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US Nuclear Regulatory Commission
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Overview of Revision to Main Steam Line Break Analysis

Prairie Island currently must maintain 20,000 ppm in the Boric Acid Storage Tanks, which requires heat tracing to prevent boron precipitation. In order to eliminate the need for maintaining this high boric acid concentration, NSP is revising the analysis of the Main Steam Line Break event to demonstrate that Safety Injection supplied by the Refueling Water Storage Tank water at a boron concentration of 2,500 ppm is acceptable in mitigating the effects of the accident.

A preliminary overview of the revised Prairie Island Steam Line Break Analysis for the reduction of boric acid storage tank boron concentration is attached for the information of the NRC Staff.

The attached information does not contain any new NRC commitments. Please contact Gene Eckholt (612) 388-1121 ext 4663 if you have any questions related to the information we have provided.

Eugene Eckholt
for Roger O Anderson
Director
Licensing and Management Issues

c: Regional Administrator - III, NRC
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Attachments: Overview of a Revised Prairie Island Main Steam Line Break Analysis for the Reduction of Boric Acid Storage Tank Boron Concentration.

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Overview of a Revised Prairie Island Main Steam Line Break Analysis for the Reduction of Boric Acid Storage Tank Boron Concentration

January 1994

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Introduction

Prairie Island currently must maintain 20,000 ppm boron in the Boric Acid Storage Tanks, which requires heat tracing to prevent boron precipitation. The Boric Acid Storage Tanks currently are the primary source of borated water for Safety Injection. NRC Generic Letter 85-16 has acknowledged the burdens that high boron concentrations have placed on operations. In addition, Generic Letter 85-16 recognizes that high boron concentrations may not be necessary to provide enough negative reactivity in the mitigation of operational transients. Due to the safety risks inherent in the present system, as well as the additional operating costs, NSP will be submitting the results from an analysis of the Main Steam Line Break event that demonstrates that Safety Injection supplied by Refueling Water Storage Tank water at a boron concentration of 2,500 ppm is acceptable in mitigating the effects of the accident. This will eliminate the need to maintain the Boric Acid Storage Tanks as safety grade components. It will also allow the concentration in the Tanks to be lowered to a level that will not require heat tracing since there will be no possibility of the boric acid precipitating out of solution.

Description of Prairie Island

Prairie Island is a two loop Westinghouse design PWR. A basic description of the plant is given in Table 1.

NSP Methodology

NSP has a fully approved methodology for performing Prairie Island core designs and non-LOCA reload safety evaluations through the submittal of two topicals, designated as the Reactor Physics Topical and the Safety Analysis Topical (References 1 and 2). NSP has been performing these analyses in-house on a 10CFR50.59 basis since 1983. Westinghouse Electric holds the LOCA analysis of record for Prairie Island.

NSP will submit the results of an analysis of the Main Steam Line Break event using a new methodology that better predicts mass and energy releases and containment response. The results will show that the acceptance criteria for the Main Steam Line Break can be met with Safety Injection supplied by Refueling Water Storage Tank water at a boron concentration of 2,500 ppm. As always, NSP will continue to evaluate all accident scenarios and plant modifications for each cycle on a 10CFR50.59 basis through the reload safety evaluation.

Computer Codes

Analysis of the Main Steam Line Break event is a two step process. The DYNODE-P code (Reference 3) is first used to calculate the mass and energy released as a function of time. The releases are then used as input to the CONTEMPT-LT code (Reference 4) to calculate containment pressure and temperature as a function of time.

Both the DYNODE-P and CONTEMPT-LT codes have been previously reviewed and approved by the NRC for use by NSP in reload safety evaluations for Prairie Island (References 1 and 2).

Main Steam Line Break Analysis for Containment Response

In the current methodology (Reference 2) for main steam line break events, the DYNODE-P and CONTEMPT-LT codes were used to benchmark to the FSAR containment response. In this benchmark case, the liquid entrainment was accounted for by reducing the mass and energy releases by a factor of 0.85. For each new cycle the containment response is not reanalyzed, if the mass and energy releases as calculated by DYNODE-P are less than the FSAR values the event is considered to be bounded.

