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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

December 30, 1980

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Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

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

Post TMI Requirements - NUREG-0737

All licensees of operating plants were mailed a letter dated October 31, 1980 from Mr D G Eisenhut, Director, Division of Licensing, Office of Nuclear Reactor Regulation, which contained revised Post TMI Requirements in a document identified as NUREG-0737. The licensees were requested to furnish confirmation that the implementation dates indicated in Enclosure 1 to NUREG-0737 will be met or to furnish justification for delays.

In the attachment to this letter, Northern States Power Company provides commitments to those hardware, procedural and organizational implementation requirements that will be met or furnishes justification and requests for exemption for those implementation requirements that may be delayed. Our commitments are dependent on equipment availability and assume no changes in regulatory positions or interpretations beyond those stated in the October 31, 1980 letter.

Mr Eisenhut's October 31, 1980 letter requires a large number of design descriptions, evaluations, and information transmittals to be submitted by January 1 or 2, 1981. Much of this material has already been submitted for NRC Staff review and other items will be addressed in Owners Group correspondence. For the remaining submittals we find we will be unable to meet the required January 1 or 2, 1981 date because of the lateness of the NRC's clarification information, the holiday vacation schedule, and the already heavy burden placed on our technical staff by other NRC requirements such as fire protection, environmental qualification, and on-going TMI modification work. We are directing our efforts to providing the written submittals for these items by February 1, 1981. Information submittals for the following NUREG-0737 items are involved:

8101050 256

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I.A.1.1(3&4)
II.K.3(17)
III.A.2(2)
III.D.3.4(1&2)

We will notify our NRC Project Manager if delays in the implementation or submittal dates are expected.

L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services
LOM/DMM/bd

cc: J G Keppler
NRC Resident Inspector
G Charnoff

Attachment

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Docket No. 50-282
50-306

LETTER DATED DECEMBER 30, 1980
RESPONDING TO NRC LETTER DATED OCTOBER 31, 1980
POST TMI REQUIREMENTS - NUREG 0737

Northern States Power Company, a Minnesota corporation, by this letter dated December 30, 1980 hereby submits information in response to the NRC letter dated October 31, 1980 which contains revised post TMI requirements in a document identified as NUREG-0737.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By /s/ L O Mayer
L O Mayer
Manager of Nuclear Support Services

On this 30th day of December, 1980, before me a notary public in and for said County, personally appeared L O Mayer, Manager of Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

/s/ Jeanne M Hacker
Jeanne M Hacker
Notary Public-Minnesota
Hennepin County
My Commission Expries May 6, 1986

Attachment
Director of NRR
December 30, 1980

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

COMMITMENTS TO IMPLEMENTATION DATES SPECIFIED IN NUREG-0737 "CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS"

Northern States Power Company has reviewed the clarifying information contained in NUREG-0737 and has committed, where possible, to the implementation dates specified in that document. For many of the Action Plan items it was necessary to summarize our understanding of the Commission's requirements and discuss, in some detail, what actions have been taken or are in progress to resolve the issue. The following Action Plan items covered by NUREG-0737 are applicable to the Prairie Island Nuclear Generating Plant. Our commitment to each of the applicable requirements is contained on the page indicated:

I. A.1.1 Shift Technical Advisor (STA)

NSP will comply with the implementation requirements for this item with the following exceptions: the description of the current STA training program and demonstration of conformance with the October 30, 1979 letter, and the description of the long-term STA program will be submitted by February 1, 1981. Preliminary information on this subject is offered below.

The Shift Technical Advisor will have a bachelor's degree or equivalent in a scientific or engineering discipline and will have received training in (a) response and analysis of the plant for transients and accidents, and (b) plant design and layout including the control room, instrumentation and controls. Any individual with a degree in engineering or science and who possesses or has possessed a senior reactor operator license and is participating in the STA or SRO requalification program is considered to meet both the short and long term training requirements.

For those who possess a degree in engineering or science or equivalent and have not had an SRO license, the Shift Technical Advisor training program involves training in the following areas:

- Reactor Theory
- Heat Transfer, Fluid Flow, and Thermodynamics
- Radiation Safety
- Chemistry
- Materials
- Instruments and Controls
- Plant and Control Room Systems
- Transient and Accident Analysis

This training consists of formal lectures (including videotape); self-study; quizzes, exams, or checkoffs; and on-the-job training.

The STA retraining program consists of lectures involving the aforementioned topical areas following similar guidelines as the licensed operator requalification program. This program also includes simulator training sessions.

In the long term, the STA program will include the following subjects:

- Reactor Theory (including calculus as appropriate)
- Reactor Chemistry
- Nuclear Materials
- Thermal Sciences
- Electrical Sciences
- Radiation Protection and Health Physics
- Plant Systems and Procedures
- Transient Analysis
- Simulator Training

The long term requalification program will be the same as the short term program.

Proposed Technical Specifications were contained in our License Amendment Request dated December 19, 1980.

I.A.1.2 Shift Supervisor Responsibilities

This requirement has been satisfied.

The following Administrative Control Directives and Section Work Instructions define the Shift Supervisor's responsibilities, duties, requirements, training and authority:

1ACD3.2	Corporate Level Plant Operations
3ACD3.2	Power Production Level Plant Operations
3ACD3.5	Power Production Level Training
5ACD3.1	Plant Level Plant Organization
5ACD3.11	Plant Level Plant Training Program
SWI-0-2	Shift Organization, Operation & Turnover

These directives and instructions were modified to reflect the requirements of NUREG-0578 and Harold Denton's letter of clarification. SWI-0-2 and 5ACD3.1 require the Shift Supervisor to remain in the control room at all times during accident situations unless properly relieved. Persons authorized to relieve the Shift Supervisor are specified.

I.A.1.3 Shift Manning

Administrative procedures presently exist which define requirements for shift manning. They follow the guidelines for overtime as defined in Darrell Eisenhut's letter of July 31, 1980. NUREG-0737 requires that our administrative controls be modified to comply with several new requirements. In general, we agree with the requirements and will modify our procedures to comply with NUREG-0737. However, we take exception to some of the statements in the clarification.

We feel that an overtime policy is important to assure operator aler[illegible]s and agree generally that there should be a break of at leas[illegible] hours between all work periods. However, this strict requi[illegible] presents problems in that it restricts the number of people [illegible]an take days off at a given time. It has the effect of taking a[illegible]a double time day for some people. It causes us to go against our labor agreement for call out. In general, we feel this requirement can lead to a decrease in operator alertness due to disrupting sleeping habits and the established time off policy. We feel 8 hours is adequate and prevents the problems described in those few cases where we could not comply with the 12 hour break period. We do not intend to utilize the 8 hour break period in cases where 12 hour work periods are required.

We agree that an individual should not be required to work several consecutive days without time off. The 14 day requirement can be complied with in all situations except when it would cause individuals to make additional adjustments to short term changes. This occurs infrequently, but when it does occur the 14 day requirement would be violated by one day. We feel a 15 consecutive day work policy with this requirement would not decrease operator alertness.

We take exception to the requirement that an operator who works more than 8 continuous hours be periodically relieved of primary duties at the control board so that periods of duty at the control board do not exceed 4 hours at a time. This requirement could lead to an overall decrease in alertness because it would require additional scheduling of operators. The requirement could decrease safety due to the frequent formal turnover requred.

