or areas contiguous to .." because all buildings in the Protected Area are contiguous to Vital Areas as described in the basis section of the NEI document. This wording also meets the intent as stated in the NEI IC.		
Difference	▪ "PROTECTED AREA" was utilized to denote "buildings or areas contiguous to .." because all buildings in the Protected Area are contiguous to Vital Areas as described in the basis section of the NEI document. This wording also meets the intent as stated in the NEI IC. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

HU2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.		1. Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end user would not need.
2. Delete "Verification of a fire detection system alarm includes actions that can be taken with the control room or other nearby site-specific location to ensure the alarm is not spurious".		2. This statement was not applicable to KNPP and therefore deleted.
3. Delete "The intent of this IC is not to include buildings (i.e., warehouses)...		3. Due to the size of KNPP's protected area, buildings in the Protected Area are contiguous to Vital Areas as described in the basis section of the NEI document
Difference	• Basis was made plant specific due it plant equipment and indications. • Because all buildings in the Protected Area are contiguous to Vital Areas as described in the basis section of the NEI document, HU2.1 was changed to Protected Area to include all areas contiguous to Vital Areas. The Protected Area includes office / administrative areas and warehouses. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.	
Deviations	None	

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant	HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS	HU3.1	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event	HU3.2	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.
Site specific	None		
Difference	None		
Deviation	None		

HU3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
None	N/A
Difference	N/A
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant	HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision	HU4.1	Security Shift Supervisor reports ANY of the following: ▪ Suspected sabotage device discovered within the plant PROTECTED AREA ▪ Suspected sabotage device discovered outside the PROTECTED AREA or in the plant switchyard ▪ Confirmed tampering with safety related equipment ▪ A hostage or extortion situation that disrupts NORMAL PLANT OPERATIONS ▪ Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS ▪ Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the PROTECTED AREA ▪ Malevolent use of a vehicle outside the PROTECTED AREA which disrupts NORMAL PLANT OPERATIONS.
Site specific	• "Suspected sabotage device discovered within the plant Protected Area. Suspected sabotage device discovered outside the Protected Area or in the plant switchyard, Confirmed tampering with safety related equipment, A hostage situation that disrupts normal plant operations, Civil disturbance or strike which disrupts normal plant operations, Internal disturbance that is not short lived or that is not a harmless outburst involving ANY individuals within the Protected Area, and Malevolent use of a vehicle outside the Protected Area which disrupts normal plant operations" comes from the list of site specific areas from the Physical Security Plan		

	• "Security Shift Supervisor " is the KNPP position for Security Shift Supervision.
Difference	• The NEI words were rearranged for readability when incorporating the bullet list. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL
Deviation	None

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	A credible site specific security threat notification	HU4.2	A credible site specific security threat notification.
Site specific	None		
Difference	None		
Deviation	None		

HU4 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.	1. Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end user would not need.
2. "A credible site-specific security threat..." was added to define what kinds of threats are meant.	2. Additional information for clarification.
Difference	Basis was made plant specific.
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE	HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	HU5.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
Site specific	None		
Difference	None		
Deviation	None		

HU5 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. None	1. N/A
Difference	Basis was made plant specific.
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA	HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	(Site-Specific) method indicates Seismic Event greater than Operating Basis Earthquake (OBE)	HA1.1	Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by activation of seismic monitor with OBE Limit Exceeded light lit in Relay Room on RR159 (SER 331 Seismic Monitor Operational Basis Earthquake)
Site specific	▪ "activation of seismic monitor with OBE Limit Exceeded light lit in Relay Room on RR159 (SER 331 Seismic Monitor Operational Basis Earthquake)" is the KNPP indication of an Operating Basis Earthquake.		
Difference	• Reworded EAL for readability. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Tornado or high winds greater than (site-specific) mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems. • Reactor Building • Intake Building • Ultimate Heat Sink • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room • Other (Site-Specific) Structures	HA1.2	Tornado or high winds GREATER THAN 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment located in Table H-1 areas or Control Room indication of degraded performance of those systems located within Table H-1 areas.
Site specific	▪ Table H-1 provides the plant-specific list of structures, which encompass plant vital areas. ▪ 100 mph winds are the KNPP USAR basis.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
3	Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control indication of degraded performance of those systems: • Reactor Building • Intake Building • Ultimate Heat Sink • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room • Other (Site-Specific) Structures.	HA1.3	Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment located in Table H-1 areas or Control Room indication of degraded performance of those systems located within Table H-1 areas.
Site specific	• Table H-1 provides the plant-specific list of structures which encompass plant vital areas		
Difference	• Changed "control" to "Control Room" to be consistent with other EALs in this subgroup. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
4	Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas: (site-specific) list.	HA1.4	Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any plant areas listed in Table H-1.
Site specific	• Table H-1 provides the plant-specific list of structures which encompass plant vital areas.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
5	Uncontrolled flooding in (site-specific) areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment	HA1.5	Uncontrolled flooding in the following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment: • Diesel Generator A Room • Diesel Generator B Room • Safeguards Alley • Relay Room • CRDM Equipment Room • RHR Pump Pits • Auxiliary Building Basement • Screen House
Site specific	• List of areas is KNPP areas that could be susceptible to flooding damage to safety related equipment.		
Difference	• The words "the following" were added for readability. This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
6	(Site-Specific) occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems	HA1.6	High or low lake level in excess of column "Alert", Lake-Forebay Level Thresholds, Table H-2 for GREATER THAN 15 minutes.
Site specific.	• Table H-2 contains the KNPP high and low lake water limits for an Alert.		
Difference	• EAL HA1.6 was added to address site specific high and low water level conditions. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

HA1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.	1. Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end used would not need.
2. Added plant specific HA1.6 to address high and low Lake Michigan water levels.	2. High or low water level conditions may threaten operability of plant cooling systems.
<u>Difference</u>	• Basis was made plant specific. • HA1.6 was added to address abnormal Lake Michigan water levels.
<u>Deviations</u>	None

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown	HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	FIRE or EXPLOSION in any of the following (site-specific) areas: (Site-specific) list **AND** Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area	HA2.1	FIRE or EXPLOSION in any of the following areas (Table H-1) **AND** Affected safety system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment needed for safe shutdown
Site specific	▪ Table H-1 provides the plant-specific list of structures, which encompass plant vital areas.		
Difference	• "needed for safe shutdown" replaced within the specified area to be consistent with wording in the basis document. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

HA2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
In general KNPP plant specific information replaced non-specific NEI information.		Convert NEI basis to KNPP specific basis.
Difference	Basis was made plant specific.	
Deviations	N/A	

