NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

October 17, 1979

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Follow-up Actions Resulting from the NRC Staff Reviews
Regarding the Three Mile Island Unit 2 Accident

A letter, dated September 13, 1979, from Darrell G. Eisenhut, USNRC, transmitted NRR's requirements which have evolved from the Lessons Learned and Emergency Preparedness task forces. All operating reactor licensees were requested to begin implementation of these requirements and to submit, within 30 days of receipt of the September 13, 1979, letter, a commitment to meet these requirements on an implementation schedule set out in enclosures to the NRC letter.

Enclosed as Attachment "A" are statements describing actions already taken on the NRR requirements at the Prairie Island Plant, plans for future actions, and expected implementation dates. The statements are keyed to the NUREG-0578 section numbers listed in Enclosure 6 to the September 13, 1979, letter on Lessons Learned and the item numbers for Emergency Preparedness in Enclosure 8.

The expected implementation dates are based upon timely receipt of NRC guidance in some areas, availability of equipment from suppliers, refueling outage schedules, and other factors not under full control of the licensee. The physical security modification program aptly demonstrated the effect on implementation schedules of limited supplies of specialized equipment for a large number of licensees. NSP will keep the NRC informed of potential delays to the schedules contained herein as they become apparent.

L. O. Mayer

L. O. Mayer, P.E.
Manager of Nuclear Support Services

LOM/amh

cc: J. G. Keppler
G. Charnoff

1206 343

7910240619 P

October 17, 1979

ATTACHMENT A

PART I

PRAIRIE ISLAND

ENCLOSURE 6
RESPONSES

1206 344

2.1.1 EMERGENCY POWER SUPPLY

Testing has shown that one group of pressurizer heaters (192 KW) is more than sufficient to maintain natural circulation. The Backup Group "A" Pressurizer Heaters are connected to the 480V Train A Safeguards Busses and are sequenced on by the voltage restoration scheme after loss of offsite power. The Backup Group "B" Pressurizer Heaters are normally energized by a non-safeguards bus but may be transferred to the "B" Train Safeguards bus by a manual transfer switch.

The PORV's are pneumatic valves operated by instrument air. They fail closed upon loss of air. They are controlled by D.C. solenoid valves which fail closed and vent upon loss of power. Each solenoid valve (two per pressurizer) is supplied by a separate train of D.C. Safeguards power.

The PORV blocking valves are motor valves each supplied by a separate train of 480V Safeguards power. They are cross-trained to the PORV they block so that loss of one train of safeguards power will not prevent closing of the pressurizer relief line.

Pressurizer level instrument channels are powered from the vital instruments busses which are uninterruptible power sources.

2.1.2 RELIEF AND SAFETY VALVE TESTING

We commit to provide performance verification by prototypical testing for relief and safety valves. The testing program will be controlled by the Utilities Owners' Group and they will attempt to meet the implementation schedule.

2.1.3.a DIRECT POSITION INDICATION OF RELIEF AND SAFETY VALVES

Prairie Island already has direct indication of the Pressurizer Power Operated Relief Valve (PORV's) and will install an acoustic monitor on the common Pressurizer Safety Valve Relief Header.

Presently the PORV's have environmental qualified Snap-Lock Limit Switches which show the valve's position. The PORV's motor operated isolation valves also have valve position indication. These limit switches are powered from instrument buses with safety-grade cabling and routing. These switches indicate the valves position in the control room on the Reactor Coolant System Panel.

The existing indication of Safety Valve flow is separate RTD's in the discharge header and Pressurizer Relief Tank Pressure, Temperature, and Level. To augment indication, a common acoustic monitor will be installed. Only one monitor is being used because the piping geometry would, to adequately determine the leaking valve, not give a separation of signals and the operator's actions would be the same no matter which safety valve had flow. This indication would trigger an alarm in the control room and also print out on the plant computer.

This system will be installed at Prairie Island in 1980 during plant maintenance and refueling outages. These outages are presently scheduled for January for Unit 2 and fall for Unit 1. We consider these

1206 345

installation dates adequate for the following reasons. The PORV's, which are the major concern of the NUREG 0578 discussion, are already instrumented with safety-grade valve position indication. The Safety Valves are a simple spring operated valve, which already has several indirect indications of Valve Leakage. In addition, during each outage the Safety Valves relief setting is checked, after which it is leak tested at operating pressure.

