NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401-1927
Telephone (612) 330-5500

December 1, 1989

Response to
Generic Letter 89-21

Director
Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42
50-306 DPR-60

Response to Generic Letter 89-21
Unresolved Safety Issues Implementation Status

Generic Letter 89-21 requested that we provide the implementation status of Unresolved Safety Issues. The purpose of this letter is to provide that status. Enclosure 1 is a list showing the status of each Unresolved Safety Issue. Enclosure 2 contains remarks on Unresolved Safety Issues.

Please contact us if further information is required.

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator - III, NRC
Sr Resident Inspector, NRC
Sr Project Manager, NRC
G Charnoff

Attachments

A012
1/1

ENCLOSURE 1 PRAIRIE ISLAND

UNRESOLVED SAFETY ISSUES FOR WHICH A FINAL TECHNICAL RESOLUTION HAS BEEN ACHIEVED

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-1	Water Hammer	SECY 84-119 NUREG-0927, Rev. 1 NUREG-0993, Rev. 1 NUREG-0737 Item I.A.2.3 SRP revisions	All	C	See Enclosure 2
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609 GL 84-04, GDC-4	PWR	C	See Enclosure 2
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844 SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	W-PWR	C	See Enclosure 2
A-4	CE Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	CE-PWR	NA	See Enclosure 2
A-5	B&W Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	B&W-PWR	NA	See Enclosure 2
A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	NA	See Enclosure 2

* C - COMPLETE
NC - NO CHANGES NECESSARY
NA - NOT APPLICABLE
I - INCOMPLETE
E - EVALUATING ACTIONS REQUIRED

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-7/ D-01	Mark I Long-Term Program	NUREG-0661 NUREG-0661 Suppl. 1 GL 79-57	Mark I-BWR	NA	See Enclosure 2
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808 NUREG-0487, Suppl. 1/2 NUREG-0802 SRP 6.2.1.1C GDC 16	Mark II-BWR	NA	See Enclosure 2
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4 10 CFR 50.62	All	I	See Enclosure 2
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619 Letter from DG Eisenhut dated 11/13/80 GL 81-11	BWR	NA	See Enclosure 2
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1 10 CFR 50.60/ 82-26	All	C	See Enclosure 2
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1 SRP Revision 5.3.4	PWR	NC	See Enclosure 2
A-17	Systems Interactions	Ltr: DeYoung to licensees - 9/72 NUREG-1174, NUREG-1229, NUREG/CR-3922, NUREG/CR-4261, NUREG/CR-4470, GL 89-18 (No requirements)	All	I	See Enclosure 2
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1 SRP 3.11 10 CFR 50.49 GL 82-09, GL 84-24 GL 85-15	All	C	See Enclosure 2

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees 8/76 NUREG-0224 NUREG-0371 SRP 5.2 GL 88-11	PWR	C	See Enclosure 2
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606 RG 1.113, RG 1.139 SRP 5.4.7	All OLs After 01/79	NA	See Enclosure 2
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612 SRP 9.1.5 GL 81-07, GL 83-42, GL 85-11 Letter from DG Eisenhut dated 12/22/80	All	C	See Enclosure 2
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802 NUREGs-0763,0783,0802 NUREG-0661 SRP 6.2.1.1.C	BWR	NA	See Enclosure 2
A-40	Seismic Design Criteria	SRP Revisions, NUREG/CR-4776, NUREG/CR-0054, NUREG/CR-3480, NUREG/CR-1582, NUREG/CR-1161, NUREG-1233, NUREG-4776 NUREG/CR-3905 NUREG/CR-5347 NUREG/CR-3509	All	I	See Enclosure 2
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1 NUREG-0313, Rev. 2 GL 81-03, GL 88-01	BWR	NA	See Enclosure 2