With the three exceptions described above, we agree with the requirements and commit to the schedule. Administrative procedures governing shift manning will be revised as required.

I.A.2.1 IMMEDIATE UPGRADING OF REACTOR OPERATOR & SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

On August 1, 1980, the revised requalification programs and topical outlines for training in heat transfer, fluid flow, thermodynamics, mitigation and control of core damage and plant transients were submitted. No response from the NRC on this submittal has yet been received. Further action on this item is awaiting NRC review and approval.

SRO Experience

NSP practice has been to encourage plant staff engineers to become licensed SRO's. The SRO training programs, as previously implemented at Prairie Island and being implemented at Monticello, is an eight phase program. This program, described in a letter from F P Tierney (by J A Gonyeau) to P Collins,USNRC, dated April 22, 1976, covers the following topics:

Phase A - Theory (Reactor Physics, Chemistry, Materials)
Phase B - Systems
Phase C - Control Room Checkoffs & Training
Phase D - Simulator Training & Certification
Phase E - RO Exam Review
Phase F - SRO Exam Review
Phase G - Pre-NRC Written Examinations
Phase H - Pre-NRC Oral Examinations

Phase C will include the requirement that the trainee spend three months on shift as an extra person for applications received by the NRC after December 1, 1980. Personnel with degrees in engineering or related science will be required to have had two years of responsible nuclear power plant experience as a technical specialist (e.g. engineer, chemist, etc.) and satisfactory completion of the aforementioned training program.

At NSP nuclear power plants, the control room operators are either licensed or trainees in accordance with the Technical Specifications. The operations training program normally consists of two years apprenticeship formal and on-the-job training followed by approximately one to two years of formal and on-the-job training as a journeyman including licensed operator training.

In has not been NSP's practice to have non-degreed personnel become "instant SRO's" except on a few rare occasions where the individuals have demonstrated superlative performance while involved in license training and may have had prior experience.

By the time a non-degreed individual has been promoted through "plant attendant" and "auxiliary operator" positions to "control room operator", the individual will normally have accumulated several years of responsible power plant operating experience. To arbitrarily require four years of experience as "control room operator" beyond that point is unwarranted.

For the reasons discussed above, we must take exception to the requirement in NUREG-0737 for non-degreed personnel to have four years of control room experience prior to becomming a SRO.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

Permanent members of the plant training staff or members of other organizations who routinely conduct training at the facility in integrated systems responses and transients associated with the licensed operator (hot license and requalification) NRC approved programs will be or have been senior reactor operator qualified or certified and be enrolled in appropriate requalification programs. When a simulator is available at the facility, this requirement will also apply to the simulator course instructors. Guest lecturers who are experts in certain areas (e.g. rod control, transient analysis, specific systems training) will not be required to meet this requirement.

I.A.3.1 REVISE SCOPE & CRITERIA FOR LICENSING EXAMINATIONS - SIMULATOR EXAMS

Prairie Island did not have a simulator onsite as of October 1, 1980, thus it would be necessary to schedule time for simulator examinations at other facilities since a plant specific simulator is not expected to be fully operational for three to four years.

Northern States Power Company believes it is appropriate that the simulator examinations, when conducted, be completed on a simulator at which the trainees have completed certification. The NRC has not as yet defined the scope or nature of the simulator examinations. Because there are differences between the simulators used for certification and the trainees's specific plant, the NRC simulator examinations and grading should take into account that trainees should not be penalized for subtle differences in simulator compared to plant behavior.

At present, the simulators used by Northern States Power are being scheduled one to two years in advance and simulator schedules may not agree with when licenses or examinations are needed. The current requirement for simulator examinations within two weeks of the NRC examinations at the plant is susceptible to schedule slippage on the part of the NRC, NSP or the simulator vendor.

For these reasons, Northern States Power requests exemption from this requirement and proposes the following alternative actions:

1. The NRC should define the scope and nature of the simulator examinations.

2. Within one year of the definition of requirements, the licensee would commit to simulator examinations. This time period would provide the licensee the opportunity to schedule more simulator training time if required.

3. The NRC should expand the time frame between when the simulator and plant examinations take place.

I.C.1 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

The Westinghouse Owners Group will submit by January 1, 1981, a detailed description of the program to comply with the requirements of Item I.C.1. The program will identify Owners Group previous submittals to the NRC, which we believe will comprise the bulk of the response. Additional effort required to obtain full compliance with this item (with proposed schedules for completion) will also be identified as discussed with the NRC November 12, 1980.

I.C.2 SHIFT TURNOVER

This item is considered complete.

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

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

The turnover procedure is reviewed by the Shift Technical Advisor each shift. The evaluation of the effectiveness of the turnover procedure is a line function and like other procedures of this nature is assessed by the Operations organization.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITY

This item is considered complete.

The following Administrative Control Directives and Section Work Instruction define the Shift Supervisor's responsibilities, duties, requirements, training and authority:

1ACD3.2	Corporate Level Plant Operations
3ACD3.2	Power Production Level Plant Operations
3ACD3.5	Power Production Level Training
5ACD3.1	Plant Level Plant Organization
5ACD3.11	Plant Level Plant Training Program
SWI-0-2	Shift Organization, Operation & Turnover

These directives and instructions were modified to reflect the requirements of NUREG-0578 and Harold Denton's letter of clarification. SWI-0-2 and 5ACD3.1 require the Shift Supervisor to remain in the control room at all times during accident situations unless properly relieved. Persons authorized to relieve the Shift Supervisor are specified.

I.C.4 CONTROL ROOM ACCESS

This item is considered complete.

Provisions exist for limiting access to the Control Room which are defined in our administrative controls. The requirements for succession of authority during an emergency are also defined in our administrative controls.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE

Northern States Power Company has systems in effect to ensure that feedback of operating experience is occuring. The plant operating staff and training staff are reviewing plant events to ensure that information pertinent to plant safety is continually supplied to operating personnel and other personnel and is incorporated into training and re-training programs. Through the use of the NSAC-INPO Notepad and SEE-IN system, events occuring outside the utility organization are being reviewed and presented to member utilities. We are receiving the output from the Notepad SEE-IN system at this time and disseminating the material internally as necessary.

Responsibilities for feedback of operating expericence to the plant staff have been identified and agreed upon. Due to internal re-organization, approved procedures to address all areas of item I.C.5 will not be available by 1/1/81. It is anticipated that completed procedures will be available by 4/1/81. Because the main functions of Item I.C.5 will be in place on 1/1/81 we see no safety significance in delayed issuance of formal procedures.

I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

We are reviewing existing Directives, Instructions and Procedures, to assure conformance with the supplemental positions in the clarification. We believe that the existing Operational Quality Assurance Program (based largely upon ANSI Standard N18.7-1976), satisfies many of these requirements. Detailed review of our administrative controls is continuing. A completion date of March 1, 1981 for completing this review has been established. Procedural changes required to achieve compliance with this requirement will be completed by June 1, 1981.

