NSP

Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

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U S Nuclear Regulatory Commission
Attn: Document Control Desk
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT

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Technical Specification Bases Changes
Correction of References and Miscellaneous Errors

In accordance with the Prairie Island Bases Control Program we have changed several Technical Specification Bases pages which are attached for your use.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Please contact Jeff Kivi (651-388-1121) if you have any questions related to this letter.

Joel P. Sorensen

Joel P. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant

Attachments

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

A001

B. Reactor Coolant System Pressure Safety Limit

Bases

The reactor coolant system (Reference 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 assured. The maximum transient pressure allowable in the reactor coolant system 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 USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established (Reference 2).

The nominal settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to assure that the pressure never reaches the reactor coolant system pressure safety limit.

In addition, the reactor coolant system safety valves (Reference 3) are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, the reactor coolant system was hydrotested at 3107 psig prior to initial operation (Reference 4).

References

1. USAR, Section 4.1
2. USAR, Section 14.4
3. USAR, Section 4.4.3.2
4. USAR, Section 4.1

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases

The power range high flux reactor trips (low set point) provides redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis (Reference 1).

The intermediate and source range high flux reactor trips provide additional protection against uncontrolled startup excursions. As power level increases, during startup, these trips are manually blocked to prevent unnecessary plant trips.

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

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor 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 reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident (Reference 3).

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 4 seconds) Reference 1), 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 (Reference 2), is always below the core safety limits shown on Figure TS.2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced (Reference 5).

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 axial power distribution, 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 set points meet this requirement and include allowance for instrument errors (Reference 2).

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The overpower and overtemperature protection setpoints include the effects of fuel densification on core safety limits.

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident:

a. Low reactor coolant flow

b. Low voltage on pump power supply bus

c. Pump circuit breaker opening (low frequency on pump power supply bus opens pump circuit breaker)

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 one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis (Reference 7). The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation.

The reactor coolant pump bus undervoltage trip is a direct reactor trip (not a reactor coolant pump circuit breaker trip) which protects the core against DNB in the event of a loss of power to the reactor coolant pumps.

The reactor coolant pump breaker reactor trip is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the reactor coolant pump breaker reactor trip is the frequency set point, $\geq$58.2 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error (Reference 2) and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system (Reference 8).

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of RATED THERMAL POWER.

The other reactor trips specified in 2.3.A.3. above provide additional protection. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection against DNB for control rod drop accidents. Most rod drop events will cause a sufficiently rapid decrease in power to trip the reactor on the negative power range rate trip signal. Any rod drop events which do not insert enough reactivity to cause a trip are analyzed to ensure that the core does not experience DNB. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

References

1. USAR, Section 14.4
2. USAR, Section 14.3
3. USAR, Section 14.6
4. deleted
5. USAR, Section 7.4.1, 7.2
6. deleted
7. USAR, Section 14.4.8
8. USAR, Section 14.4.10

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

The OPERABILITY of the low temperature overpressure protection system is determined on the basis of their being capable of performing the function to mitigate an overpressure event during low temperature operation. OPERABILITY of a low temperature overpressure protection system PORV requires that the low pressure set point has been selected (enabled), the upstream isolation valve is open and the backup air supply is charged.

The low temperature overpressure protection system is designed to perform its function in the event of a single failure and is designed to meet the requirements of IEEE-279. The backup air supply provides sufficient air to operate the PORVs following a letdown isolation with one charging pump in operation for a period of ten minutes after receipt of the overpressure alarm. These specifications provide assurance that the low temperature overpressure protection system will perform its intended function.

The reactor coolant vent system is provided to exhaust noncondensible gases from the reactor coolant system that could inhibit natural circulation core cooling. 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. An inoperable vent path valve is defined as a valve which cannot be opened or whose position is unknown.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow of the reactor coolant makeup system.

