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50-306 Prairie Island Nuclear Station, Unit 2, Northern Stat 05000306

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Office of Nuclear Reactor Regulation

SUBJECT: Forwards response to NRC 800507 ltr on addl TMI-2 requirements re shift manning,licensing exams,operating experience feedback,Bulletins & Orders Task Force recommendations, & control room habitability.

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

June 11, 1980

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

Additional TMI-2 Related Requirements

Mr Darrell Eisenhut's May 7, 1980 letter identified five new TMI-2 requirements related to shift manning, licensing examinations, operating experience feedback, B&O Task Force final recommendations, and control room habitability for the Prairie Island Nuclear Generating Plant. The "shift manning" requirements (Item I.A.1.3) have not yet been identified by the NRC staff.

Attachment 1 provides our response to four of the five Items of the May 7 letter. As noted, we intend to comply with the requirements and implementation dates specified except in the cases of items I.A.3.1, II.K.3.1, II.K.3.5, II.K.3.10, II.K.3.30, and II.K.3.31. Bases for the delay in implementation dates or exceptions are provided in the Attachment 1 discussion of those items.

Our agreement to meet the implementation dates specified in the May 7 letter is dependent on equipment availability and assumes no changes in regulatory position beyond those stated in the May 7 letter.

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

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/JAG/ak

cc: J G Keppler
G Charnoff

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Attachment 1 to
June 11, 1980 NSP Letter

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Docket No. 50-282	License No. DPR-42
50-306	DPR-60

Five Additional TMI-2 Related Requirements

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Item I.A.1.3 Shift Manning

Response postponed until 30 days after receipt of the NRC letter spelling out shift manning requirements.

Item I.A.3.1 Revised Scope and Criteria for Licensing Examination

Northern States Power Company commits to the requirements and schedule with the following exceptions:

(1) A.1.a. Experience

"Exceptions can be made as determined by the Training Supervisor provided at least four years of power plant experience and two years of competent operating experience, including a minimum of 1 year of control room related experience, have been satisfied."

We feel this is a justifiable position based on experience gained with the Nuclear Plant Operator Training Programs. At least 1 year of control room related experience is a minimum requirement.

(2) A.2.b. Training - Control Room Operator

"Staff engineer applicants for an operator's license shall have 3 months of control room related training."

We feel this is a justifiable position for staff engineers. The intent is not to limit control room training but to provide flexibility.

(3) "Effective date: Present programs have been modified in response to Bulletins and Orders. Revised programs should be submitted for OLB review by October 1, 1980".

We feel the above extension of the effective date is necessary to allow additional preparation of the formal program submitted in light of the increased workload associated with item A.2.d.

(4) A.2.d. Training - Instructor Competence

"Effective date: Applications should be submitted no later than December 1, 1980 for individuals who do not already hold a senior operator license."

We feel the above extension of the effective date is necessary to allow additional time for exam preparation.

We agree that training instructors must be expert in their course of instruction - systems, integrated responses, transient and simulator courses. A senior reactor operator or shift technical advisor knowledge level is generally required for instruction in integrated responses, transient, and simulator courses. Our 10 years of experience in training operators and engineers for RO and SRO licenses have shown that in some cases the expert knowledge possessed by plant systems engineers or accident analysis engineers (who may not possess an SRO) can provide a greater in-depth training benefit to the students. In addition, we have found that selected training consultant personnel (e.g. former NRC operator licensing examiners or others who may not possess an SRO license) have been able to provide very capable instruction in systems, integrated responses, and transients. Thus we feel this requirement is overly restrictive.

In addition, we interpret this requirement to mean that personnel who have passed an SRO examination previously and whose licenses may have lapsed (but have been engaged in nuclear power plant operation and/or support activities) have demonstrated "their competence to NRC by successful completion of a senior operator examination."