I.D.1 CONTROL ROOM DESIGN REVIEWS

The submittal date of April, 1982 can only be met if NUREG-0700 is issued prior to April, 1981. No schedule commitments can be made until this report is issued.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Clarification of the requirements for the safety parameter display system have not yet been issued by the Commission. Submittal and implementation dates have not been firmly established and no commitments can be made at this time.

II.B.1 REACTOR COOLANT SYSTEM VENTS

We will comply with the implementation schedule. A complete description of the design and operation of the reactor coolant gas vent system will be submitted to the NRC prior to July 1, 1981. This description will include the information required by NUREG-0737 section II.B.1.

System installation has begun and is expected to be completed for both units prior to the July 1, 1982 deadline. Procedures for use of the system will not be implemented prior to NRC authorization and the valves will have power secured to preclude inadvertant actuation.

Procedures for pre-implementation review will be issued by the Westinghouse Owners Group.

II.B.2 PLANT SHIELDING

The purpose of this item is to determine what actions can be taken to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident.

Essential areas of the plant have been identified. Areas requiring continuous occupancy have been determined to be the Control Room, Technical Support Center, and the Operations Support Center. The Security Center has not been included in this review. Access to the Security Center is not necessary in order to give access to other areas of the plant since normally locked vital areas of the plant can all be entered using a by-pass key. Design Basis Accident dose rates have been calculated for these areas. The dose rates at time=o are greater than 15 mrem/hour. However, the 30 day average is well below the 15 mrem/hour dose rate guideline for areas requiring continuous occupancy. The Records Room has been designated as the Operations Support Center (OSC). Should the actual dose rates in the Records Room be excessive (as may be the case for a few hours in a Unit 1 affected accident) the OSC will be moved to the I & C Shop.

Essential areas requiring only infrequent access have also been identified. These include the sampling stations (Reactor Coolant and Containment Gas), sample analysis areas, post LOCA hydrogen control stations, Radwaste Panel, manual Emergency Core Cooling System (ECCS) alignment area, Hot Shutdown Panels, most Safeguards Motor Control Centers (MCC) and Switchgear Rooms, Battery Room, and Emergency Diesel Generator Rooms. Floor plans have been developed to show the dose rates in these areas and in the general areas required to gain access to them. Stay times were estimated based on aligning the Residual Heat Removal system to recirculation mode and any other operations, such as resetting breakers, that may be required. Shielding modifications were then determined based on not exceeding 5 Rem whole body, or its equivalent, to any part of the body.

The design review determined the dose rates in the essential areas from systems that may, as a result of an accident, contain highly radioactive fluids. The systems evaluated were the High Head Safety Injection System (SI), Residual Heat Removal System (RHR), Containment Spray (CS), Shield Building Ventilation System (SBVS), Auxiliary Building Special Ventilation System (ABSVS), Containment Sampling System, (SM) and the Containment structure.

The Chemical and Volume Control System (CVCS) was not a part of this review. The letdown system should not be used in a high activity situation. The letdown portion of the CVCS will be isolated at a predetermined radioactivity level in the event of large fuel failure. Letdown will not be required to degas the Primary System. The head vent, venting to the Pressurizer Relief Tank (PRT) and ultimately the containment, will be used to degas the Primary System, should that become necessary. Letdown is not required to borate the primary system or to maintain seal injection flow. The charging pumps can take suction from either the Boric Acid Storage tanks (BAST) or the Refueling Water Storage tank (RWST). The head vent can be used to relieve water inventory from the primary system, and allow continuation of charging or seal injection. A Reactor Coolant System (RCS) cooldown and depressurization can be accomplished using the steam generators and the head vent. The head vent can function like the letdown system, only the vent would relieve eventually to the containment instead of the Volume Control Tank or Hold Up Tank. The BAST and RWST would supply ample borated water for a cooldown and eventual RHR System recirculation. Should the BAST and RWST be empty, because of SI and/or RHR injection into the primary system, it would indicate a large RCS leak. Long term recirculation would be initiated, and letdown would not be required. If, for some unforeseen reason, the RWST and BAST have been injected into the RCS and the water is now on the containment floor, the head vent can be used to lower the RCS pressure sufficiently to allow SI pump injection. The SI pumps would get suction from the RHR pumps, which are taking suction from the containment sump. Reactor Coolant Pump (RCP) seal injection can be maintained for at least eight hours, using the boric acid that is left in the BAST after the tank has been isolated from the SI pump suction. There is at least 850 gallons of 12% boric acid that can be blended to 1% and injected into the RCP seals. This would allow ample time to batch more boric acid and cool and depressurize the RCS, eventually going on RHR. Isolating the letdown system also eliminates the need for analyzing the dose rates from the Waste Gas System.

The dose rates have been calculated assuming letdown was isolated after a gap activity release accident. Dose rates from the CVCS piping were determined from the gap activity levels.

The activity levels in the RHR, SI, and CS systems have been conservatively estimated. It was assumed that 100% of the core equilibrium radioactive noble gas inventory and 50% of the core equilibrium radioactive halogen inventory have been diluted into the combined volume of the RCS, and the RWST. This assumes that the water in the RWST has been injected and the the RHR recirculation mode is in use. These dose rates are conservative because they are based on a time=0 activity and no degassing of the recirculation water by the blowdown into containment. NUREG-0737 permits the assumption that the water contains no noble gases when determining the source term for recirculated, depressurized, cooling water leaving 50% of the core equilibrium halogen inventory and 1% of all others. The 1% of all others was not included in our dose rate calculations. It has been assumed to be small in comparison to the non-degassed activity levels we used.

The evaluated systems were physically traced out to determine high dose areas. Direct radiation from the systems was calculated based on existing shielding as well as the shielding to be added as a result of this design review. Indirect radiation was also evaluated where applicable.

Dose rates from the Shield Building Ventilation System filters was determined. The design criteria leakage of 0.25 w/o per day from the containment vessel to the shield building was used as the basis of the SBVS filter isotope build-up. The effect of Containment Spray has been included in the analysis.

Auxiliary Building Special Ventilation System dose rates were also calculated. These dose rates are based on RHR pump seal failure in combination with the design criteria of 0.1 w/o per day leakage which by passes the SBVS and is deposited on the ABSVS filters. This is conservative because if the 0.1 w/o is deposited on the ABSVS then only 0.15% could be deposited on the SBVS filters.

Dose rates from the containment structure itself have also been included in the dose rates to the essential and general areas of the plant. 100% of the core noble gas inventory and 50% of the core equilibrium halogen inventory have been assumed to be uniformally mixed within the containment atmosphere. Dose rates in the Auxiliary Building have been calculated. These calculations take credit for the floors and walls within the Auxiliary Building. Large shield wall penetrations have also been analyzed. The penetrations that have been analyzed include ventilation penetrations on the 755' level, and main steam penetrations on the 715' level. Certain safeguard motor control centers that are located close to these shield wall portals have relatively high dose rates at time=0. However, these MCC's would probably be accessible within one hour of the event. The particular MCC's are not required for RHR recirculation mode line-up.