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown	HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Report or detection of toxic gases within or contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH)	HA3.1	Report or detection of toxic gases within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH)
Site specific	• Table H-1 provides the plant-specific list of structures which encompass plant vital areas.		
Difference	• Added "Safe Shutdown" to be consistent with the title of Table H-1. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA	HA3.2	Report or detection of gases in concentrations GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1)
Site specific	• Table H-1 provides the plant-specific list of structures which encompass plant vital areas and areas contiguous to plant vital areas.		
Difference	• Added "Safe Shutdown" to be consistent with the title of Table H-1. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

HA3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. In general KNPP plant specific information was added.	1. Convert NEI basis to KNPP specific basis.
2. Added "(in actual contact with or immediately adjacent)" after the word "contiguous".	2. Added clarifying information.
Difference	Basis was made plant specific and clarifying information was added.
Deviations	N/A

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA4	Confirmed Security Event in a Plant PROTECTED AREA	HA4	Confirmed Security Event in a Plant PROTECTED AREA
Mode App.	All		All
Site specific	None		
Difference	None.		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE	HA4.1	INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE
Site specific	None		
Difference	None		
Deviation	None		

<table>
<tr><th>NEI EAL#</th><th>NEI EAL Wording</th><th>KNPP EAL#(s)</th><th>KNPP EAL Wording</th></tr>
<tr><td>2</td><td>Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision</td><td>HA4.2</td><td>Security Shift Supervisor reports any of the following:
▪ Sabotage device discovered in the plant PROTECTED AREA
▪ Standoff attack on the protected area by a HOSTILE FORCE (i.e., Sniper)
▪ ANY Security event of increasing severity that persists for GREATER THAN 30 minutes:
• Credible bomb threats
• Hostage / Extortion
• Suspicious Fire or Explosion
• Significant Security System Hardware Failure
• Loss of contact with Security Officers</td></tr>
<tr><td>Site specific</td><td colspan="3">• Sabotage device discovered in the Protected Area, Standoff attack on the Protected Area by a hostile force (i.e., Sniper), ANY security event of increasing severity that persists for GREATER THAN 30 minutes, Credible bomb threats, Extortion, Suspicious fire or explosion, Significant Security System Hardware Failure, Loss of contact with Security Officers" comes from the list of site specific areas from the Physical Security Plan</td></tr>
<tr><td>Difference</td><td colspan="3">• Reworded EAL for readability.
• This change is not a deviation because it meets the intent of NEI 99-01.</td></tr>
<tr><td>Deviation</td><td colspan="3">None</td></tr>
</table>

<table>
<tr><th colspan="2">HA4 – Basis Justification</th></tr>
<tr><th>KNPP Specific Additions/Deletions</th><th>Justification</th></tr>
<tr><td>In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.</td><td>Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end used would not need.</td></tr>
<tr><td><u>Difference</u></td><td>Basis was made plant specific.</td></tr>
<tr><td><u>Deviations</u></td><td>N/A</td></tr>
</table>

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA5	Control Room Evacuation Has Been Initiated	HA5	Control Room Evacuation Has Been Initiated
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Entry into (site-specific) procedure for control room evacuation	HA5.1	Entry into E-O-06, Fire in Alternate Fire Zone for Control Room Evacuation
Site specific	• "E-O-06, Fire in Alternate Fire Zone" is the site-specific procedure for control room evacuation.		
Difference	None		
Deviation	None		

HA5 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
KNPP plant specific information (E-O-06) was added.	Plant specific reference to E-O-06 procedure was added to convert NEI basis to KNPP specific basis.
Difference	Basis was made plant specific.
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert	HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels	HA6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.
Site specific	None.		
Difference	None		
Deviation	None		

HA6 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added the following to the basis: Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.	Information added on location of EPA limits.
Difference	N/A
Deviations	N/A

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HS1	Confirmed Security Event in a Plant VITAL AREA	HS1	Confirmed Security Event in a Plant VITAL AREA
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE	HS1.1	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision	HS1.2	Security Supervision reports ANY of the following: • A security event that results in the loss of control of ANY VITAL AREAS (other than Control Room) • Imminent loss of physical control of the facility (remote shutdown capability) due to a security event • A confirmed sabotage discovered in a VITAL AREA
Site specific	• The "Security Shift Supervisor" is the title of the site-specific security supervision • Addition of security events to cover the following: loss of control of any vital area, imminent lose of physical control and sabotage in vital area.		
Difference	• Reworded EAL for readability. • This change is not a deviation because it meets the intent of NEI 99-01.		
Deviation	None		

HS1 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.	Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end user would not need.
Difference	Basis was made plant specific.
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established	HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Control room evacuation has been initiated. **AND** Control of the plant cannot be established per (site-specific) procedure within (site-specific) minutes	HS2.1	Control room evacuation has been initiated **AND** Control of the plant cannot be established per E-O-06, Fire in Alternate Fire Zone within 15 minutes
Site specific	• "E-O-06, Fire in Alternate Fire Zone" is the site-specific procedure for control room evacuation. • Fifteen minutes is the NEI standard without additional justification.		
Difference	None		
Deviation	None		

HS2 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.	Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end used would not need.
Difference	Basis was made plant specific.
Deviations	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency	HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary	HS3.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.
Site specific	None		
Difference	None		
Deviation	None		

HS3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added the following to the basis: Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.	Added information on location of EPA Limits.
<u>Difference</u>	N/A
<u>Deviations</u>	N/A

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HG1	Security Event Resulting in Loss Of Physical Control of the Facility	HG1	Security Event Resulting in Loss Of Physical Control of the Facility
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions	HG1.1	A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of EITHER: A VITAL AREA (including the Control Room) such that operation of equipment required for safe shutdown is lost **<u>OR</u>** Spent fuel pool cooling systems if imminent fuel damage is likely
Site specific	None		
Difference	• "A VITAL AREA (including the Control Room) such that operation of equipment required for safe shutdown is lost OR Spent fuel pool cooling systems if imminent fuel damage is likely" is KNPP site specific description of equipment required to maintain safety functions. The Control Room is included because there is a sufficient loss of control even if operators have established control at the Dedicated Shutdown Panel.		
Deviation	None		

HG1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
In general KNPP plant specific information was added or replaced non-specific NEI information. Unnecessary NEI EAL development information was deleted.		Convert NEI basis to KNPP specific basis. Also, remove unnecessary information the end user would not need.
Difference	Basis was made plant specific.	
Deviations	N/A	

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency	HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency
Mode App.	All		All
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	HG2.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.
Site specific	None		
Difference	None		
Deviation	None		