2.1.3.b INSTRUMENTATION FOR INADEQUATE CORE COOLING

Procedures to use RCS parameters as a means of detecting inadequate core cooling are presently being reviewed. Appropriate procedures and/or revisions to existing procedures will be completed as required by the implementation schedule.

We will install primary coolant saturation meters by 1-1-80 or as soon as they can be delivered from the supplier. Upgrading of the RCS pressure and temperature inputs to safety grade will be done in the Fall 1980 outage for Unit 1 and the Spring 1981 outage for Unit 2.

Evaluation of new instrumentation to be installed for monitoring core cooling is continuing. Reactor vessel level indication is included in this category. New instrumentation proposals will be submitted to the NRC for review.

2.1.4 DIVERSE CONTAINMENT ISOLATION

Containment Isolation is initiated by a Safety Injection signal which is generated by high containment pressure, low pressurizer pressure, and low steam line pressure. In addition, a high radiation level in the containment or ventilation system ducting is used as a diverse parameter for automatic Containment Ventilation Isolation.

Refer to Table 2.1.4 below.

TABLE 2.1.4

(abbreviations on following page)

Penetration	System	Essential	Non-Essential	Basis (Req'd if Essential)
PRT to GA	RC		X	
PRT N_2 Supply	RC		X	
Prim Vnt Hdr	WD		X	
RCDT	WD		X	
Main Steam	MS		X	
Main Feedwater	FW		X	
SGBD	SGBD		X	
RHR	RHR	X		Emergency Core Cooling
Letdown	CVCS		X	
Charging	CVCS	X		Maintain Primary System Makeup
RCP S1 Wtr Sply	CVCS	X		Maintain RCP Operation

1206 346

Penetration	System	Essential	Non-Essential	Basis (Req'd if Essential)
RCP Sl Wtr Rtrn	CVCS		X	
Przr Stm Smpl	SM		X	
Przr Liq Smpl	SM		X	
Lp B Htlg Smpl	SM		X	
Instrument Air	SA		X	
RCDT to GA	WD		X	
Cntmt Air Smpl	RD		X	
Cntmt Smp A Dschg	WD		X	
Cntm Purge	ZP		X	
Safety Inj	SI	X		Emergency Core Cooling
Cntm Spray	CS	X		Engineered Safeguard System
Cntm Smp B Suct	SI		X	Essential for Long Term Cooling
N_2 to Accumulator	SI		X	
CC for RCP	CC	X		Maintain RCP operation
SI Test Line	SI		X	
Clg Wtr For FCU	CL	X		Required for Cntmt Fan Coil Clg
CC For Ex Ltdn Hx	CC		X	
Cntmt Vacuum Bkr	ZC		X	Essential for Long Term Condition
POST LOCA H_2 Cntrl	HC		X	Essential for Long Term Condition
Cntmt In-Serv Prg	ZV		X	
RMU to PRT	RM		X	
AF	AF	X		Provide necessary heat sink

Abbreviations

PRT	Pressurizer Relief Tank	GA	Gas Analyzer
RC	Reactor Coolant	N_2	Nitrogen
WD	Waste Disposal	RCDT	Reactor Coolant Drain Tank
MS	Main Steam	FW	Feedwater
SGBD	Steam Generator Blowdown	RHR	Residual Heat Removal
CVCS	Chemical & Volume & Control System	RCP	Reactor Coolant Pump
Przr	Pressurizer	Cntmt	Containment
RD	Radiation Monitoring System	ZP	Containment Purge Ventilation System
SI	Safety Injection	CC	Component Cooling System
Clg Wtr	Cooling Water	FCU	Fan Coil Units
Ex Ltdn Hx	Excess Letdown Heat Exchanger	RMU	Reactor Makeup
RM	Reactor Makeup System	CL	Cooling Water System
ZC	Containment Fan Coil Ventilation System	AF	Auxiliary Feedwater System
ZV	Containment In-Service Ventilation System		

- Non-Essential Systems to be automatically isolated by Containment Isolation Signals.

RESPONSE: All Non-Essential Systems are automatically isolated by a Containment Isolation Signal.

1206 347

- Resetting of Containment Isolation Signals shall not result in the Automatic Loss of Containment Isolation.

RESPONSE: Resetting or Containment Isolation Signals does not result in the Loss of Containment Isolation.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

Two independent 2" lines for containment hydrogen control exist presently at Prairie Island for each unit. These lines are safety related QA I lines and the valves are supplied by independent QA I power supplies.