"We agree that any individuals who provide training instruction in systems, integrated responses, transient and simulator courses at the NSP nuclear plants should be or have been NRC licensed or have equivalent knowledge level (e.g. systems engineers). Exceptions (as noted above) would be granted only by the Manager-Production Training or designate who holds or has held a Senior Reactor Operator License for a large light water reactor."

(5) A.2.e. Training - Requalification Programs

"Effective date: Programs should be initiated May 1, 1980. Programs should be submitted to OLB for review by December 1, 1980."

We feel the above extension of the effective date is necessary to allow additional preparation of the formal program submittal. The Training Group's workload is taxed especially with the A.2.d position requiring extensive training within the training group.

(6) A.3. Facility Certifications

"Certifications completed pursuant to Sections 55.10(a)(6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be signed by the plant managers."

It is felt that the plant manager is a more appropriate level of authority to attest to the validity of the license applications. Higher levels of corporate management would have significantly less personal knowledge of the validity of the application and the capability of the applicant. We propose that the past practice of issuing such items under signature of the plant manager be retained.

(7) B.1, D.1, D.2 and D.3

These items require further action by the NRC and are therefore not appropriate for commitments by the licensee at this time.

Item I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

Northern States Power Company commits to the requirement and schedule. Presently, a system exists that assures distribution of pertinent information important to plant safety. This operating experience assessment function will be reviewed in light of the position; modifications, as necessary, to this system will be completed by 1-1-81.

Item II.K.3.1 Installation & Testing of Automatic PORV Isolation System

We do not believe an automatic PORV isolation system should be required. This is based on Westinghouse Owners Group analyses of the ultimate heat sink function, and the decreased intensity of a number of plant transients, given the PORV(s) operation. Failure of the proposed automatic PORV isolation system could impair this function. In addition, the plant modifications, procedure changes, and operator training (e.g., NUREG-0578 requirements) provide assurance that the function of the automatic isolation system will be provided by operator action. In addition, failure to isolate stuck open PORV(s) has been analyzed and results in no core uncovery.

Item II.K.3.2 PORV Failure Report

A report on PORV failure reduction will be submitted to the NRC by January 1, 1981. It is currently anticipated that this report will be in the form of a generic Westinghouse Owners Group submittal.

Item II.K.3.3 Reporting Safety & Relief Valve Failures and Challenges

Prairie Island agrees to 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 can be contacted, the failure will be reported to the IE-III office. Documentation of failures and challenges will be included in an annual 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.

Item II.K.3.5 Automatic Trip of Reactor Coolant Pumps During LOCA

In our response to IE Bulletin 79-06C, we reference the Westinghouse Owners Group analysis of delayed RCP trip during small break LOCAs documented in WCAP-9584. This WCAP is the basis for the Westinghouse and Owners Group position on RCP trip (i.e., automatic RCP trip is not necessary for a Westinghouse PWR since sufficient time is available for manual tripping ofthe RCPs). This philosophy has been incorporated in the Westinghouse Emergency Operating Instructions which were reviewed and approved by the NRC Bulletins and Orders Task Force and subsequently incorporated in the plant emergency operating procedures. In addition the Westinghouse criteria (basically a RCS pressure below the shutoff head of SI pumps) provides for continued RCP operation and therefore forced circulation and decreased reliance on operator action for non-LOCA events. As requested by the NRC in a letter dated April 15, 1980 and as discussed with the NRC during the May 22, 1980 meeting on this subject, we anticipate that the Westinghouse Owners Group will provide predictions of the LOFT test L3-6. The NRC has indicated that small break tests at the Semiscale and LOFT facilities, as well as Owners Group test predictions, will aid in NRC resolution of this issue. Therefore, we believe that it is not appropriate to take any additional actions on this issue until the results of the NRC sponsored testing (in particular L3-5 and L3-6) and Owners Group predictions are completed and the results evaluated.

Item II.K.3.9 Proportional Integral Derivative (PID) Controller Modification

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.

Item II.K.3.10 Proposed Anticipatory Trip Modification

Prairie Island has modified unit #2 to include a 50% P-9 permissive below which omits reactor trip on turbine trip. The modification was reviewed by the NRC. The setpoint was placed at 30%. The modification on unit #1 is near completion.