Several modifications have resulted from the design review which will reduce radiation levels in the Auxiliary Building. (1) The installation of the Reactor Head and Pressurizer Vent System eliminates the need for the Letdown System. This eliminates the Waste Gas System and the CVCS as a source of radiation. (2) The RHR Sump will be rerouted from the Waste Hold-up Tank to the Containment via the Annulus Sump. (3) Sampling System drains will be rerouted to the RHR Sump. Modifications 1,2, and 3 will ensure that all highly radioactive liquid and gaseous leakage will be pumped into or remain in the containment. (4) The SI pump area will be shielded with approximately 16 inches of concrete. (5) Shield "curtains" will be installed in the area of the SI pumps to reduce direct and indirect radiation. (6) The opening from the Containment Spray pump room will be shielded. (7) RHR pit covers will be redesigned so that they may be left in place except when maintenance or inspection is going on in the pits. (8) The safety injection line for reactor vessel injection will be rerouted. (9) The safety injection line for cold leg injection will be rerouted. (10) The 715' level above the RHR lines will be further shielded. (11) Miscellaneous shielding modifications to the sample room. (12) RHR supply to the SI pumps will need additional shielding. (13)The opening where the RHR lines enter the CS pump room will be shielded. (14) Stairway enclosures on the 715' level will have additional shielding added. (15) Safeguard MCC's on the 755' Level will be shielded from the SBVS and ABSVS filters.

Modifications (1),(8), and (9) were completed for Unit 1 during the refueling outage in September and October of 1980. Modifications (2),(3), and (4) are in the construction phase at this time. All shielding modifications will be completed prior to January 1, 1982.

The environmental qualification of safety related electrical equipment has been addressed in Bulletin 79-01B pursuant to the Commission Memorandum and Order dated May 27, 1980.

A review of non-electrical safety related equipment has also been performed to determine the radiation resistance of this equipment. This review included the following systems:

> Residual Heat Removal (RHR), High Pressure Safety Injection (SI), Containment Internal Spray (CS), Coolant Sampling (SM), Shield Building Ventilation (SBVS), Auxiliary Building Special Ventilation (ABSVS), and Post LOCA Hydrogen Control System (HC).

In addition, the containment isolation valves have been evaluated to ensure containment integrity with the radiation levels encountered during a Design Basic Accident (DBA). The Letdown and Waste Gas Systems were not considered for radiation qualification since letdown will be isolated as discussed earlier.

The major non-electrical components in the evaluated systems have been reviewed. This includes the pumps, heat exchangers, fans, and filters. Those valves that would be in contact with highly radioactive fluid in the event of a DBA have also been included in the review. In addition, containment isolation components, such as valves and airlock gaskets, have been evaluated to ensure that the radiation effects on these components do not cause failure resulting in loss of containment integrity.

The source terms used in the non-electrical equipment Total Integrated Dose (TID) evaluations were taken from the Owner's Group recommendations. For all liquid systems sump diluted source terms were used. This implies dilution of the core sources in the combined volumes of the primary coolant and the RWST. Noble gases were assumed completely stripped from the sump source. The system calculations were made with the dose point located at the pipe center to maximize he dose rate. These calculated doses were doubled to accommodate contributions from additional piping runs in close proximity. Any finer details on doses would require an evaluation of the physical configuration of the system at points of interest. Where appropriate, the normal plant operational doses have been included, and the expected dose to the system is the sum of the one year DBA TID and the 40 year operational dose.

Shield Building Ventilation and Auxiliary Building Special Ventilation systems one year TID have also been calculated. These doses were calculated based on design criteria leakage from containment and RHR pump seal failure.

Total Integrated Doses for the containment center, outer radius, and annulus have been calculated based on the NUREG-0737 sources (100% Core noble gases, 50% core halogens, and 1% others) for containment. Containment isolation valves and Post LOCA Hydrogen Control valve doses were assumed from these values. In some cases, more detailed calculations have been made for certain valves located inside containment taking credit for existing floors and walls.

In general most non-electrical equipment was found to be qualified for the calculated DBA doses. Some actions were required as a result of this design review. (1) CS pump secondary seals in the Durametallic Mechanical Seal will be replaced with qualified material. (2) Post LOCA Hydrogen Control System valve seats will be replaced with qualified material. (3) Containment radiation monitor sample valve seats will be replaced with qualified material. (4) The RHR and RCS Loop B sample valves will be replaced. (5) Containment airlock and equipment hatch seals will be replaced with qualified material. (6) Various valves in the RHR, SI, and CS systems require repacking with a qualified packing material.

Qualified valve packing material was required for those valves where the packing serves as a primary boundary between the radioactive fluid and the Auxiliary Building. Qualified packing was not required for those valves located inside containment.

The above modifications required for equipment qualification will be completed by June 30, 1982, or by the completion date established by IE Bulletin 79-01B future supplements or correspondence (in the event a later date is approved in certain cases and circumstances).

II.B.3 POST ACCIDENT SAMPLING CAPABILITY

Interim

Interim sample procedures were in place 1/1/80 to enable plant personnel to obtain reactor coolant and containment atmosphere samples and analyze them within three hours. These interim procedures provide for using the existing sample room to obtain the samples. No auxilary isolated system is necessary to use the sample points post accident. The ventilation from the sample room is exhausted through fume hoods to PAC filter units. Four inches of lead shielding was installed on sample coolers and drain traps that would be used during sampling. One inch of lead was placed on some sample lines that would be used for post accident sampling.

The containment air sample is obtained using the post-loca sample point on the installed Westinghouse gas analyzer. This unit utilizes a metal bellows pumps to obtain the sample and pass it through an MSA thermal conductivity hydrogen analyzer. This analyser will detect hydrogen from zero to one hundred percent in three ranges. The exhaust of this analyzer can be directed through a bomb with a septum where a grab sample may be obtained for isotopic analysis. A trailer based mobile count room located at Prairie Island was equipped with a computer based Ge(HP) analysis system, a proportional scalar, lights and heat, and a 5kw electrical generator. This self contained isotopic analysis unit may be moved anywhere to take advantage of wind direction and existing shielding. The Ge(HP) system was calibrated with geometries up to thirty six inches from the crystal for counting post-accident liquid, gas, and silver zeolite cartridge samples.

The reactor coolant sample is taken in the existing fume hoods in the sample room. Samples can be taken from either the RCS loop B sample line or the RHR system sample line. Mobile lead shielding is available for either units' fume hood. A small aliquot of coolant will be drawn and placed directly into a shielded carrier for transport to a lead brick hot cell located in the Turbine Building. This hot cell has reach rod capability, lead glass windows, and a fume hood that exhausts to a PAC unit. This hot cell has provisions for boron analysis by carminic acid method (ASTM D3082 1975 Ed.) on a diluted sample, a chloride by turbidimetric method on a diluted sample and multiple dilution capability to prepare liquid samples for Ge(HP) analysis in the trailer based mobile count room. Hydrogen analysis would be done by taking a small pressurized sample in a bomb and performing the analysis in the hot laboratory by gas partitioner.