References

1. USAR, Section 14.4.9.

2. Testimony by J Knight in the Prairie Island Public Hearing on January 28, 1975.

3. NSP Letter to USNRC, "Reactor Vessel Overpressurization", dated July 22, 1977.

3.1 REACTOR COOLANT SYSTEM

Bases continued

Specification 3.1.C.3 specifies actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.2, 7.5.3
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

3.1 REACTOR COOLANT SYSTEM

Bases continued

F. Isothermal Temperature Coefficient (ITC)

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during low power PHYSICS TESTS in order to verify analytical prediction. The units of the isothermal temperature coefficient are pcm/°F, where 1pcm = 1×10^{-5} $\Delta k/k$,

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive isothermal temperature coefficient could exist at beginning of cycle (BOC). Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F requirements are waived during low power PHYSICS TESTS to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these PHYSICS TESTS. In addition, the strong negative Doppler coefficient (Reference 1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

References:

1. FSAR Figure 3.2-10

3.3 ENGINEERED SAFETY FEATURES

Bases continued

The containment cooling function is provided by two independent systems: containment fan cooler (CFC) and containment spray. The CFC system consists of two separate trains with each train consisting of two fan coil units. The containment spray system consists of two independent trains, except that each train is supplied sodium hydroxide from a single, common spray additive tank. During normal operation, two CFC trains are utilized to remove heat lost from equipment and piping within the containment. In the event of the Design Basis Accident, one containment fan cooler unit plus one containment spray pump will provide sufficient cooling to reduce containment pressure and maintain off-site and control room doses within regulatory limits (Reference 4). One CFC train is permitted to be inoperable during POWER OPERATION. This is an abnormal operating situation, in that plant operating procedures require that inoperable CFC units be repaired as soon as practical. However, because of the difficulty of access to make repairs, it is important on occasion to be able to operate temporarily with only one CFC train. One CFC train can provide adequate cooling for normal operation when the CFC units are cooled by the chilled water system (Reference 3). Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit, either following a loss-of-coolant accident or during normal plant shutdown. The four pumps of the two-unit facility can be cross connected as necessary to accommodate temporary outage of the pump. If, during the post-accident phase, the component cooling water supply were lost, core and containment cooling could be maintained until repairs were effected (Reference 5).

Cooling water can be supplied by either of the two horizontal motor-driven pumps, by a safeguards motor-driven pump or by either of two safeguards diesel-driven pumps. (Reference 6). Operation of a single cooling water pump provides sufficient cooling in one unit during the injection and recirculation phases of a postulated loss-of coolant accident plus sufficient cooling to maintain the second unit in a hot standby condition.

TS.3.3.D.1.a assures that an automatic Safety Injection signal to the cooling water header isolation valves will not align both OPERABLE safeguards pumps to the same safeguards train.

TS.3.3.D.1.a also assures that 121 cooling water pump is aligned to provide cooling water to the same train as the train from which it is being powered (e.g., if 121 cooling water pump is aligned to Train B cooling water header, it needs to be powered from Bus 26 and, ultimately, Diesel Generator D6 in the event of a loss of offsite power). Otherwise, the single failure of a diesel generator could leave one train of engineered safety features without power and the other train without cooling water.

The minimum fuel supply of 19,000 gallons wil supply one diesel-driven cooling water pump for 14 days. Note that the 19,000 gallon requirement is included in the 70,000 gallon total diesel fuel oil requirement of Specification 3.7.A.5 for Unit 1.

3.3 ENGINEERED SAFETY FEATURES

Bases continued

The Safeguards Traveling Screens and Emergency Intake Line together with the Intake Canal and Traveling Screens are designed to provide a supply of screened cooling water to the safeguards bay in the event of a design basis earthquake. The design basis earthquake is postulated to destroy Lock & Dam No. 3. River level decreases over time, making the Intake Canal unavailable. The Safeguards Traveling Screens and Emergency Cooling Water Supply line provide an alternate supply of water to the Safeguards Bay, which contains the two diesel driven and the one vertical motor driven cooling water pumps. Their normal supply is from the Circ Water Bay thru one of two sluice gates. Either one of the two sluice gates or one of the two Safeguards Traveling Screens will adequately supply any of the three cooling water pumps. The Safeguards Traveling Screens are not considered part of the "engineered safety features associated with the operable diesel-driven cooling water pump" for determination of operability of diesel-driven cooling water pumps.

The component cooling water system and the cooling water system provide water for cooling components used in normal operation, such as turbine generator components, and reactor auxiliary components in addition to supplying water for accident functions. These systems are designed to automatically provide two separate redundant paths in each system following an accident. Each redundant path is capable of cooling required components in the unit having the accident and in the operating unit.