HG2 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
Added the following to the basis: Refer to EPIP-AD-19 for EPA Protective Action Guideline exposure levels.	Added information on location of EPA Limits.
Difference	N/A
Deviations	N/A

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU1	Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes	SU1	Loss of All Offsite Power To Essential Busses for GREATER THAN 15 minutes
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of power to (site specific) transformers for greater than 15 minutes **AND** At least (site specific) emergency generators are supplying power to emergency busses.	SU1.1	Loss of all offsite power to Bus 5 <u>AND</u> Bus 6 for GREATER THAN 15 minutes <u>**AND**</u> Emergency diesel generators are supplying power to Bus 5 <u>AND</u> Bus 6
Site specific	▪ "Loss of all offsite power to Bus 5 <u>AND</u> Bus 6" has been used in place of "Loss of power to (site specific) transformers" to focus the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses ▪ Bus 5 and Bus 6 are the KNPP emergency safeguards buses. ▪ "Emergency diesel generators are supplying power to Bus 5 <u>AND</u> Bus 6" was added to describe the emergency diesel generator configuration at KNPP.		
Difference	▪ The NEI example EAL condition "Loss of power to (site-specific) transformers for greater than 15 minutes" has been changed to "Loss of all offsite power to Bus 5 <u>AND</u> Bus 6 for GREATER THAN 15 minutes." The KNPP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses. This simplifies the EAL wording and concisely meets the intent of the NEI IC ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SU1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. Description of the KNPP ESF power distribution system was included into the EAL basis.		1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.		2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.	
Deviations	None	

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU2	Inability to Reach Required Shutdown Within Technical Specification Limits	SU2	Inability to Reach Required Shutdown Within Technical Specification limits
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Plant is not brought to required operating mode within (site-specific) Technical Specifications LCO Action Statement Time	SU2.1	Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.
Site specific	▪ None		
Difference	• Site specific times are not included due to the varied length of time associated with individual LCOs. Therefore, EAL is generic to cover all LCO's. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SU2 – Basis Justification	
KNPP Specific Additions/Deletions	Justification
1. General KNPP plant specific information was added or replaced non-specific NEI information.	1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.
Deviations	None

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes	SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 minutes
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	UNPLANNED loss of most or all (site-specific) annunciators or indicators associated with safety systems for greater than 15 minutes	SU3.1	UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console •
Site specific	• "Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console" contain the site-specific annunciators or indicators associated with safety systems		
Difference	None		
Deviation	None		

SU3– Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. General KNPP plant specific information was added or replaced non-specific NEI information.	1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.
Deviations	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU4	Fuel Clad Degradation	SU4	Fuel Cladding Degradation
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	(Site-specific) radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits	SU4.1	RCS Letdown Line (R-9) radiation monitor GREATER 2000 mR/hr indicating fuel clad degradation
Site specific	• RCS Letdown Line (R-9) is the site specific monitor designated to indicate fuel clad failure. • 2000 mR/hr is equal to the Technical Specification allowable limits.		
Difference	• Reworded NEI EAL for readability • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	(Site-specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits	SU4.2	Coolant sample activity GREATER THAN <u>ANY</u> of the following indicating fuel clad degradation: • 1.0 μCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval • 60 μCi/gram dose equivalent Iodine-131. • 91/Ē μCi/cc gross radioactivity
Site specific	• The following are KNPP Technical Specification Limits for coolant sample activity: • 1.0 μCi/gram dose equivalent Iodine-131 for more than 48 hours in one continuous time interval • 60 μCi/gram dose equivalent Iodine-131. • 91/Ē μCi/cc gross radioactivity		
Difference	• "technical specification allowable limits" – was deleted as it duplicates the site specific tech spec value already listed in the EAL • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SU4 – Basis Justification	
KNPP Specific Additions/Deletions	**<u>Justification</u>**
1. Added site-specific information on R-9 setpoint for SU4.1	1. This information was added to clarify the information used to determine the R-9 setpoint
<u>Difference</u>	Added KNPP site specific information
<u>Deviations</u>	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU5	RCS Leakage	SU5	RCS Leakage
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Unidentified or pressure boundary leakage greater than 10 gpm	SU5.1	Unidentified or pressure boundary leakage GREATER THAN 10 gpm
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Identified leakage greater than 25 gpm	SU5.2	Identified leakage GREATER THAN 25 gpm
Site specific	None		
Difference	None		
Deviation	None		

SU5 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.		This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.	
Deviations	None	

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	SU6.	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of all (site-specific list) onsite communications capability affecting the ability to perform routine operations	SU6.1	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations
Site specific	▪ Table C-1 lists onsite communications systems.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	Loss of all (site-specific list) offsite communications capability.	SU6.2	Loss of all Table C-2 offsite communications capability.
Site specific	Table C-2 lists offsite communications systems		
Difference	None		
Deviation	None		

SU6 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
General KNPP plant specific information was added in Tables C-1 and C-2.	This information was added for explanation and clarification of site specifics.

Difference	Added KNPP site specific information.
Deviations	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SU8	Inadvertent Criticality	SU8	Inadvertent Criticality
Mode App.	Hot Standby, Hot Shutdown		Hot Shutdown, Intermediate Shutdown
Site specific	KNPP Modes Hot Shutdown and Intermediate Shutdown are equivalent to NEI 99-01 modes Hot Standby and Hot Shutdown.		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	An UNPLANNED extended positive period observed on nuclear instrumentation	N/A	N/A.
Site specific	N/A		
Difference	Not applicable, BWR NEI EAL.		
Deviation	N/A		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
2	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	SU8.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
Site specific	None		
Difference	N/A		
Deviation	None		

SU8 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.	This information was added for explanation and clarification of site specifics.

Difference	Added KNPP site specific information.
Deviations	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SA2	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful	SA2	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was Successful
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Indication(s) exist that indicate that reactor protection system setpoint was exceeded and automatic scram did not occur, and a successful manual scram occurred	SA2.1	Indication(s) exist that a Reactor Protection System setpoint was exceeded **AND** RPS automatic trip did not reduce power to LESS THAN 5%: **AND** Any of the following actions are successful in reducing power to LESS THAN 5% ▪ Use of Manual Reactor Trip push buttons ▪ De-energizing Buses 33 AND 43
Site specific	▪ In response to industry questions concerning the definition of a successful reactor trip, NEI and the NRC agreed in System Malfunction Question #7 of "Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Rev. 2 Questions and Answers" that "...the scram is considered unsuccessful when enough control rods have not inserted to cause the reactor power to fall below that percent power associated with the ability of the safety systems to remove heat and continue to decrease." To implement the intent of this position, the KNPP EAL wording includes the phrase "...power range $\leq$5%." ▪ Use of Reactor Trip buttons / De-energizing Buses 33 AND 43		
Difference	None		
Deviation	None		