The position requests delays in this type of modification until small break LOCA probability analysis resulting from a stuck open PORV is completed and shows there is little effect by the addition of this modification. It is our intention to complete the P-9 modification on unit #1. The setpoint on both units will be lowered to approximately 10% which is the present setpoint for the existing P-10 permissive.

Item II.K.3.12 Confirm Existence of Anticipatory Trip Upon Turbine Trip

Prairie Island has an anticipatory reactor trip on turbine trip.

Item II.K.3.17 Report of Outage of ECC Systems

Prairie Island commits to submitting a report detailing outage dates and lengths of outages for our ECC systems for the last five years of operation. The report will include the causes of the outages. Our interpretation of ECC systems includes the high head injection system, low head injection system and the accumulators. The report will be submitted by 1-1-81.

Item II.K.3.25 - Effect of Loss of AC Power on Pump Seals

We understand that this item stems from NUREG-0626 which specifically addresses BWR plants and is not applicable to Prairie Island. No response is required.

Item II.K.3.29 Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles

We understand that this item stems from NUREG-0626 which specifically addresses BWR plants and is not applicable to Prairie Island. No response is required.

Item II.K.3.30 Revised Small Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K

The present Westinghouse small break evaluation model used for the Prairie Island NGP is in conformance with 10 CFR 50 Appendix K. Analyses previously reported (April 14, 1975) demonstrated that Appendix K criteria were met with a minimum Peak Clad Tmperature (PCT) margin of almost 500F to the 2200F limit for the worst calculated case. This value was about 450F less than the PCT reported for the large worst case break LOCA at that time. These analyses were conducted with higher reactor thermal power, linear heat rate, and F_Q (102% of 1722 Mwt, 102% of 14.54 kw/ft 2.32 respectively) than currently allowed by the Technical Specifications (1650 Mwt, 14.31 kw/ft, 2.21). Thus there remains adequate assurance that the Prairie Island NGP is well in compliance with 10 CFR 50 Appendix K. The K(Z) curve used in the technical specifications provides additional assurance that the small break LOCA is not limiting.

Westinghouse has indicated that they do plan to address the specific items of interest noted in NUREG 0611 in a forthcoming model change planned for submittal by January 1, 1982.

We do not believe that it would be appropriate for our current fuel vendor (Exxon Nuclear Company) to submit revised small break LOCA methods for Prairie Island Units 1 and 2. As noted above, previous analysis by our NSSS vendor, Westinghouse, has shown that the small break LOCA is definitely non-limiting from the point of view of the fuel and, therefore, that core thermal power limits--the sole LOCA analysis area in which our fuel vendor has been involved--are completely defined by the large break LOCA. Furthermore, it is well known that fuel characteristics are of very secondary importance in determining the plant response to a small break LOCA; the most important parameters being the licensed core power level, the performance characteristics of the ECCS, the normal primary coolant loop operating temperature, and the elvation of the core with respect to the hot and cold legs. In the unlikely circumstance that the small break LOCA should become limiting, either due to the discovery of previously unknown phenomena or as a result of additional licensing conservatisms, it would of course then become necessary for our fuel vendor to provide suitable documentation for its methods.

Item II.K.3.31 <u>Plant Specific Calculations to Show Compliance with 10 CFR 50.46</u>

Previous analyses, noted in our response to Item II.K.3.30, demonstrate that the Prairie Island plant is in conformance with 10 CFR 50.46 and that the small break LOCA is not the limiting event that the large break LOCA is. Thus we do not believe that additional plant specific calculations are required.

Item II.K.3.44 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure

We understand that this item stems from NUREG-0626 which specifically addresses BWR plants and is not applicable to Prairie Island. No response is required.

Item III.D.3.4 Control Room Habitability

Prairie Island commits to the requirement and will submit a schedule by 1-1-81 and will modify the control room as necessary by 1-1-83.