Pending

Work continues on the reactor coolant sampling equipment. Central Research Inc. has built a shielded sampling manipulator package with two specially modified Model G manipulators. This package has the capability of serving either units' fume hood to obtain an aliquot of primary coolant in a vial which is placed in a lead carrying shield. The manipulator package is scheduled for installation by the end of the year. Two reactor coolant gas sampling packages were custom built by Beckman Instruments. Both panels are installed in the existing sample room, one for each unit. These panels provide for hydrogen analysis by thermal conductivity and the capability to obtain a gas grab sample for isotopic analysis. The gas grab sample has the ability to be diluted with a fixed volume of nitrogen to alleviate handling problems due to radiation dose rate. Both panels have four inches of lead shielding. All operations on the panels are manual. All panel vents are directed to the fume hood exhaust which passes through PAC filters. All panel drains are directed, in the post-accident condition, to the particular units' RHR sump and pumped to that units' containment. Final installation and pre-operational testing of this equipment will occur in 1981.

The containment gas sample bomb on the exhaust of the installed Westinghouse gas analyzer is being relocated to a lead shielded enclosure and the tubing is being sheathed in one inch of lead. Hydrogen analysis of containment gas sample will be done with the installed gas analyzer using thermal conductivity. There will be a back-up method using a grab sample obtained from the shielded bomb and run on gas chromatography. Isotopic analysis of the gas sample will be performed in the trailer based mobile count room using grab samples obtained from the shielded bomb. Liquid samples will be transported to the hot cell in a shielded carrier for analysis. Presently the boron analysis is by carminic acid (ASTM D3082 1975 Ed.).

New methods including Dionex Ion Chromatography and specific ion electrode are being evaluated. Chloride analysis is presently performed by turbidimetric analysis; however, Dionex Ion Chromatography is being evaluated. Isotopic analysis will be performed in the trailer based mobile count room using samples diluted in the hot cell. Another on site location for a back-up hot cell has been selected and the site is being prepared with appropriate equipment and shielding. A location for the mobile count room to park with shielding and services is also in the planning stages.

Provisions are being made to allow purging of the sample lines back to the affected units' containment. The remainder of sample lines that serve the new panels in the sample room are being sheathed with one inch of lead. There will be provisions to back flush the sample lines with either demineralized water or compressed nitrogen to remove obstructions. Flow restriction to limit reactor coolant loss is provided by sample line size, 3/8", and the containment isolation valves The sample points for liquid, loop B and RHR, and for containment gas are deemed representative of the areas required to be analyzed. The sample lines will be the ones originally installed which minimize fluid volume through tubing size and routing.

The sampling of the Reactor Coolant System and the Containment Building atmosphere could take place immediately after an accident without exceeding the criteria of GDC 19, Appendix A, 10 CFR Part 50.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Northern States Power Company submitted an outline of this training program on August 1, 1980. (J A Gonyeau, NSP, to Director of Nuclear Reactor Regulation, USNRC). It is our intent to initiate the program by April 1, 1981 and complete the initial program by October 1, 1981. Additional information on the training program will be provided if necessary.

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

As a sponsor of the EPRI PWR Safety and Relief Valve Test Program, Northern States Power intends to comply with the requirements of NUREG-0578, Item 2.1.2. By letter dated December 15, 1980, R C Youndahl of Consumers Power Company has provided the current PWR Utilities' positions on NUREG-0737, Item II.D.1 clarifications. Briefly those positions are:

A. Safety and Relief Valves and Piping - the EPRI "Program Plan for Performance Testing of PWR Safety and Relief Valves", Revision 1, dated July 1, 1980, does provide a program that satisfies the NRC requirements. Disucssion with the NRC staff and their consultants are resolving specific detailed issues.

B. Block Valves - The EPRI Program has not formally included the testing of block valves. However, a small number of block valves have been tested at the Marshall Steam Station Test Facility. The PWR Utilities and EPRI can not provide a detailed block valve test program until results of the Wyle and CE relief valve test are available. Therefore, a block valve test program will not be provided before July, 1981. The PWR Utilities and EPRI believe that the proper operation of the TMI-2, and Crystal River block valves and other operational experience, plus knowledge of the Marshall tests, support a less hurried and more rational approach to block valve testing.

C. ATWS Testing - PWR Utilities will not support additional efforts for ATWS valve testing until regulatory issues are resolved. The major safety and relief valve test facility (CE) is nearing completion and some measures were taken to provide additional test capability beyond the current program requirements. The NRC should recognize that results from the current program are likely to provide most of the information necessary to address ATWS events (i.e., relief capability at high pressures).

II.D.3 <u>DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION</u>

A resolution of this concern acceptable to the NRC Staff was described in our letter dated December 31, 1979. The only revision to the information submitted earlier relates to the environmental testing schedule.

The valve position monitoring system will be environmentally and seismically qualified to meet the requirements of a Babcock and Wilcox Owners Group. These requirements are more conservative than the IEEE 323 (1974) model or the Prairie Island requirements. The test program is managed for the Owners Group by Babcock and Wilcox. The testing will be performed by Approved Engineering Test Lab. A brief schedule for the testing of the monitor is as follows:

1. Initial Aging Analysis and Test Program Development	In Progress
2. Begin Irradiating of Components	December, 1980
3. Test Plan Review and Approval by Owners Group	January, 1981

4. Begin Testing — January-Feb., 1981

5. Finish Test Program — April-May, 1981

6. Finish Review of Report — June, 1981

The schedule dates were confirmed on December 2, 1980.

Since the displays and alarms are already installed in the control board, the human-factor analysis will be performed on the monitors in conjunction with the overall review of the plants main control board.

Proposed Technical Specifications were contained in our License Amendment Request dated December 19, 1980.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Design changes recommended by the NRC Staff in their letter of October 16, 1979 and in followup discussions are in progress. These include freeing the turbine driven pumps of AC power and providing pump suction protection. These changes will be completed at the first refueling outage of each unit following receipt of necessary equipment.

Improvements in condensate storage tank level instrumentation requested by the NRC Staff have been completed.

A re-evaluation of the auxiliary feedwater system flowrate design bases and criteria is being performed by Westinghouse. Our review of a preliminary report provided to us by Westinghouse indicated that substantial additional work is required to provide all of the information requested by the NRC Staff in Enclosure (2) to their October 16, 1979 letter. The re-evaluation should be completed by February 1, 1981. We will keep the NRC Staff informed of further delays in this work if they arise.

II.E.1.2 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

Part 1

Documentation of the reliability of the automatic actuation circuitry for the Prairie Island auxiliary feedwater system has been provided to the NRC Staff for their review in correspondence dated:

a) November 21, 1979
b) December 28, 1979 (included schematics and diagrams)
c) December 18, 1980

We have been responding to questions from the NRC Staff on an on-going basis related to this issue. We believe that no design modifications to the auxiliary feedwater system actuation circuitry will be required at Prairie Island.

Part 2

Our response dated December 31, 1979 stated that auxiliary feedwater flow indication to each steam generator would be upgraded to safety grade. This was a long term requirement indicated in the October 30, 1979 letter. Our present design satisfies the revised requirements in NUREG-0737 for Westinghouse plants since it provides one auxiliary feedwater flowrate indicator and one wide-range steam generator level indicator for each steam generator. Flow indication will not be upgraded as previously stated.

II.E.3.1 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

The Prairie Island pressurizer heater emergency power supply was described in our letters dated December 31, 1979, March 13, 1980, and April 11, 1980. This power supply was found acceptable by the NRC Staff in their Safety Evaluation dated April 18, 1980.

Training in use of the emergency power supply will be included in both the hot license and requalification license training programs for the Prairie Island plant.