System Malfunction

SA2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.		This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.	
Deviations	None	

<table>
<tr><th>NEI IC#</th><th>NEI IC Wording</th><th>KNPP IC#(s)</th><th>KNPP IC Wording</th></tr>
<tr><td>SA4</td><td>UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable</td><td>SA4</td><td>UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable</td></tr>
<tr><td>Mode App.</td><td>Power Operation, Startup, Hot Standby, Hot Shutdown</td><td></td><td>Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown</td></tr>
<tr><td>Site specific</td><td colspan="3">None</td></tr>
<tr><td>Difference</td><td colspan="3">None</td></tr>
<tr><td>Deviation</td><td colspan="3">None</td></tr>
</table>

<table>
<tr><th>NEI EAL#</th><th>NEI EAL Wording</th><th>KNPP EAL#(s)</th><th>KNPP EAL Wording</th></tr>
<tr><td>1</td><td>UNPLANNED loss of most or all (site-specific) annunciators or indicators associated with safety systems for greater than 15 minutes.
AND
Either of the following: (a or b)
a. A SIGNIFICANT TRANSIENT is in progress.
OR
b. Compensatory non-alarming indications are unavailable.</td><td>SA4.1</td><td>UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console
<u>AND</u>
Either of the following: (a or b):
a. A SIGNIFICANT TRANSIENT is in progress.
<u>OR</u>
b. COMPENSATORY NON-ALARMING INDICATIONS are unavailable.</td></tr>
<tr><td>Site specific</td><td colspan="3">▪ “Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console” contain the site-specific annunciators or indicators associated with safety systems.</td></tr>
<tr><td>Difference</td><td colspan="3">▪ COMPENSATORY NON-ALARMING INDICATIONS was capitalized because it is a definition in the basis document.</td></tr>
<tr><td>Deviation</td><td colspan="3">None</td></tr>
</table>

<table>
<tr><td colspan="3">SA4 – Basis Justification</td></tr>
<tr><td colspan="2"><u>KNPP Specific Additions/Deletions</u></td><td><u>Justification</u></td></tr>
<tr><td colspan="2">1. General KNPP plant specific information was added or replaced non-specific NEI information.</td><td>1. This information was added for explanation and clarification of site specifics.</td></tr>
<tr><td colspan="2">2. Development information contained in the NEI Basis was deleted.</td><td>2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.</td></tr>
<tr><td colspan="2">3. Added definition for significant transient and compensatory non-alarming indication.</td><td>3. Clarifying information to assist in classification. Definitions consistent with NEI 99-01.</td></tr>
<tr><td><u>Difference</u></td><td colspan="2">Added KNPP site specific information.</td></tr>
<tr><td><u>Deviations</u></td><td colspan="2">None</td></tr>
</table>

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SA5	AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout	SA5	AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	AC power capability to site-specific essential busses reduced to a single power source for greater than 15 minutes **AND** Any additional single failure will result in station blackout	SA5.1	AC power capability to Bus 5 AND Bus 6 reduced to only one of the following sources for GREATER THAN 15 minutes • One emergency diesel generator (A or B) • TAT • RAT • MAT on backfeed **AND** Any additional single failure will result in station blackout.
Site specific	▪ Bus 5 and Bus 6 are the KNPP emergency safeguards buses. ▪ "only one of the following sources for GREATER THAN 15 min. (one source away from station blackout): for readability. ▪ KNPP Site Specific power sources are: • One emergency diesel generator (A or B) • TAT (Tertiary Auxiliary Transformer) • RAT (Reserve Auxiliary Transformer) • MAT (Main Auxiliary Transformer)on backfeed This provides a plant-specific list of AC power sources and clearly implements the intent of the generic EAL.		
Difference	None		
Deviation	None		

SA5 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. General KNPP plant specific information was added or replaced non-specific NEI information.		1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.		2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.	
Deviations	None	

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SS1	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses	SS1	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of power to (site-specific) transformers. **AND** Failure of (site-specific) emergency generators to supply power to emergency busses. **AND** Failure to restore power to at least one emergency bus within (site-specific) minutes from the time of loss of both offsite and onsite AC power	SS1.1	Loss of ALL power to Bus 5 AND Bus 6 for GREATER THAN 15 minutes
Site specific	▪ Bus 5 and Bus 6 are the KNPP emergency safeguards buses. ▪ GREATER THAN 15 minutes was inserted as time to restore Bus 5 and Bus 6		
Difference	▪ The NEI example EAL condition "Loss of power to (site-specific) transformers…" has been changed to "Loss of ALL power to Bus 5 AND Bus 6…" The plant EAL wording focuses the classification on the loss of power capability rather than the status of one or more transformers that may or may not be capable powering the essential buses. This simplifies the EAL wording and concisely meets the intent of the NEI IC. ▪ KNPP EAL was reformatted from three NEI conditions to an encompassing one condition. This was done to simplify the classification and the economy of words. Stating that "ALL power" is lost to Bus 5 and Bus 6 covers the first two NEI conditions (loss of power to the transformers and failure of the emergency diesel generators). "For GREATER THAN 15 minutes" is equivalent to the third NEI condition (time to restore power to at least one emergency bus). ▪ These changes are not deviations because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SS1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
1. General KNPP plant specific information was added or replaced non-specific NEI information.		1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.		2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.	
Deviations	None	

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SS2	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful	SS2	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Reactor Trip Was NOT Successful
Mode App.	Power Operation, Startup		Power Operation, Hot Standby
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Indication(s) exist that automatic and manual scram were not successful	SS2.1	Indication(s) exist that automatic and manual reactor trip were NOT successful in reducing power to LESS THAN 5%. Manual Reactor Trips include use of Manual Reactor Trip push buttons or De-energizing Busses 33 AND 43.
Site specific	None		
Difference	• "LESS THAN 5%" - In response to industry questions concerning the definition of a successful reactor trip, NEI and the NRC agreed in System Malfunction Question #7 of "Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Rev. 2 Questions and Answers" that "...the scram is considered unsuccessful when enough control rods have not inserted to cause the reactor power to fall below that percent power associated with the ability of the safety systems to remove heat and continue to decrease." 5% is the power level specified in Subcriticality-RED path. • "Manual Reactor Trips include use of Manual Reactor Trip push buttons or De-energizing Busses 33 AND 43" was added to prevent from taking credit for reactor trip initiated outside of the Control Room. • These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SS2 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.		This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.	
Deviations	None	