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-43	Containment Emergency Sump Performance	NUREG-0510, NUREG-0869, Rev. 1 NUREG-0897, RG 1.82 (Rev. 0), SRP 6.2.2 GL 85-22 (No requirements)	All	NA	See Enclosure 2
A-44	Station Blackout	RG 1.155 NUREG-1032 NUREG-1109 10 CFR 50.63	All	I	See Enclosure 2
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260 NUREG-1289 NUREG/CR-5230 SECY 88-260 (No requirements)	All	I	See Enclosure 2
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211/ GL 87-02, GL 87-03	All	I	See Enclosure 2
A-47	Safety Implication of Control Systems	NUREG-1217, NUREG-1218 GL 89-19	All	E	See Enclosure 2
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	10 CFR 50.44 SECY 89-122	All, except PWRs with large dry containments	NA	See Enclosure 2
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99 SECY 82-465 SECY 83-288 SECY 81-687 10 CFR 50.61/, GL 88-11	PWR	C	See Enclosure 2

Enclosure 2

USI Remarks

A-1 Water Hammer

The Prairie Island Operator Training Program covers water hammer events.

Prairie Island responded to NRC letter dated May 12, 1975 on May 29, 1975, September 2, 1975 and submitted a report on January 29, 1976 entitled, "Analysis of PWR Secondary System Fluid Flow Instability." Prairie Island responded to NRC letter dated September 2, 1977 on December 30, 1977. The December 30, 1977 submittal refers to the January 29, 1976 submittal. The NRC issued a Safety Evaluation Report on September 13, 1979.

A-2 Asymmetric Blowdown Loads in RCS

On October 24, 1984 Prairie Island submitted Westinghouse Report WCAP 10640, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Prairie Island Unit 1." On October 21, 1985 Prairie Island submitted Westinghouse Reports WCAP 10928 and 10930, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Bases for Prairie Island Unit 2" and "Toughness Criteria for Thermally Aged Cast Stainless Steel" respectively. The NRC issued a Safety Evaluation Report on December 22, 1986.

A-3, A-4, and A-5 Steam Generator Tube Integrity

Prairie Island provided steam generator inspection program information in response to Generic Letter 85-02, dated June 19, 1985.

A-6 Mark I Containment Short Term Program

This is a BWR issue.

A-7 Mark I Long Term Program

This is a BWR issue.

A-8 Mark II Containment Pool Dynamic Loads

This is a BWR issue.

A-9 ATWS per 10 CFR 50.62

The NRC issued a Safety Evaluation Report dated August 17, 1988 on the Prairie Island AMSAC design. Modification was completed in March 1989 for

Unit 2 and will be completed in February 1990 for Unit 1. NRC Inspection Report No. 88022 dated August 31, 1989 verified ATWS completion for Unit 2.

A-10 BWR Feedwater Nozzle Cracking

This is a BWR issue.

A-11 Reactor Vessel Materials Toughness

Prairie Island submitted its Reactor Vessel Material Surveillance Program on October 31, 1977.

A-12 Potential of Low Fracture Toughness and Lamellar Tearing in PWR SG and RCP Supports

Prairie Island responded to NRC letter dated September 14, 1977 with a report on November 11, 1977 entitled, "An Examination of Reactor Coolant Pump and Steam Generator Supports"; the report was updated on December 27, 1977. A review of the report indicates that the Prairie Island equipment supports are free of the design and material deficiencies described in the NRC letter.

A-17 Systems Interactions in Nuclear Power Plants

Prairie Island is preparing a PRA which sould resolve this issue. The schedule for PRA/IPE completion is attached to NSP letter dated October 30, 1989, "Response to Individual Plant Examination for Severe Accident Vulnerabilities Generic Letter No. 88-20, Supplement No. 1." The Prairie Island PRA should be completed in February 1993.

A-24 Qualification of Class 1E Equipment

The NRC transmitted a Safety Evaluation Report and a Technical Evaluation Report on April 25, 1983. NSP responded on May 5, 1983, May 19, 1983 and May 27, 1983. The NRC transmitted a Safety Evaluation Report on June 3, 1983. NSP provided additional information on August 10, 1983. The NRC issued a final Safety Evaluation Report on March 25, 1985.