In the revised methodology, the mass and energy releases are calculated using DYNODE-P and CONTEMPT-LT is used to calculate the containment response for each cycle.

The important systems modeled in the new Main Steam Line Break analysis are described below. Differences between the current methodology (Reference 2) and the revised methodology are noted in the text that follows.

Kinetics:

In the current methodology (Reference 2), the DYNODE-P code is used with a point kinetics model of a Prairie Island core. Conservative reactor physics parameters are used in the analyses in accordance with the methods described in the Reactor Physics and Safety Analysis Topicals. The physics parameters are calculated for end of life conditions with the most reactive control rod stuck out of the core. The core is also assumed to have decay heat generation calculated from the ANSI/ANS 1971 decay heat standard.

The only difference for the revised methodology is that the initial decay heat is calculated using the ANSI/ANS 1979 standard.

Heat Structures:

In the revised methodology, the reactor vessel and primary system piping metal heat structures are modeled. The steam generator metal heat structures are also modeled. It is more accurate and conservative to model metal heat structures.

The current methodology (Reference 2) does not include heat structures.

Steam Generator Model:

In the revised methodology, the steam generator liquid level is set conservatively high to increase the mass available for blowdown. Steam in the unisolated portions of the steam line and feedwater line is also available for blowdown. Reverse heat transfer is maximized in the intact loop. No credit is taken for tube uncovery.

In the current methodology (Reference 2), unisolated steam and feedwater lines are not modeled. However, high SG liquid level and tube uncovery assumptions remain the same. No credit is taken for tube plugging.

Safety System Model:

In the revised methodology, conservative delay times and setpoints are used for safety system signals and actuations. All relevant safety systems are modeled including Reactor Trip, Turbine Trip, Feedwater Pump Trip, Containment Isolation, MSIV closure, Safety Injection, Containment Sprays, and Fan Coil Units. The sequence of safety signal actuations are based on calculated steam flows and pressures, primary coolant temperatures and pressures, and containment pressure.

In the current methodology (Reference 2), all actuations and signals are based on an assumed SI signal time of 5 seconds. This includes feedwater isolation and pump trip, and MSIV closure. The Containment Sprays and FCU actuation are not necessary since CONTEMPT-LT is not utilized. Reactor trip is not important since all transients are initiated from zero power.

Feedwater System Model:

In the revised methodology, the Advanced Digital Feedwater Control System is modeled and the flow split between the intact and broken loops is calculated. The main feedwater pumps run at full speed and begin to coast down on the Feedwater Pump Trip signal. The condensate pumps are not tripped during the event. The feedwater regulator and bypass valves are kept full open until the time that complete isolation would occur, they are then closed instantaneously.

The current methodology (Reference 2) does not model the flow split. Instead, full feedwater flow is equally split between the two steam generators. This method reflects a limitation in previous versions of DYNODE-P that were unable to calculate the flow split based on system pressure. Full flow is assumed until the SI-induced MSIV closure. The bypass and regulator valves are assumed to close instantaneously at this time as well.

Auxiliary Feedwater System Model:

In the revised methodology, the auxiliary feedwater flow split between the intact and broken loops is calculated. The auxiliary feedwater flow is initiated prior to the time of actual initiation and a conservatively high enthalpy is used for the properties of this flow.

The flow split is not modeled with the current methodology (Reference 2). Instead, auxiliary feedwater is assumed to supply only the broken loop SG at runout conditions.

Blowdown and Entrainment Model:

In the revised methodology, the pressure balance line is modeled to allow reverse steam flow from the intact to the broken loop. Blowdown is calculated with the effect of containment backpressure taken into account. The Moody model for critical flow is used with no credit taken for friction losses in the steam generator or associated piping. Liquid entrainment in the blowdown was conservatively modeled in accordance with Westinghouse report WCAP-8822 (Reference 5).

