



Kewaunee Nuclear Power Plant  
 N490, State Highway 42  
 Kewaunee, WI 54216-9511  
 920-388-2560



Operated by  
 Nuclear Management Company, LLC

April 23, 2001

10 CFR 50.36

U.S. Nuclear Regulatory Commission  
 Attention: Document Control Desk  
 Washington, D.C. 20555

Ladies/Gentlemen:

DOCKET 50-305  
 OPERATING LICENSE DPR-43  
 KEWAUNEE NUCLEAR POWER PLANT  
BASES REVISIONS TO THE KEWAUNEE NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS

Nuclear Management Company (NMC), licensee for the Kewaunee Nuclear Power Plant (KNPP), hereby submits a revision to the Bases for the Technical Specifications (TS). The changes are as follows:

- Pages TS B2.3-1, TS B3.1-13, TS B3.1-14, TS B3.4-4, TS B3.4-5, TS B3.5-3, TS B3.5-4, TS B3.7-1, TS B3.8-2 and TS B3.8-3: Modified, added, and deleted KNPP Updated Safety Analysis Report (USAR) references contained in Bases Footnotes, to reflect approved USAR Revision 16.
- Page TS B3.4-4: Updated the upper limit for allowable primary to secondary leakage in a steam generator faulted loop from 9.0 gpm to 3.69 gpm. This was a correction identified in Westinghouse advisory letter WNSAL 2000-004, evaluated and accepted in KNPP OEA 2000-034 and 50.59 Safety Evaluation No. 00-22.
- Page TS B3.6-2: Updated the required refueling boron concentration from 2100 ppm to 2200 ppm, to reflect approved TS Amendment No. 147.
- Page TS B3.7-1: Clarified Bases discussion regarding plant auxiliary power and 125-V d-c power by specifying supplies are safeguards.

These changes have been screened for evaluation pursuant to the requirements of 10 CFR 50.59 in accordance with approved KNPP procedures and were determined to be acceptable.

A001

Document Control Desk

April 23, 2001

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Attached is a copy of the revised TS Bases pages for your controlled Technical Specifications.

Sincerely,

A handwritten signature in black ink, appearing to read "Tom Webb", written over a horizontal line.

Thomas J. Webb

Site Licensing Director

PRR

Attachments

cc - NRC Regional Administrator  
NRC Resident Inspector  
Electric Division, PSCW

## BASIS

### 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.<sup>(1)</sup>

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.<sup>(2)</sup>

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 RCCA's fall into the core. The circuits are independent and assure 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.<sup>(3)</sup> The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error<sup>(2)</sup> and transient level overshoot before the reactor trips.

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<sup>(1)</sup>USAR Section 14.1.1

<sup>(2)</sup>USAR Section 14.0

<sup>(3)</sup>USAR Section 14.3.1

### 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  $\leq 60\%$  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.<sup>(19)</sup>

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 TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.<sup>(19)</sup>

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

The requirement that the reactor is not to be made critical when the moderator coefficient is  $> 5.0$  pcm/ $^{\circ}$ F 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 5.0 pcm/ $^{\circ}$ F 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<sup>(19)</sup> and the small integrated  $\Delta k/k$  would limit the magnitude of a power excursion resulting from a reduction in moderator density.

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<sup>(19)</sup>USAR Section 3.2.1

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  $\leq 5.0$  pcm/ $^{\circ}$ F 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  $\leq 5.0$  pcm/ $^{\circ}$ F. 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  $-8.0$  pcm/ $^{\circ}$ F 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<sup>(20) (21)</sup> 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  $-8.0$  pcm/ $^{\circ}$ F, 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.

In the event that the limits of TS 3.1.f.3 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 in TS 3.1.f.3. 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.

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

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

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  $\geq$  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 are applied. 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.

#### Condensate Storage Tank (CST)(TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.<sup>(1)</sup>

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.

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<sup>(1)</sup>USAR Section 8.2.4

#### Secondary Activity Limits (TS 3.4.d)

An evaluation was performed to determine the maximum permissible steam generator primary-to-secondary leak rate during a steam line break event. The evaluation considered both a preaccident and accident initiated iodine spike for offsite dose and control room operator dose. The results of the evaluation show that the control room operator dose with an accident initiated spike yields the limiting leak rate. This evaluation was based on a 30 REM thyroid dose and initial primary and secondary coolant iodine activity levels of 0.20  $\mu\text{Ci/gm}$  and 0.1  $\mu\text{Ci/cc}$  DOSE EQUIVALENT I-131 respectively. A leak rate of 3.69 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the steam generator faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm (150 gpd per TS 3.1.d.2), the standard operating leakage limit applied for the tube support plate voltage-based plugging criteria specified in TS 4.2.b.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.<sup>(4)</sup> 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

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<sup>(4)</sup>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.

Accident analysis assumes a charcoal adsorber efficiency of 90%.<sup>(1)</sup> 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.

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. 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.

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<sup>(2)</sup> or more if the Containment System is to be disabled with the reactor pressure vessel open.

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<sup>(1)</sup>USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

<sup>(2)</sup>USAR Table 3.2-1

## BASIS

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<sup>(1)</sup>: 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.<sup>(2)</sup> 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.<sup>(3)</sup>

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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<sup>(1)</sup>USAR Figure 8.2-1 and 8.2-2

<sup>(2)</sup>USAR Section 8.2.3

<sup>(3)</sup>USAR Section 8.2.2 and 8.2.3

leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

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

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 100 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 100 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%.<sup>(3)</sup> 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.

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<sup>(3)</sup>USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"