Because there are two separate trains for each unit, a single active failure (such as a valve stuck shut) will cause only one train to be inoperable for containment hydrogen control. Failure of a single power source (or bus) can only cause one train to be inoperable because of independent power sources.

Containment integrity is assured for supply lines by a check valve inside containment and a normally closed (control switch locked) motor valve in shield building. Vent lines to the shield building and gas analyzer are isolated from containment by a normally closed motor valve inside containment (CS locked) and fail closed air-operated valves in the shield building. (CS also locked) Requirements of 10CFR50 APPENDIX A criteria 54 & 56 are met.

Each line is tested before heat-up to ensure a flow of 25 SCFM. Motor valves and air-operated valves are stroked quarterly and are leaked tested at refueling intervals.

POWER SUPPLIES UNIT 1

IA Unit 1 Train "A"

Power supply is Bus 110 (fed from 4160/480 volt transformer 101 which is supplied from 4160 V Bus 15) for:

MV32274
MV32271
CV31923
CV31925
SV33990

IB Unit 1 Train "B":

Power supply is Bus 120 (fed from 4160/480 volt transformer 102 which is supplied from 4160 volt Bus 16) for:

MV32276
MV32273
CV31927
CV31929
SV33991

1206 348

POWER SUPPLIES UNIT 2

IIA Unit 2 Train "A":

Power supply is Bus 210 (fed from 4160/480 volt transformer 201 which is supplied from 4160 V Bus 26) for

MV32293
MV32290
CV31924
CV31926
SV33992

IIB Unit 2 Train "B":

Power supply is Bus 220 (fed from 4160/480 volt transformer 202 which is supplied from 4160 V Bus 25) for:

MV32295
MV32292
CV31928
CV31930
SV33993

1206 349

2.1.5.c - H_2 RECOMBINERS

The plant has a pressurization/purge system for long-term post accident combustible gas control which does not require Hydrogen recombiners.

No action on the part of the licensee is required at this time in accordance with the letter from D G Eisenhut, USNRC, dated September 13, 1979.

2.1.6.a - SYSTEM INTEGRITY FOR HIGH RADIOACTIVITY

We commit to the NRC position on leakage detection and management as outlined in NUREG-0578, 2.1.6.a.

Gaseous systems leak measurement will be performed using a volumetric inventory method.

Liquid system leak detection will use walk down of piping using boric acid residue as leak telltale. Leak measurement for liquid systems will be deduced from piping and component external examination.

A preventive maintenance program is in effect to inspect/replace components which have a history of leakage.

2.1.6.b - PLANT SHIELDING REVIEW

A design review will be made to assure adequate shielding from systems outside containment which may contain primary coolant or gases to maintain adequate access to equipment in the plant, using the Lessons Learned source terms. The design review should be completed by 1-1-80 and the modifications should be completed in early winter 1981 contingent on refueling outage schedules, extent of modifications, and material procurement.

2.1.7.a - AUTOMATIC INITIATION OF THE AUXILIARY FEEDWATER SYSTEMS FOR PWRs

The following signals AUTOMATICALLY start the pump motors and open the steam admission control valve to the turbine driven pump, of the affected unit:

1. Low-Low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety Injection
4. Undervoltage on both 4.15KV normal buses (turbine driven pump only)

In addition, both local control (from the Hot Shutdown Panel) and remote control (from the control Room) can be used to MANUALLY initiate AFW.

The initiating signals and circuits for starting both the motor and turbine driven pumps are capable of being tested.

The instrumentation and control power supplies are from the 120 VAC vital bus system. There are four vital buses per unit, each supplied by an inverter connected to the 480 VAC emergency bus and the 125 VDC power system. The motor driven pump breaker controls are powered from the 125 VDC control batteries which are charged by battery chargers connected to the 480 VAC emergency buses.

1206 350

The AFW motor driven pumps are included in the load restoration sequencing on the emergency buses. Motor operated valves (also Safety Injection Pumps) are not stripped from the emergency buses, i.e., remain energized.

A failure in the automatic circuitry will not affect the manual capability to initiate the AFWS from the control room.

The SI actuation circuits which initiate AFW are of safety grade, being separated and trained. The SI actuation contacts which trip off the normal feedwater pumps are also of safety grade.

2.1.7.b - AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS

Control room indication of auxiliary feedwater flow to each steam generator is provided for both Prairie Island Units 1 and 2.