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SS3	Loss of All Vital DC Power	SS3	Loss of all vital DC power
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of All Vital DC Power based on (site-specific) bus voltage indications for greater than 15 minutes	SS3.1	Loss of All Vital DC power based on LESS THAN 105 VDC on Train A AND Train B Safeguards DC Distribution Systems for GREATER THAN 15 minutes.
Site specific	▪ LESS THAN 105 VDC on Train A AND Train B Safeguards DC Distribution System is the KNPP design voltage and specific DC buses.		
Difference	▪ The design of the KNPP 125v DC Distribution System is such that a loss of different combinations of distribution panels and buses could constitute a loss of DC power to a Train. These combinations that would cause a loss of DC power are covered in the basis for this EAL. ▪ This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SS3 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.	This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.
Deviations	None

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SS4	Complete Loss of Heat Removal Capability	SS4	Complete Loss of Heat Removal Capability
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of core cooling and heat sink (PWR)	SS4.1	Loss of core cooling and heat sink
Site specific	None		
Difference	None		
Deviation	None		

SS4 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. KNPP plant specific terminology replaced non-specific NEI information.	1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.
Deviations	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SS6	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress	SS6	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	N/A		
Difference	None		
Deviation	None		

System Malfunction

<table>
<tr><th>NEI EAL#</th><th>NEI EAL Wording</th><th>KNPP EAL#(s)</th><th>KNPP EAL Wording</th></tr>
<tr><td>1</td><td>a. Loss of most or all (site-specific) annunciators associated with safety systems.
AND
b. Compensatory non-alarming indications are unavailable.
AND
c. Indications needed to monitor (site-specific) safety functions are unavailable.
AND
d. SIGNIFICANT TRANSIENT in progress.</td><td>SS6.1</td><td>Loss of most or all annunciators associated with safety systems on Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console.
AND
SIGNIFICANT TRANSIENT in progress.
AND
COMPENSATORY NON-ALARMING INDICATIONS are unavailable.
AND
Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact are unavailable.</td></tr>
<tr><td>Site specific</td><td colspan="3">▪ "Mechanical Vertical Panels A, B and C, Mechanical Control Consoles A, B and C, Electrical Vertical Panel and Electrical Control Console" contain the site-specific annunciators associated with safety systems.
▪ "the ability to shut down the reactor, maintain the core cooled, maintain the reactor system intact, and maintain containment intact" is the site specific list of safety functions</td></tr>
<tr><td>Difference</td><td colspan="3">▪ SIGNIFICANT TRANSIENT placed 2nd on list to provide user with clear escalation path criteria between EALs (Formatting change only)
▪ COMPENSATORY NON-ALARMING INDICATIONS was capitalized because it is a definition in the basis document.
▪ These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL</td></tr>
<tr><td>Deviation</td><td colspan="3">None</td></tr>
</table>

SS6 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. General KNPP plant specific information was added or replaced non-specific NEI information.	1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
3. Added definition for significant transient and compensatory non-alarming indication.	3. Clarifying information to assist in classification. Definitions consistent with NEI 99-01.
Difference	Added KNPP site specific information.
Deviations	None

System Malfunction

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SG1	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses	SG1	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby, Hot Shutdown, Intermediate Shutdown
Site specific	None		
Difference	None		
Deviation	None		

System Malfunction

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Loss of power to (site-specific) transformers. **AND** Failure of (site-specific) emergency diesel generators to supply power to emergency busses. **AND** Either of the following: (a or b) a. Restoration of at least one emergency bus within (site-specific) hours is <u>not</u> likely **OR** b. (Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.	SG1.1	Loss of all offsite power to Bus 5 <u>AND</u> Bus 6 **AND** Failure of all emergency diesel generators to supply power to Bus 5 <u>AND</u> Bus 6 **AND** Either of the following: (a or b) a. Restoration of either Bus 5 <u>OR</u> Bus 6 within 4 hours is <u>not</u> likely **OR** b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by a Core Cooling-RED or Core Cooling-ORANGE
Site specific	▪ Bus 5 and Bus 6 are the KNPP emergency safeguards buses. ▪ "Core Cooling-RED or Core Cooling-ORANGE" is KNPP site specific indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.		
Difference	• The NEI example EAL condition "Loss of power to (site-specific) transformers..." has been changed to "Loss of all offsite power to Bus 5 <u>AND</u> Bus 6..." The plant EAL wording focuses the classification on the loss of power capability rather than the status of one or more transformers that may or may not be capable powering the essential buses. • This change is not a deviation because it does not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL		
Deviation	None		

SG1 – Basis Justification		
KNPP Specific Additions/Deletions		**Justification**
General KNPP plant specific information was added or replaced non-specific NEI information.		This information was added for explanation and clarification of site specifics.
Difference	Added KNPP site specific information.	
Deviations	None	

NEI IC#	NEI IC Wording	KNPP IC#(s)	KNPP IC Wording
SG2	Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core	SG2	Failure of the Reactor Protection System to Complete an Automatic Reactor Trip and Manual Reactor Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Hot Standby
Site specific	None		
Difference	None		
Deviation	None		

NEI EAL#	NEI EAL Wording	KNPP EAL#(s)	KNPP EAL Wording
1	Indications exist that automatic and manual scram were not successful. **AND** Either of the following: (a or b) a. Indication(s) exists that the core cooling is extremely challenged. **OR** b. Indication(s) exists that heat removal is extremely challenged	SG2.1	Indication(s) exist that automatic and manual reactor trip were NOT successful in reducing power to LESS THAN 5%. **AND** Either of the following: (a or b) a. Indication(s) exists that the core cooling is extremely challenged as indicated by Core Cooling - RED. **OR** b. Indication(s) exists that heat removal is extremely challenged as indicated by Heat Sink - RED.
Site specific	▪ "Core Cooling-RED" represents the site-specific indication that core cooling is extremely challenged. ▪ "Heat Sink-RED" represents the site-specific indication that heat removal is extremely challenged. ▪ LESS THAN 5%- In response to industry questions concerning the definition of a successful reactor trip, NEI and the NRC agreed in System Malfunction Question #7 of "Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Rev. 2 Questions and Answers" that "...the scram is considered unsuccessful when enough control rods have not inserted to cause the reactor power to fall below that percent power associated with the ability of the safety systems to remove heat and continue to decrease." 5% is the power level specified in Subcriticality-RED path.		
Difference	None		
Deviation	None		

SG2 – Basis Justification	
KNPP Specific Additions/Deletions	**Justification**
1. General KNPP plant specific information was added or replaced non-specific NEI information.	1. This information was added for explanation and clarification of site specifics.
2. Development information contained in the NEI Basis was deleted.	2. Development information is not necessary after the site specific information has been developed. The basis would be very confusing if these statements were left in along with the site specific information.
Difference	Added KNPP site specific information.
Deviations	None

Kewaunee Nuclear Power Plant EALs vs NEI 99-01 rev 4 EALs
Differences / Deviations / Site Specific Information

Kewaunee Nuclear Power Plant EALs vs NEI 99-01 rev 4 EALs

Definitions

The following definitions were used from RIS 2003-18 Sup 1 to define Differences and Deviations contained in this submittal:

> **Difference:** An EAL change where the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the site-specific proposed EAL
>
> **Deviation:** An EAL change where the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the site-specific proposed EAL.