Technical Specification changes were proposed in a License Amendment Request dated December 19, 1980.

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

Prairie Island has a redundant, safety-grade, and dedicated hydrogen purge system. No further action needs to be taken for this item.

II.E.4.2 CONTAINMENT ISLOATION DEPENDABILITY

Refer to our submittals of November 20, 1979, March 13, 1980, and April 11, 1980. Additionally, we have the following information to offer justifying the existing containment pressure setpoint.

We concur with the basic requirement that the containment setpoint pressure that initiates containment isolation of non-essential penetrations should be the minimum setpoint compatible with normal operating conditions. However, we cannot concur with the clarification statement that 1 PSI above normal operation pressure is an appropriate minimum pressure setpoint. This value is not sufficient to account for instrumentation drift and accuracy as well as abnormal operation which can increase containment pressure above normal but for which isolation of non-essential penetrations is not desirable. Containment indicated pressure may rise due to normal changes in atmospheric pressure. An increase in containment indicated pressure of approximately 0.75 PSIG will occur if atmospheric pressure decreases approximately 1.5" Hg. Containment pressure will also increase if containment temperature increases. A temperature increase of $40^{\circ}F$ can occur if the Containment Fan Coolers water supply is switched from chilled water to cooling water. Assuming ideal gas behavior, a temperature increase of $40\ ^{\circ}F$ will result in a pressure increase of 1.1 PSIG. It should be noted that the same pressure setpoint that actuates containment isolation also actuates Safety Injection and trips the reactor.

Proposed Technical Specifications were contained in our License Amendment Request dated December 19, 1980.

II.F.1 ACCIDENT MONITORING

Stack Gas Monitors

Post accident ventilation of the Auxiliary Building is made via the Auxiliary Building Special Ventilation System to the Shield Building ventilation stack. Radwaste Building Ventilation and Spent Fuel Pit Normal Ventilation will be isolated following an emergency. The Shield Building stack is equipped with one low range gas monitor supplied through an isokinetic sample nozzle. The high range stack gas monitor will utilize the same isokinetic nozzle as the low range monitor, but the high range monitor's flow will be directed to the Turbine Building where the sample goes through filter assemblies, a gas sample chamber and a blower which discharges back to the Shield Building Stack. The gas sample chamber is viewed by a Victoreen Model 845 Area Monitor which will indicate an alarm in the Control Room. The Control Room readout in mR/hr is converted to uci/cc Xe-133 equivalent with a calibration curve. A full scale reading will correspond to about $1x10^4$ uci/cc Xe-133 equivalent. The monitor and sample pump are supplied by a safeguards power supply.

The Victoreen model has a range of 0.1 mR/hr to 10^7 mR/hr with a sensitivity of ± 10% up to 3 mev x or gamma rays. The monitors will be checked monthly and calibrated at refueling intervals with solid sources traceable to initial gas calibration.

Interim procedures have been developed for utilizing direct radiation readings with several portable survey meters for determining release activity.

The new high range stack monitors will be installed and operable by 1/1/82.

Secondary Steam Relief Gas Monitors

Monitoring of the steam generator power operated reliefs and safety valves will be accomplished by an exposure rate meter viewing a section of steam line from inside a shield and collimator. Curves will be generated to compensate for isotopic ratio changes with shutdown time and to compensate for the shielding afforded by the steam line, guard pipe and insulation. Valve position indicators and flow devices will be used to determine the discharge flow rate from the relief and safety valves. Outputs will be available in the Technical Support Center for determining the quantity of activity released.

Interim procedures have been written for determining the amount of activity being released in an accident situation. This is accomplished with portable survey meters which view the steam line from a shielded collimator. The new instrumentation on the secondary side will be installed and operable by 1/1/82.

Secondary Side Air Ejector Discharges

The discharge of the air ejector condensors is being rerouted to the Shield Building stack where it will be monitored by the high range stack monitor. This will be completed by 1/1/82.

Interim procedures have been developed for quantifying the activity released using portable survey meters.

Sampling & Analysis of Plant Effluents

Post accident ventilation of the Auxiliary Building is made via the Auxiliary Building Special Ventilation System to the Shield Building ventilation stack. The Shield Building high range stack gas monitor in the Turbine Building will be equipped with two sets of particulate filters and charcoal absorbers. The shielded high flow filter and absorber are for low activity samples and the low flow filter and absorber are for high activity, 10^2 uci/cc, samples.

Isokinetic nozzles are employed in the Shield Building stack for accident situations and normal operations. An isokinetic flow splitter will divide the flow between the high range and low range stack samplers. Heat tracing on the iodine absorber sample line will add sufficient heat to ensure the sample relative humidity is less than 80%. The particulate and iodine samples can be counted in the Mobile Trailer. Several geometries have been made for counting high activity samples. The mobile counting system may use either plant power or the trailer generator. Presently, a sampling system with safeguards power is installed. The upgraded system, including the isokinetic nozzle design change, will be completed by 1/1/82.

Containment High Range Radiation Monitors

The high range Containment radiation monitors will be Class IE Safety Monitors qualified to LOCA conditions per IEEE 323-1974 and meet the requirements of NUREG-0578. The monitors, Model RP-23 from General Atomics, will have a range of 10^0 to 10^8 R/hr with readout units in the Control Room. The redundant monitors will be physically separated and view a major portion of the containment. The monitors will be type tested on five of the lower six decades for linearity. The monitors will also have a calibration test performed after installation. The monitors have installed sources to provide continuous readings and an electronic signal to provide about 1/2 to 2/3 of full scale reading to verify system operability. These monitors will be installed and operable by January 1, 1982.

Containment Pressure Indication, Containment Water Level Indication and Containment Hydrogen Indication

Our response dated December 31, 1979 stated that installation of containment pressure, water level and hydrogen monitoring equipment would be completed for both units by January 1, 1981. NUREG-0737 imposed new design requirements as well as new implementation dates. Design and installation, consistent with these new requirements, will be completed by the required implementation date of January 1, 1982 for both units.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Subcooling Meters

The installed subcooling meters and the plant process computer subcooling program are described in our submittals of December 31, 1979, March 13, 1980 and April 11, 1980. Until such time as the inputs to our subcooling meters are upgraded to meet Appendix B, the plant process computer subcooling program as described in the above referenced submittals will be used as the primary method of determining subcooled margin. The inputs to the subcooling meters, wide range RCS pressure and incore thermocouples, are scheduled to be upgraded as follows: Wide range pressure by Janaury 1, 1982 for both Units 1 and 2, and incore thermocouples by January 1, 1982 for Unit 1 and by the end of the February, 1982 refueling outage for Unit 2. Equipment availability precludes installation during the unit 2 refueling outage in February, 1981. We therefore request an extension in the completion schedule for Unit 2 to the end of the February, 1982 refueling outage.

Incore Thermocouples

The following addresses the items presented in Section II.F.2, Attachment 1, "Design and Qualification Criteria for Pressurized-Water Reactor Incore Thermocouples".

1) Thirty six core exit thermocouples uniformly distributed throughout the core along with 2 core inlet temperature RTD's are of sufficient number to indicate radial power distribution and coolant enthalpy rise. Calculations of core power distribution symmetry using incore thermocouples are available on demand.