Power supply for the analog devices providing this indication is from the Number 4 instrument inverter and instrument bus for the respective unit. For each unit this is one of four separate safety grade inverters and bus instrument AC power circuits. Each inverter is continuously supplied power from one of the safeguards 480 volt AC buses and from one of the 125 volt DC emergency batteries.

The primary function of this flow measurement is indication of adequate heat sink availability. The presently installed wide range steam generator level channels provide a diverse fulfillment of this function. There is one wide range level channel per steam generator continuously recorded in the control room. Each of these channels is powered from one of the three inverters not supplying power to auxiliary feed flow.

Additional local indication, in the auxiliary building, of auxiliary feed flow is provided by separate instruments at the location of the flow transmitters for control room indication. These local indicators are readily accessible to operators and are used for auxiliary feed system operability testing.

2.1.8.a - POST ACCIDENT PRIMARY COOLANT SYSTEM AND CONTAINMENT ATMOSPHERE SAMPLING SYSTEM

The requirements of Lessons Learned will be met. Spectra analysis capability will be available. The design review for obtaining samples will be completed by 1-1-80. Interim sampling procedures will be written by 1-1-80. Design modifications should be performed by early 1981 contingent on refueling schedules, extent of the modifications and material procurement.

2.1.8.b - INCREASED RANGE OF RADIATION MONITORS

Post accident ventilation of the auxiliary building is made via Auxiliary Building Special Ventilation System to the shield building vent duct. Each shield building stack will be equipped with two radiation monitors (one low range and one high range with at least one decade overlap), a sample pump for each radiation monitor, and a charcoal and particulate sample filter for each. The low range system will be installed by

1-1-80. If not installed at that time other methods for estimating the release based on stack gas radiation levels will be completed by 1-1-80.

Measurement of Iodines and Particulates in stack effluents will be based on the continuous samples and analyzed on a mobile GeLi system. This equipment will be available by 1-1-80.

The in-containment high radiation monitor (10^7 R/hr photon radiation only) will be procured and installed in the Fall 1980 and early Winter 1981 refueling outages. We are presently purchasing Victoreen instruments which claim to meet the design safety grade.

2.1.8.c - IMPROVED IN-PLANT RADIOIODINE MONITORING

A mobile GeLi system will be operable by 1-1-80 which will allow the samples to be analyzed accurately for iodines and particulates. Samples will be taken in-plant and moved to a low-background area and analyzed. Operating Procedures and training will be completed by 1-1-80.

1206 352

2.1.9 - TRANSIENT AND ACCIDENT ANALYSIS

As identified at the Chicago regional meeting, specific requirements are being developed in a continuing series of meetings between the Owners' Group and the Bulletin and Orders Task Force. As these requirements are clarified, we will respond to them. In response to attachment 5 of the September 13 letter, we offer the following:

TASK 1:

The small break LOCA analysis and emergency procedure guidelines have been developed and submitted to the NRC for their review. Interchange is continuing on these items at this time.

TASK 2:

Key plant personnel are attending a Westinghouse Owners' Group sponsored seminar the week of October 15. At the seminar, the most recent revisions of the Westinghouse emergency procedure guidelines will be presented along with sessions on accident analysis - procedure interface. Pending final resolution of NRC questions on the Westinghouse guidelines, the plant specific procedures can be revised and operator training can begin. As indicated in our response to IE Bulletin 79-06C, final implementation of the revised plant specific procedures can be completed three to four months after issuance of NRC favorably reviewed procedure guidelines.

TASKS 3 through 7:

These tasks are the subject of continuing discussion between the NRC and the Westinghouse Owners' Group.

2.1.9 - CONTAINMENT PRESSURE INDICATION, CONTAINMENT WATER LEVEL INDICATION AND CONTAINMENT HYDROGEN INDICATION

We will investigate and proceed with design and installation of extended range containment pressure, containment level and hydrogen measurement instrumentation pursuant to the ACRS requirements, the requirements of Reg Guide 1.89 and the forthcoming revision of Reg Guide 1.97. Installation of the sensors will be during the 1980 and 1981 outages for Prairie Island Units 1 and 2.

2.1.9 - RCS VENTING

We agree to comply with the requirement for a head venting system and will submit our design by 1-1-80. Installation is expected to be completed during our 1981 refueling outages.