There are several manual valves and manually-controlled motor-operated valves in the engineered safety feature systems that could, if one valve is improperly positioned, prevent the required injection of emergency coolant (Reference 7). These valves are used only when the reactor is subcritical and there is adequate time for actuation by the reactor operator. To ensure that the manual valve alignment is appropriate for safety injection during power operation, these valves are tagged and the valve position will be changed only under direct administrative control. For the motor-operated valves, the motor control center supply breaker is physically locked in the open position to ensure that a single failure in the actuation circuit or power supply would not move the valve.

References

1. USAR, Section 3.3.2
2. USAR, Section 14.6
3. USAR, Section 6.3.2
4. USAR, Section 6.3
5. USAR, Section 10.4.2
6. USAR, Section 10.4.1
7. USAR, Figure 6.2-1A
   USAR, Figure 6.2-1B
   USAR, Figure 6.2-5
   USAR, Figure 10.2-11

3.5 INSTRUMENTATION SYSTEM

Bases continued

Instrument Operating Conditions (continued)

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 a concurrent channel.

References

1. USAR, Section 7.4.1, 7.4.2
2. USAR, Section 14.6
3. USAR, Section 14.5.5
4. FSAR, Appendix I

3.6 CONTAINMENT SYSTEM

Bases continued

The 2 psig 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 (Reference 1).

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

The containment has a nil ductility transition temperature of 0°F. Specifying a minimum temperature of 30°F will provide adequate margin above NDTT during power operation when containment is required.

The conservative calculation of off-site doses for the loss of coolant accident (References 1, 4) is based on an initial shield building annulus air temperature of 60°F and an initial containment vessel air temperature of 104°F. The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified 44°F temperature difference is consistent with the LOCA accident analysis (Reference 4).

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS.4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS.3.6.E.2). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5% to 0.25%/day would offset the increased leakage (Reference 3).

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

Air locks are provided with two doors, each of which is designed to seal against the maximum containment pressure resulting from the limiting DBA. Should an air lock become inoperable as a result of an inoperable air lock door or an inoperable door interlock, power operation may continue provided that at least one OPERABLE air lock door is closed. With an air lock door inoperable, access through the closed or locked OPERABLE door is only permitted for repair of inoperable air lock equipment.

3.6 CONTAINMENT SYSTEM

Bases continued

OPERABILITY of air locks is required to ensure that CONTAINMENT INTEGRITY is maintained. Should an air lock become inoperable for reasons other than an inoperable air lock door, the air lock leak tight integrity must be restored within 24 hours or actions must be taken to place the unit in a condition for which CONTAINMENT INTEGRITY is not required.

References

1. USAR, Section 5
2. USAR, Section 10.3.4 and Appendix G
3. Letter to NSP dated November 29, 1973
4. Letter to NSP dated September 16, 1974

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

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

Operation of each train of the system for 10 hours every month will demonstrate OPERABILITY of the system and remove excessive moisture which may build up on the adsorber.

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.

The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

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

A minimum containment shell temperature of 30°F has been specified to provide assurance that an adequate margin above NDTT exists. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in COLD SHUTDOWN. Requiring containment shell temperature to be verified to be above 30°F prior to plant heatup from COLD SHUTDOWN provides assurance that this temperature is above NDTT prior to establishing conditions requiring CONTAINMENT INTEGRITY (Reference 7).

A maximum temperature differential between the average containment and annulus air temperatures of 44°F has been specified to provide assurance that offsite doses in the event of an accident remain below those calculated in the USAR. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in COLD SHUTDOWN. Requiring this temperature differential to be verified to be less than 44°F prior to plant heatup from COLD SHUTDOWN provides assurance that this parameter is within acceptable limits prior to establishing conditions requiring CONTAINMENT INTEGRITY (Reference 7).

References

1. USAR, Section 5 and Appendix K
2. USAR, Section 14 and Appendix G
3. Safety Evaluation Report, Sections 6.2 and 15.0
4. USAR, Section 6
5. USAR, Section 5
6. Letter to NSP from AEC dated November 29, 1973
7. "Prairie Island Containment Systems Analyses," dated April 9, 1976

4.7 MAIN STEAM ISOLATION VALVES

Bases

The main steam isolation valves serve to limit the cooldown rate of the reactor coolant system and the reactivity insertion that could result from a main steam break incident (Reference 1). Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds is selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis (Reference 1).

Reference

1. USAR, Section 14.5.5

4.9 REACTIVITY ANOMALIES

Bases

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.

Reference

USAR, Section 3.7.1