2) The primary operator display of the core exit thermocouples is the plant process computer. This display has the following capabilities:

   a) A spatially oriented core map indicating individual thermocouple temperatures and temperature difference across the core is available on demand.

   b) Selective reading or continuous trending, visually or on a typewriter, of any thermocouple is available on demand.

   c) Capabilities a) and b) are available for thermocouple ranges from 200°F to 1375°F.

   d) Trend capabilities of any selected number of thermocouples is available on demand.

   e) Computer alarms will be developed to be consistent with operator procedure requirements when these procedures are developed.

   f) Future installation of new display devices will be human factor evaluated.

3) A backup display will be located on the incore instrumentation panel capable of reading greater than 16 operable thermocouples, 4 from each quadrant, in less than 6 minutes. This digital display will range from 200°F to 2300°F.

4) A human factors review of the core exit thermocouple displays will be conducted.

5) through 9) The instrumentation will be evaluated for conformance with Appendix B "Design and Qualification Criteria for Accident Monitoring Instrumentation" as modified by items 6 through 9 prior to installation. Any discrepancies identified by this evaluation will be transmitted at that time.

Level Instruments

Refer to our letter dated November 20, 1980, "Request for Relief from Implementation Schedule for NUREG-0737, Item II.F.2, Instrumentation for Detection of Inadequate Core Cooling."

Technical Specifications

Proposed Technical Specifications were included in our License Amendment Request dated December 19, 1980.

II.G.1 POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATORS

The Prairie Island power supplies for pressurizer relief valves, block valves, and level indicators was described in our letters dated December 31, 1979, March 13, 1980, and April 11, 1980. The plant design was found to be acceptable by the NRC Staff in their Safety Evaluation dated April 18, 1980.

II.K.1 IE BULLETINS

Further action on our part is awaiting the outcome of NRC review of our information submittals.

II.K.2.13 THERMAL MECHANICAL REPORT -- EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

To completely address the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, a program will be completed and documented to the NRC by January 1, 1982. This program will consist of analyses for generic Westinghouse PWR plant groupings.

Following completion of this generic program, additional plant specific analyses, if required, will be provided. A schedule for the plant specific analysis will be determined based on the results of the generic analyses.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEMS DURING TRANSIENTS

The Westinghouse Owners Group is currently addressing the potential for void formation in the Reactor Coolant System (RCS) during natural circulation cooldown conditions, as described in Westinghouse letter NS-TMA-2298 (T M Anderson, to P S Check, NRC). We believe the results of this effort will fully address the NRC requirement for analysis to determine the potential for voiding in the RCS during anticipated transients. A report describing the results of this effort will be provided to NRC before January 1, 1982.

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

The transient analysis code, LOFTRAN, and the present small break evaluation analysis code, WFLASH, have both undergone benchmarking against plant information or experimental tests. These codes under appropriate conditions have also been compared with each other. The Westinghouse Owners Group will provide on a schedule consistent with the requirement of item II.K.2.19, a report addressing the benchmarking of these codes.

II. K.3.1 AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM and

II.K.3.2 REPORT ON POWER-OPERATED RELIEF VALVE FAILURES

The Westinghouse Owners Group is in the process of developing a report (including historical valve failure rate data and documentation of actions taken since the TMI-2 event to decrease the probability of a stuck-open PORV) to address the NRC concerns of Item II.K.3.2. However, due to the time-consuming process of data gathering, breakdown, and evaluation, this report is scheduled for submittal to the NRC on March 1, 1981. As required by the NRC, this report will be used to support a decision on the necessity of incorporating an automatic PORV isolation system as specified in Item II.K.3.1.

II.K.3.3 REPORTING SAFETY VALVE AND RELIEF VALVE FAILURES AND CHALLENGES

Prairie Island will report, on a prompt basis, failures of pressurizer relief or safety valves. Prompt reporting is interpreted to mean within 24 hours by telephone (the same as LER reporting). Reporting to the Resident Inspector or Assistant Resident Inspector is considered adequate. If neither of these individuals can be contacted, the failure will be reported to the IE-III office. Documentation of failures and challenges will be included in a report covering the period 4-1-80 to 12-31-80 initially, and annually thereafter. The annual report will be submitted within 90 days of the end of the calendar year.

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS OF COOLANT ACCIDENT

The Westinghouse Owners Group resolution of this issue has been to perform analyses using the Westinghouse small break evaluation model (WFLASH) to show ample time is available for the operator to trip the reactor coolant pumps following certain size small breaks (see WCAP-9584). In addition the Owners Group is supporting a best estimate study using the NOTRUMP computer code to demonstrate that tripping the reactor coolant pumps at the worst trip time after a small break will lead to acceptable results.

For both of these analysis efforts, the Westinghouse Owners Group is performing blind post-test predictions of LOFT experiment L3-6. The input data and model to be used with WFLASH on LOFT L3-6 has been submitted to the staff on 12/1/80 (NS-TMA-2348). The information to be used with NOTRUMP on LOFT L3-6 will be submitted prior to performance of the L3-6 test as stated in letter OG-45 dated 12/3/80.

The LOFT prediction from both models will be submitted to the staff on February 15, 1981 given that the test is performed on schedule. The best estimate study is scheduled for completion by April 1, 1981.

Based on these studies, the Westinghouse Owners Group believes that resolution of this issue will be achieved without any design modifications. In the event that this is not the case, a schedule will be provided for potential modifications.

II.K.3.9 PROPORTIONAL INTEGRAL DERIVATIVE (PID) CONTROLLER

This modification was completed at Prairie Island when the two-out-of three low pressurizer pressure safety injection actuation logic change was made. The setpoint of the PORV interlock bistables was changed to 2335 psig. This in effect raised the permissive to the same setting as the trip setpoint.

Our derivative time constant in the PID controller for the PORV is set at zero which, in effect, removes the derivative action from the controller. Removal of the derivative action will decrease the likelihood of opening the PORV since the actuation signal for the valve is then no longer sensitive to the rate of change of pressurizer pressure.

II.K.3.10 PROPOSED ANTICIPATORY TRIP MODIFICATIONS

Prairie Island received NRC permission to include a P-9 permissive to omit reactor trip on turbine trip below a preset power level (50% was the original power level selected for P-9).

. II.K.3.10 requires delays in this type of modification until the small break LOCA probability analysis resulting from a stuck open PORV is completed and shows there is little effect from the addition of this modification. The setpoint on Unit 1 has been lowered to approximately 10% which is the present setpoint for the existing P-10 permissive. Unit 2 will be lowered from 30% to 10% during the next shutdown. This item is considered complete.

II.K.3.12 CONFIRM EXISTENCE OF ANTICIPATORY TRIP UPON TURBINE TRIP

Prairie Island has an anticipatory reactor trip on turbine trip. This item is considered complete.

II.K.3.17 ECCS SYSTEM OUTAGES

Prairie Island commits to submit a report by February 1, 1981 detailing outage dates, length of outages and cause of each outage on ECCS systems for the last five years of operation. The corrective action necessary to prevent or minimize outages will be specified.

An outage is defined as making an ECC system inoperable when coolant temperature is above 200°F. The ECC systems are the high and low head safety injection systems and the accumulators.