2.2.1.a - SHIFT SUPERVISOR RESPONSIBILITY

We commit to the requirement and implementation schedule. Provisions exist and are defined in our Administrative Controls which define the Shift Supervisor's responsibility. We are investigating his administrative and command and control functions in light of Lessons Learned. We are investigating the corporate level Directives and Policies to ensure proper management direction.

1206 353

2.2.1.b - SHIFT TECHNICAL SUPPORT

As identified at the Chicago regional meeting, this item is scheduled for generic resolution between the licensees and the NRC. By January 1, 1980, Prairie Island will have a Shift Technical Advisor on duty that meets the generic requirements arrived at as a result of regional and topical meetings between the licensees and the NRC. If complete compliance with the generic resolution by January 1, 1980 is not possible by practical application of available resources, we will inform the NRC as soon as possible, but before January 1, 1980.

The training of the Shift Technical Advisor is planned to be completed by January 1, 1981.

2.2.1.c - SHIFT & RELIEF TURNOVER PROCEDURE

A checklist is presently in use for shift turnover, which defines shift change status information between Shift Supervisor, Lead Plant Equipment and Reactor Operator and Plant Equipment and Reactor Operator. Information includes significant equipment/system not available for service, significant equipment repaired and made available for use during the shift, outstanding surveillance procedures, significant work requests, operational plans for the coming shift, and new operational and administrative procedures. Important Technical Specification limits and plant parameters are indicated on control room annunciators.

Other plant sections such as maintenance and instrument technicians normally do not work shifts. All work is controlled by the control room so that critical system line up and status is the control room responsibility.

2.2.2.a - CONTROL ROOM ACCESS

We commit to the requirement and implementation schedule. Presently, provisions exist for limiting access to the Control Room which are defined in our administrative controls. We will review and modify as necessary our administrative controls to define succession of authority during emergencies.

2.2.2.b - TECHNICAL SUPPORT CENTER

By January 1, 1980 a Technical Support Center will be established in the "engineering conference area" of the plant administrative offices. This engineering conference area is located across the turbine hall from the Units 1 and 2 main control room and is in "close proximity" to the control room. The T.S.C. will have the following capability by January 1, 1980:

1. Plant process computer output in the form of "slave" CRT and dedicated output typer that can access any computer input.

   NOTE: This is a present feature of the plant engineering area.

2. As built plant records that include general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists and single line electrical diagrams.

1206 354

3. Provisions will be provided to detect whole body and air borne radiation.

4. An extension for the NRC "red phone".

5. Upgraded communications with the Main Control Room.

6. Procedures will be provided to address the role of the T.S.C. during the incident.

Also by January 1, 1980, a detailed submittal will be provided that will describe the final configuration of the T.S.C. that we plan to have implemented by January 1, 1981. This submittal will describe a TSC that will provide the functions required by NUREG 0578.

2.2.2.c - ONSITE OPERATIONAL SUPPORT CENTER

By January 1, 1980, an Onsite Operational Support Center will be established. The center will be located in the "Plant Operating Records Room" which is immediately adjacent to the main control room. The room presently functions as an operator's "duty room" and will function as a reporting point for operations support personnel during an emergency.

Procedures will be revised to reflect the existence of this center by January 1, 1980. Communications between the Operational Support Center, the Technical Support Center and the Control Room will also be established by 1/1/80.

1206 355

October 17, 1979

ATTACHMENT A

PART II

PRAIRIE ISLAND

ENCLOSURE 8
RESPONSE

1206 356

NEAR TERM EMERGENCY PREPAREDNESS IMPROVEMENTS

NSP commits to the Near Term Emergency Preparedness Improvements and the implementation schedule for those improvements.

Item 1 The emergency plans will be upgraded to Regulatory Guide 1.101.

Item 2 Lessons Learned items identified in NUREG-0578 will be implemented as previously discussed. The onsite support center and technical support center have been described.

Item 3 Emergency centers for off-site support agencies have been developed. The intent is for these centers to meet the NRC concerns and timetable.

Item 4 Off-site monitoring capability (increased number of TLD's) will be improved per the recommended schedule of mid 1980.

Item 5 The adequacy of State/local plans will be assured. New plans are presently with the NRC for concurrence.

State/local plans will be updated as plant plans are improved and equipment improvements are made.

Item 6 Tests of the plant plans after improvements will be made by mid 1980. Tests of the state plans are scheduled for this fall in order to obtain NRC concurrence with those plans.