Generic Differences

The generic changes listed below are not deviation because they do not alter the meaning or intent of associated EAL's, such that classification of the event could be different between the NEI guidance and the plant EAL:

- Each EAL was numbered sequentially in the subgroups. In example: "RU1.1" is the first EAL in subgroup RU1. This was done to improve communication at the KNPP emergency facilities and with offsite agencies.
- ISFSI and Permanently Defueled NEI EAL section are not applicable to KNPP and have been deleted.
- The symbols <, >, etc. were replaced with "LESS THAN", "GREATER THAN", etc. is to be consistent with other KNPP documents (Procedures Writer Guide GNP-03.01.04) and for human factors.
- The "LESS THAN", "GREATER THAN", etc wording is capitalized to be consistent with other KNPP documents (Procedures Writer Guide GNP-03.01.04) and for human factors.
- "Reactor Scram" was replaced with "Reactor Trip" to be consistent with KNPP wording.
- Capitalization and Bold of logic "AND" / "OR" is to be consistent with other KNPP documents and for human factors.
- NEI Operating Mode Applicability "Startup" was deleted and "Intermediate Shutdown" was added to conform to KNPP's Operating Modes. Also, Power Operation was changed to Operating to conform to KNPP's Operating Modes. Refer to Technical Specification page TS 1.0-4.
- NOUE (Notice of Unusual Event) was changed to KNPP wording of UE (Unusual Event).
- The words "increase" and "decrease" has been replaced with "raise" / "rise" and "lowering". This change was done to be consistent with KNPP communication standards

and for human factors. The words 'increase' and 'decrease' are not normally used because they are easily misunderstood.

- "Reactor Vessel" was used in place of "RPV" to match site procedure verbiage and PWR terminology.

- "Exceeds" was replaced with "GREATER THAN" to allow consistency between other KNPP documents.

- Words that are defined in the EAL Bases were capitalized to indicate define words.

<u>General Development Information</u>

Unless otherwise documented in the EAL bases, values and setpoints contained with in the EAL's were made in accordance with NEI 99-01 rev 4 guidelines and are not addressed separately in the Difference and Deviation Matrix.

NEI 99-01 / KNPP Cross Reference

99-01 IC	99-01 EAL #	PBNP EAL Number
Abnormal Radiation Levels / Radiological Effluent		
AU1	1	RU1.1
	2	RU1.2
	3	RU1.3
	4	N/A
	5	N/A
AU2	1	RU2.1
	2	RU2.2
AA1	1	RA1.1
	2	RA1.2
	3	RA1.3
	4	N/A
	5	N/A
AA2	1	RA2.1
	2	RA2.2
AA3	1	RA3.1
	2	RA3.2
AS1	1	RS1.1
	2	RS1.2
	3	N/A
	4	RS1.3
AG1	1	RG1.1
	2	RG1.2
	3	N/A
	4	RG1.3
Cold Shutdown / Refueling System Malfunction		
CU1	1	CU1.1
	2	CU1.2
CU2	1	CU2.1
	2	CU2.2
CU3	1	CU3.1
CU4	1	CU4.1
	2	CU4.2
CU5	1	CU5.1
	2	CU5.2
CU6	1	CU6.1
	2	CU6.2
CU7	1	CU7.1
CU8	1	N/A
	2	CU8.1
CA1	1	CA1.1
	2	CA1.2
CA2	1	CA2.1
	2	CA2.2
CA3	1	CA3.1
CA4	1	CA4.1
	2	CA4.2
	3	CA4.3
CS1	1	CS1.1
	2	CS1.2

99-01 IC	99-01 EAL #	PBNP EAL Number
CS2	1	CS2.1
	2	CS2.2
CG1	1	CG1.1
	2	CG1.1
	3	CG1.1
Defueled Station Malfunction		
D-AU1	1	N/A
	2	N/A
D-AU2	1	N/A
D-SU1	1	N/A
	2	N/A
D-HU1	1	N/A
D-HU2	1	N/A
D-HU3	1	N/A
	2	N/A
	3	N/A
	4	N/A
	5	N/A
	6	N/A
	7	N/A
	8	N/A
D-AA1	1	N/A
	2	N/A
D-AA2	1	N/A
	2	N/A
D-HA1	1	N/A
D-HA2	1	N/A
Events Related to Independent Spent Fuel Storage Installations		
E-HU1	1	N/A
	2	N/A
	3	N/A
E-HU2	1	N/A
Fission Product Barrier Degradation		
FU1		FU1.1
FA1		FA1.1
FS1		FS1.1
FG1		FG1.1
Fuel Cladding Loss - 1		Fuel Cladding Loss - 1
Fuel Cladding Loss - 2		Fuel Cladding Loss - 2
Fuel Cladding Loss - 3		Fuel Cladding Loss - 3
Fuel Cladding Loss - 4		Fuel Cladding Loss - 4
Fuel Cladding Loss - 5		Fuel Cladding Loss – 5
Fuel Cladding Loss - 6		N/A
Fuel Cladding Loss - 7		Fuel Cladding Loss – 6
Fuel Cladding P-Loss - 1		Fuel Cladding P-Loss - 1
Fuel Cladding P-Loss - 2		Fuel Cladding P-Loss - 2
Fuel Cladding P-Loss - 3		Fuel Cladding P-Loss - 3
Fuel Cladding P-Loss - 4		Fuel Cladding P-Loss - 4
Fuel Cladding P-Loss – 5		Fuel Cladding P-Loss – 5
Fuel Cladding P-Loss – 6		N/A
Fuel Cladding P-Loss - 7		Fuel Cladding P-Loss - 6
RCS Loss - 1		RCS Loss - 1

