

ATTACHMENT I

To

Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC)

Dated

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BASES - Operational Components (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 1 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 1 will keep 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)

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

When the average reactor coolant temperature is $\leq 350^{\circ}\text{F}$ 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 $\leq 200^{\circ}\text{F}$, 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.

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, 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.

(1) USAR Section 7.2.2

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 relief valve become inoperable.

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 assure the ability to maintain pressurizer pressure. Annual 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 noncondensable gases from the high points of the RCS to assure 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.

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

(3) Hot functional test (PT-RC-31)

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)

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).

(4) Letter from E. R. Mathews to S. A. Varga dated 5/21/82

(5) ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

(6) Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

(7) WCAP-13229, "Heatup and Cooldown Limit Curves for Normal Operation for Kewaunee," M. A. Ramirez and J. M. Chicots, March 1992 (Westinghouse Proprietary Class 3)

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:

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

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.

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. 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. At any given temperature, the allowable pressure is taken to be the lesser of the three 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.

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. 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. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan (8)(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 9878 (10), weld metal Charpy test specimens from Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (235°F).

(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) ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

(10) 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.

The results of Irradiation Capsules V, R, and P analyses are presented in WCAP 8908⁽¹¹⁾, WCAP 9878, and WCAP-12020⁽¹²⁾, 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 20 effective full-power years.

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 assure 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.

Maximum Coolant Activity (TS 3.1.c)

This specification is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases⁽¹³⁾ which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.⁽¹³⁾

⁽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 P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

⁽¹³⁾ USAR Section 14.2.4

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose, rem = 1/2 \left[\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10}) (1.33 \times 10^{-11}) \right]$$

Where: \bar{E} = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m³)

$\bar{E}A$ = 91 Mev Ci/dis m³ (the maximum per this specification)

$\frac{X}{Q}$ = 2.9×10^{-4} sec/m³, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m³, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

Leakage of Reactor Coolant (TS 3.1.d) ⁽¹⁴⁾

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

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

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

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

Twelve 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.

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.

⁽¹⁴⁾ USAR Sections 6.5, 11.2.3, 14.2.4

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.

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 assured under all operating conditions. (TS)

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 (16). Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

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

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range. (17)(18)

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

(16) USAR Section 9.2

(17) USAR Table 3.2-1

(18) USAR Figure 3.2-8

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient⁽¹⁹⁾ and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

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

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.⁽²⁰⁾

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.

⁽¹⁹⁾ USAR Figure 3.2-9

⁽²⁰⁾ USAR Table 3.2-1