II.K.3.25 EFFECT OF LOSS OF AC POWER ON PUMP SEALS

In item II.K.3.25 the NRC defines loss of alternating-current to be loss of offsite power. The NRC also indicates if seal failure is the consequence of loss of cooling water to the reactor coolant pump seal coolers, one acceptable solution would be to supply emergency power to the component cooling pumps. Prairie Island's component cooling pumps are supplied from an emergency power supply. During July of 1980 offsite power was lost to Unit II and no abnormal reactor coolant pump seal leakage occurred.

No modifications are required to Prairie Island Unit I and II as a result of item II.K.3.25.

II.K.3.30 SMALL BREAK LOCA METHODS

NSP believes that our fuel vendors, Westinghouse and Exxon are the most appropriate group to work with the NRC Staff to resolve staff concerns with small break LOCA models for Prairie Island. We understand that the NRC Staff is holding meetings with the various vendors on this item. Accordlingly, the staff should continue to direct their questions regarding the scope and schedule for this requirement to the vendors. We believe that the November 15, 1980 requirement for a submittal on this item is now moot. It is our understanding that the vendors will submit this information in response to the NUREG-0737 requirement on a generic basis. NSP will file a letter with the NRC, if appropriate, to reference the vendors' submittals on the Prairie Island dockets.

II.K.3.31 COMPLIANCE WITH 10 CFR 50.46

If the results of the new models (and subsequent NRC review and approval) indicate that the present small-break LOCA analyses are not in conformance with 10 CFR 50.46, NSP will submit the plant-specific analyses using the revised models by January 1, 1983 or one year after any model revisions are approved by the NRC, whichever is later.

III.A.1.1 EMERGENCY PREPAREDNESS, SHORT TERM

It is our understanding that all short term requirements have been satisfied.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

In previous submittals we indicated our plans to upgrade the Technical Support Center (TSC) and Operational Support Center. Because of changing requirements, we will not meet our original commitment dates for upgrading the two facilities. Plans are being made to upgrade the Technical Support Center and Operational Support Center in 1981 as follows:

a) Physical Size and Staffing

The TSC will be sized to accommodate approximately 25 people. A minimum of 600 square feet will be available in the TSC for equipment and personnel. The main functional areas in the TSC will be: recording and display panels, plant process computer engineers consoles, conference area and communication cubicles.

The TSC is located in close proximity to the normal plant administrative area such that records of as-built conditions and drawings of structures, systems and components will be readily available. Also, normal administrative functions will be available as needed. Because the use of these functions will be limited and protective measures could be applied to their use, the administrative functions will not be included in the habitability envelope of the TSC.

The maximum staffing level for the TSC is anticipated to be as follows:

a. 5 NRC personnel including a site inspector

b. 3 Westinghouse members of the "site response team"

c. 1 Corporate Management Representative

d. 1 communications person

e. 1 administrative specialist

f. 11 members of plant management (Operations Committee)

g. The remaining personnel would consist of technical specialists from the plant staff or outside consultants as appropriate for the occurrence.

b) Building

The area being considered for the TSC is presently designated as the Administration Building Annex. The structure is located on the same elevation as the Control Room, approximately 100 feet away across the Turbine Room floor. Easy access from the TSC to the Control Room is, therefore, assured.

The area under consideration is large enough to house 25 persons, informational displays, radiation monitoring equipment and the necessary technical data.

The structure is constructed of concrete blocks and precast concrete panels. This structure has been modified to afford protection from direct radiation. A pressurizing-type ventilation system will be added which will include particulate and charcoal filtering.

c) Communications

A minimum of three direct telephone lines to the Corporate Office Telephone Exchange will be installed. These lines are part of Northern States Power's microwave communications system and do not depend upon the plant telephone system or the local telephone system. In addition, a multi-channel radio phone will be installed in the TSC. The radio will have the ability to communicate with the following:

1. Sheriffs Department
2. NSP Dispatcher and State Emergency Operations Center
3. Portable Radios (Plant)
4. Mobile Radios (Trucks, etc.)

d) Information Display and Storage

The engineers computer console will be utilized to provide access to the parameters needed by the TSC staff to assess the consequences of and supply engineering support to the control room operating staff following an accident or severe transient in the nuclear steam supply system.

A series of graphic displays in the form of simplified flow diagrams has been developed for the systems or components necessary to follow the course of an accident or mitigate the consequences of an accident. Major instrumentation and valves will be shown on these diagrams along with their computer addresses. A matrix will also be developed to cross reference the computer addresses with the 1) instrument identification number 2) control board identification number, and 3) instrument range.

Vital information will be available to the Technical Support Center via the engineer's console and a data acquisition system. Key parameters needed to analyze the severity and course of an accident will be historically available in the Technical Support Center starting at the time of the event.

e) Operations Support Center

The Operational Support Center will be upgraded by installation of a multi-channel intercome system to enhance communications between the Operational Support Center, Technical Support Center and the Control Room.

III.A.2 EMERGENCY PREPAREDNESS

NSP upgraded emergency plans were previously submitted. NRC comments were received and plans are being revised to accommodate comments. The plan (as revised) will be implemented by April 1, 1981.

The applicable State and Local plans have been submitted to the NRC according to Part 50.54 (s)(1) by our letter dated December 18, 1980.

The NSP emergency plan implementing procedures are in preparation and will be submitted by March 1, 1981.

Study and investigation work is being carried out for proposed meteorological systems. Because some of the regulatory guidance has not yet been finalized (Reg. Guide 1.23 and NUREG-0696) only a tentative plan can be submitted by February 1, 1981 describing staged implementation of meteorololgical systems to meet the full operational capability required by June 1,1983.

III.D.1.1 PRIMARY COOLANT OUTSIDE CONTAINMENT

A program acceptable to the NRC was described in our letters dated December 31, 1979 and March 13, 1980. Proposed Technical Specifications were submitted to the Commission on December 19, 1980.

III.D.3.3 IMPROVED INPLANT RADIO-IODINE MONITORING

This requirement has been satisfied.

The Technical Support Center and the Control Room are supplied with iodine and particulate continuous air monitors (CAM's) for post-accident air monitoring. These CAM's are equipped with silver zeolite cartridges. The iodine portion of the CAM's are equipped with single channel analyzers set for the I-131 peak at 365 Kev.

Portable samplers are available for obtaining iodine and particulate samples in any other essential area of the plant. These samples will be collected on silver zeolite cartridges (these cartridges are stored in plant emergency lockers) and removed to a low background area for analysis.

A mobile Ge (HP) detector system is available which is mounted in a trailer and equipped with a 220 volt connection and portable generator. This system enables the plant technicians to accurately analyze samples at the plant or at a desired remote distance from the plant (depending on background readings).

Operating procedures are available. Training has been accomplished on the mobile Ge (HP) system, portable samplers, CAM's, and use of silver zeolite cartridges.

III.D.3.4 CONTROL ROOM HABITABILITY

The control room emergency air treatment system will be evaluated in accordance with the latest NRC criteria as specified for this item. Modifications which may be required to meet NRC requirements will be identified and a schedule for their completion will be provided if necessary.

The required evaluation, description of modifications, and schedule will be submitted for NRC review by February 1, 1981.