99-01 IC	99-01 EAL #	PBNP EAL Number
RCS Loss - 2		RCS Loss - 2
RCS Loss - 3		RCS Loss - 3
RCS Loss - 4		RCS Loss - 4
RCS Loss – 5		N/A
RCS Loss - 6		RCS Loss - 5
RCS P-Loss -1		RCS P-Loss -1
RCS P-Loss -2		RCS P-Loss -2
RCS P-Loss -3		RCS P-Loss -3
RCS P-Loss -4		RCS P-Loss -4
RCS P-Loss –5		N/A
RCS P-Loss -6		RCS P-Loss -5
Containment Loss - 1		Containment Loss - 1
Containment Loss – 2		Containment Loss – 2
Containment Loss – 3		Containment Loss – 3
Containment Loss – 4		Containment Loss – 4
Containment Loss – 5		Containment Loss – 5
Containment Loss – 6		Containment Loss – 6
Containment Loss - 7		N/A
Containment Loss - 8		Containment Loss - 7
Containment P-Loss - 1		Containment P-Loss - 1
Containment P-Loss - 2		Containment P-Loss - 2
Containment P-Loss - 3		Containment P-Loss - 3
Containment P-Loss – 4		Containment P-Loss – 4
Containment P-Loss – 5		Containment P-Loss – 5
Containment P-Loss – 6		Containment P-Loss – 6
Containment P-Loss – 7		N/A
Containment P-Loss - 8		Containment P-Loss - 8
Hazards and Other Conditions Affecting Plant Safety		
HU1	1	HU1.1
	2	HU1.2
	3	HU1.3
	4	HU1.4
	5	HU1.5
	6	HU1.6
	7	HU1.7
HU2	1	HU2.1
HU3	1	HU3.1
	2	HU3.2
HU4	1	HU4.1
	2	HU4.2
HU5	1	HU5.1
HA1	1	HA1.1
	2	HA1.2
	3	HA1.3
	4	HA1.4
	5	HA1.5
	6	HA1.6
HA2	1	HA2.1
HA3	1	HA3.1
	2	HA3.2
HA4	1	HA4.1
	2	HA4.2

99-01 IC	99-01 EAL #	PBNP EAL Number
HA5	1	HA5.1
HA6	1	HA6.1
HS1	1	HS1.1
	2	HS1.2
HS2	1	HS2.1
HS3	1	HS3.1
HG1	1	HG1.1
HG2	1	HG2.1
System Malfunction		
SU1	1	SU1.1
SU2	1	SU2.1
SU3	1	SU3.1
SU4	1	SU4.1
	2	SU4.2
SU5	1	SU5.1
	2	SU5.2
SU6	1	SU6.1
	2	SU6.2
SU8	1	N/A
	2	SU8.1
SA2	1	SA2.1
SA4	1	SA4.1
SA5	1	SA5.1
SS1	1	SS1.1
SS2	1	SS2.1
SS3	1	SS3.1
SS4	1	SS4.1
SS6	1	SS6.1
SG1	1	SG1.1
SG2	1	SG2.1

NMC
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Kewaunee, WI 54216-9511
920-388-2560

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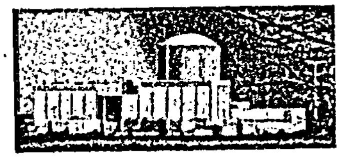
Nuclear Management Company, LLC



KNPL 2004-0019

October 7, 2004

Lori Hucek, Kewaunee Emergency Management
416 Fremont Street
Kewaunee, WI 54201

Dear Lori:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs) at the meeting conducted on Thursday, October 7, 2004, held at Stevens Point.

The purpose of the meeting was to discuss the difference between the current NUREG-0654 scheme EALs and proposed conversion to NEI 99-01, Rev. 4 scheme.

The NRC approved the current Kewaunee Nuclear Plant EALs in their 1982 Safety Evaluation Report. The changes we discussed today will incorporate the NEI 99-01 EAL scheme into the Kewaunee Nuclear Plant EALs. The NRC endorsed the NEI 99-01, Rev. 4 scheme via Reg. Guide 1.101, Rev. 4, July 2003.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

Please review the Kewaunee EAL Bases document by October 21, 2004. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8117.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie A. Coleman

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Lori Hucek TR — 10/13/2004

Lori Hucek, Kewaunee Emergency Management — Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

Attachment 1

Pa

Kewaunee County Emergency Management

Lori Hucek, Director

KNPL 2004-0023

October 21, 2004

Lori Hucek, Kewaunee Emergency Management
416 Fremont Street
Kewaunee, WI 54201

Dear Lori:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time this morning to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs). Listed in Attachment 1 are the significant changes made to the KNPP NEI 99-01 EAL scheme since our meeting on October 7, 2004. The changes are based upon feedback from the NRC, Challenge Board, Plant Operation Review Committee and NMC Peer Group.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

Based on the discussion we this morning, I understand that you find the changes acceptable and concur with Kewaunee's additional changes to convert to NEI 99-01, Rev. 4 scheme. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8675.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Lori Hucek

Lori Hucek, Kewaunee Emergency Management

10/21/04
Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

416 Fremont Street • Algoma, WI 54201 • Phone (920) 487-2940 • Fax (920) 487-2963

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KNPL 2004-0020

October 7, 2004

Nancy Crowley, Manitowoc Emergency Management
1025 South 9th Street
Manitowoc, WI 54220

Dear Nancy:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs) at the meeting conducted on Thursday, October 7, 2004, held at Stevens Point.

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Please review the Kewaunee EAL Bases document by October 21, 2004. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8117.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie A. Coleman

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Nancy H. Crowley — 10-12-04

Nancy Crowley, Manitowoc Emergency Management — Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

MANITOWOC COUNTY

EMERGENCY SERVICES DIVISION

04-0024

October 21, 2004

Nancy Crowley, Manitowoc Emergency Management
1025 South 9th Street
Manitowoc, WI 54220

Dear Nancy:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time this morning to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs). Listed in Attachment 1 are the significant changes made to the KNPP NEI 99-01 EAL scheme since our meeting on October 7, 2004. The changes are based upon feedback from the NRC, Challenge Board, Plant Operation Review Committee and NMC Peer Group.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

Based on the discussion we this morning, I understand that you find the changes acceptable and concur with Kewaunee's additional changes to convert to NEI 99-01, Rev. 4 scheme. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8675.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Nancy H. Crowley

Nancy Crowley, Manitowoc Emergency Management

10-21-04
Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

Nancy H. Crowley, R.N., C.E.M
Division Coordinator
Emergency Management Director
Phone: 920-683-4207
Fax: 920-683-4568
e-mail: nhcrowley@sbcglobal.net

Kay Beilke
Administrator
Joint Dispatch Center
Phone: 920-683-5033
Fax: 920-683-4946
e-mail: klb0803@mtso.manitowoc.wi.u

1025 S. 9th Street • Manitowoc, Wisconsin • 54220

Attachment 1

Tab 2 Abnormal Rad Levels / Radiological Effluent Section

RU1.1 RA1.1

Radiation values for radiation monitors R-13, R-14, R-12 and R-21 have been corrected for the Alert and Unusual Event setpoints. The value remains two times the ODCM setpoints as stated in the EAL document. Incorrect base numbers led to the error.