With the current methodology (Reference 2), the pressure balance line is modeled and the Moody critical flow model is used to predict the break flow. Containment backpressure and liquid entrainment are not specifically modeled.

Containment Model:

In the revised methodology, a two compartment model is used with all metal and concrete heat structures included. Heat transfer coefficients for the heat structures are calculated using the NRC approved procedure described in the Safety Analysis Topical (Reference 2) for both turbulent and stagnant blowdown conditions. For large breaks credit was taken for condensate revaporization, while for smaller breaks this was not modeled.

With the current methodology (Reference 2), all assumptions remain the same as above except that credit is not taken for condensate revaporization.

Break Spectrum:

In the revised methodology, analyses were done for a range of power levels from zero power to 102% power. Double ended ruptures and split breaks were modeled over a range of break areas to determine the effects of entrainment and dry steam blowdown, and to provide a bounding analysis. The most limiting single failures modeled were the following; failure of a feedwater regulator valve, failure of a MSIV, and failure of one train of containment safeguards equipment. Consideration was also given to events with and without the availability of offsite power.

With the current methodology (Reference 2), the only MSLB transient that is modeled initiates from EOC and HZP. This break is a double ended rupture at the steam generator exit nozzle. One safeguards train is assumed to be failed and entrainment is not included.

Conclusion

NSP will be submitting a request to allow suction for Safety injection to be taken directly from the Refueling Water Storage Tank at a boron concentration of 2,500 ppm. Current plans are to submit this request in the Spring of 1994. The Main Steam Line Break analysis supporting the change will be included in the License Amendment Package that will be submitted for the application for amendment to the operating license.

TABLE 1
Prairie Island General Description

PRIMARY SYSTEMS	
Reactor Size & Type:	Westinghouse 2-Loop PWR with Upper Plenum Injection
Rated Power:	1650 MWth
Rated Pressure:	2250 psia
RCS Design Flow:	68.2×10^6 lb/hr
Coolant Tavg:	
HFP	560°F
HZP	547°F
Number of fuel Assemblies:	121
Type of Fuel:	14x14 Westinghouse Vantage+
Current nominal enrichment:	4.95 w/o U-235
Burnable Absorber:	Gadolina with up to 8 w/o
Safety Systems:	
SI	Flow to cold leg (nominal) or upper plenum
SI Pumps	2 pumps, suction from BAST (primary) or RWST (secondary)
BAST	2 tanks, 5000 gal, 20000 ppm
RWST	1 tank, 275000 gal, 2500 ppm
Fan Co[illegible] Units	4, two per train
Cont. S[illegible]	2 pumps, suction from RWST (Primary) or Cont. Sump (secondary)
SECONDARY SYSTEMS	
Steam Generators:	2 Westinghouse Model 51 U-tube
Secondary Pressure:	
HFP	750 psia
HZP	1000 psia
SG Feedwater:	
Main FW Pumps	2
Aux FW Pumps	2 (steam driven and motor driven)

References

Reference 1 "Qualification of Reactor Physics Methods for Application to PI Units", NSPNAD-8101-A, Rev. 1, December 1982.

Reference 2 "Reload Safety Evaluation Methods for Application to PI Units", NSPNAD-8102-A, Rev. 5, March 1987.

Reference 3 Kern, R.C., "DYNODE-P: A Reactor Core Transient Simulator for Pressurized Water Reactors", NSP Version 91204 Rev. 0, October 1991.

Reference 4 Wheat, L.L., et. al., "CONTEMPT-LT: A Computer Program for Predicting Containment Pressure - Temperature Response to a Loss of Coolant Accident", NSP Version 80240, October 1982.

Reference 5 Land, R.E., "Mass and Energy Releases Following a Steam Line Rupture", Westinghouse WCAP-8822, September 1976.

Osborne, M.P., et. al., "Supplement 1- Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture", WCAP-8822-S1-P-A, January 1985.

Butler, J.C., et. al., "Supplement 2- Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs", WCAP-8822-S2-P-A, September 1986.