NRC Inspection Report Nos. 50-282/86012 and 50-306/86014 dated December 29, 1986 documents the Environmental Qualification inspection to verify compliance with the Environmental Qualification rule.

A-26 Reactor Vessel Pressure Transient Protection

Prairie Island responded to NRC letter dated August 13, 1976 on September 3, 1976 and December 10, 1976. The Prairie Island response to NRC letter dated January 10, 1977 on February 25, 1977, stated that the modifications would be completed by the end of 1977, and that a backup air supply would be installed in Unit I in the Spring 1978 outage and in Unit II in the Fall 1977 outage. Prairie Island responded to NRC letter dated February 11, 1977 on April 4, 1977. Prairie Island submitted a report on July 22, 1977 entitled, "Pressure Mitigating Systems Transient Analysis Results."

Prairie Island responded to the NRC request for Tech Specs dated May 1, 1978 on August 4, 1978. The NRC issued a Safety Evaluation Report with License Amendment Nos. 38 and 32 on August 3, 1979.

A-31 RHR Shutdown Requirements

Not Applicable. License received prior to January 1979.

A-36 Control of Heavy Loads, Phase I and II

Prairie Island responded to the NRC request dated May 17, 1978 for information on control of heavy loads near spent fuel on July 21, 1978.

The NRC closed out "Control of Heavy Loads - Phase I" in a Safety Evaluation Report dated June 6, 1983. The Technical Evaluation Report was sent to NSP on May 31, 1984. Generic Letter 85-11, dated June 28, 1985 closed out "Control of Heavy Loads - Phase II."

A-39 Determination of SRV Pool Dynamic Loads and Temperature Limits

This is a BWR issue.

A-40 Seismic Design Criteria

Prairie Island is participating in the Seismic Qualification Utility Group (SQUG). See USI A-46.

A-42 Pipe Cracks in Boiling Water Reactors

This is a BWR issue.

A-43 Containment Emergency Sump Performance

A-43 is applicable to new construction only.

A-44 Station Blackout

Prairie Island provided information required by 10 CFR Part 50 Section 50.63(c)(1), loss of all alternating current power, on April 13, 1989. The modifications and procedure changes identified in the April 13, 1989 submittal will be completed prior to the star[illegible] of Cycle 15 for Unit 2 (1992).

A-45 Shutdown Decay Heat Removal Re[illegible]ements

Prairie Island is preparing a PRA which sould resolve this issue. The schedule for PRA/IPE completion is attached to NSP letter dated October 30, 1989, "Response to Individual Plant Examination for Severe Accident Vulnerabilities Generic Letter No. 88-20, Supplement No. 1." The Prairie Island PRA should be completed in February 1993.

A-46 Seismic Qualification of Equipment in Operating Plants

Prairie Island response to Generic Letter 87-02, on October 6, 1988 stated that Prairie Island plans to have seismic verification plant walkdowns within 60 days following conclusion of the third refueling outage after receipt of the final staff Safety Evaluation Report supplement and resolution of all open issues and completion of walkdown team training.

A-47 Safety Implication of Control Systems in LWR Nuclear Power Plants

Prairie Island's response to Generic Letter 89-19 is due March 19, 1990.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Since Prairie Island is a PWR with a large dry containment Unresolved Safety Issue A-48 does not apply. However, in response to NUREG-0737 Prairie Island hydrogen recombiners were installed; Unit 1 in the 198[illegible] refueling outage and Unit 2 in the 1983 refueling outage. High point vents were installed in 1981 refueling outage for Unit 1 and in the 1982 refueling outage for Unit 2.

Technical Specifications were requested on July 1, 1983. License Amendment Nos. 68 and 62 were issued on February 21, 1984.

A-49 Pressurized Thermal Shock

Prairie Island letter dated January 10, 1986 submitted an Assessment of Pressurized Thermal Shock Reference Temperature in Accordance with 10 CFR Part

5C, Section 50.61. The NRC issued Safety Evaluation Reports for Unit 1 on August 18, 1986 and for Unit 2 on June 23, 1986.