Radiation Monitor for Waste Disposal System Liquid (R-18) setpoint was changed to 2x's the Calculated ODCM Setpoint for UE and 200x's the Calculated ODCM Setpoint for Alert. This change was due to alarm setpoint being calculated for each discharge permit and is dependent upon flow rate of discharge.

Normal Effluent Release Monitor Classification Thresholds		
Monitor	Alert	UE
Auxiliary Building		
R-13 Aux. Bldg. Vent Exhaust	2.61E+07 cpm	2.61E+05 cpm
R-14 Aux. Bldg. Vent Exhaust	2.62E+07 cpm	2.62E+05 cpm
Reactor Building		
R-12 Containment Gas	4.41E+07 cpm	4.41E+05 cpm
R-21 Containment Vent	4.40E+07 cpm	4.40E+05 cpm
Liquid Radwaste		
R-18 Waste Disposal System Liquid	200 X Calculated ODCM Setpoint	2 X Calculated ODCM Setpoint

RA2.2

Based upon a formal calculation performed at KNPP, water level in Refueling Cavity was corrected to 50% Refueling Cavity level

RA3.2

NMC EAL Peer Group discussed the setpoint associated with this EAL and agreed that the method of determining radiation level value should be consistent but the stay times would be site specific. Per discussions with the KNPP Operation Group, it was decided to use 30 minutes (was 15 minutes) as a stay time and therefore changed radiation level value to 6 R/hr from 12 R/hr.

Tab 3 Cold Shutdown / Refueling System Malfunction

CU5

Based upon NRC feedback to the NMC and Challenge Board, it would not be acceptable to delete NEI 99-01 CU5 EAL #1. NEI 99-01 CU5 EAL #1 states, "(Site-specific) radiation monitor readings indicate fuel clad degradation greater than Technical Specification allowable limit". KNPP added an EAL to CU5.1 (CU5.1) to address the NEI EAL. The added KNPP EAL uses R-9, Letdown Radiation Monitor to detect the failed fuel clad at a value of 2.0 R/hr.

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KNPL 2004-0022

October 7, 2004

Paul Schmidt, DHFS-Radiation Protection Section
1 West Wilson Street
P.O. Box 2659
Madison, WI 53701-2659

Dear Paul:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs) at the meeting conducted on Thursday, October 7, 2004, held at Stevens Point.

The purpose of the meeting was to discuss the difference between the current NUREG-0654 scheme EALs and proposed conversion to NEI 99-01, Rev. 4 scheme.

The NRC approved the current Kewaunee Nuclear Plant EALs in their 1982 Safety Evaluation Report. The changes we discussed today will incorporate the NEI 99-01 EAL scheme into the Kewaunee Nuclear Plant EALs. The NRC endorsed the NEI 99-01, Rev. 4 scheme via Reg. Guide 1.101, Rev. 4, July 2003.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

Please review the Kewaunee EAL Bases document by October 21, 2004. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8117.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie A. Coleman

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Paul Schmidt

Paul Schmidt, DHFS-Radiation Protection Section

10-12-04
Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

KNPL 2004-0026

October 21, 2004

Paul Schmidt, DHFS-Radiation Protection Section
1 West Wilson Street
P.O. Box 2659
Madison, WI 53701-2659

Dear Paul:

PROPOSED CONVERSION OF EMERGENCY ACTION LEVEL SCHEME

Thank you for your time this morning to discuss the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs). Listed in Attachment 1 are the significant changes made to the KNPP NEI 99-01 EAL scheme since our meeting on October 7, 2004. The changes are based upon feedback from the NRC, Challenge Board, Plant Operation Review Committee and NMC Peer Group.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

Based on the discussion we this morning, I understand that you find the changes acceptable and concur with Kewaunee's additional changes to convert to NEI 99-01, Rev. 4 scheme. If you have any questions or comments during your review, please contact John Egdorf at (920) 776-2141. To document your agreement with these changes, please sign below and fax this document back to me. My fax number is (920) 388-8675.

Thank you again for your time. If you have any questions, please call me at (920) 388-8719.

Sincerely,

Jerrie Coleman
Emergency Preparedness Manager Kewaunee

Paul Schmidt — 10-22-04

Paul Schmidt, DHFS-Radiation Protection Section — Date

cc: Tom Coutu, Site Vice President, Kewaunee Nuclear Plant
Jerry Riste, Licensing, Kewaunee Nuclear Plant

STATE OF WISCONSIN \ DEPARTMENT OF MILITARY AFFAIRS
WISCONSIN EMERGENCY MANAGEMENT

2400 WRIGHT STREET
P.O. BOX 7865
MADISON, WISCONSIN 53708-7865

October 21, 2004

Ms. Jerrie Coleman
Emergency Preparedness Manager
Kewaunee Nuclear Power Plant
N490 Hwy. 42
Kewaunee, WI 54216-9511

Dear Ms. Coleman,

Re: October 7 Steven's Point meeting and October 21 conference call regarding EAL changes for Kewaunee Nuclear Power Plant.

The purpose of the meetings was to discuss the difference between the current NUREG-0654 scheme EALs and the proposed conversion to NEI 99-01, Rev. 4 scheme and changes made since the October 7 meeting based on feedback from the NRC, Challenge Board, Plant Operation Review Committee and the NMC Peer Group.

The NRC approved the current Kewaunee Nuclear Plant EALs in their 1982 Safety Evaluation Report. The changes we discussed today will incorporate the NEI 99-01 EAL scheme into the Kewaunee Nuclear Plant EALs. The NRC endorsed the NEI 99-01, Rev. 4 scheme via Reg. Guide 1.101, Rev. 4, July 2003.

Bob Host participated in the October 7 meeting discussion and both Bob Host and Teri Engelhart participated in the October 21 conference call discussion regarding the proposed changes to Kewaunee Nuclear Plant Emergency Action Levels (EALs).

The changes discussed will be included in the Kewaunee Nuclear Power Plant Emergency Plan upon approval by the Nuclear Regulatory Commission.

10 CFR 50, Appendix E, states "...emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC."

I have reviewed the changes and concur with Kewaunee's plan to implement these changes.

If you have any questions or if I can be of further assistance please contact me.

Sincerely,

William Clare

William Clare
Planning Section Supervisor

cc: Johnnie Smith, WEM Administrator
Tom Coutu, Site Vice President, Kewaunce Nuclear Plant
Jerry Riste, Licensing, Kewaunce Nuclear Plant