SEP 2 2 1986

Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Midland Square, 4th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

On July 28 and 29 1986, the NRC staff (Dr. S. N. Saba (EICSB) and the Prairie Island Project Manager) met with your staff at the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 for the purpose of resolving NRC's concerns related to the Detailed Control Room Design Review (DCRDR). As a result of this site visit, we have concluded that all open issues associated with Item I.D.1.2 of NUREG-0737 (DCRDR) have been resolved. The resolution of some issues is based on your staff's commitments as described in Section 6 of the meeting summary. Based on the resolution of these previously open issues, we can now proceed with the completion of our safety evaluation on this matter by October 1986.

Enclosed is a copy of our meeting summary.

/s/

Dominic C. DiIanni, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosure:
As stated

*SEE PREVIOUS CONCURRENCE

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Prairie Island Nuclear Generating Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos: 50-282
and 50-306

LICENSEE: Northern States Power Company

FACILITY: Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP)

SUBJECT: MEETING SUMMARY OF JULY 28 AND 29, 1986 WITH NORTHERN STATES POWER COMPANY (NSP)

On July 28 and 29, 1986, the NRC staff met with representatives of Northern States Power Company for the purpose of resolving the NRC's concerns related to the Detailed Control Room Design Review (DCRDR) for the Prairie Island Nuclear Generating Plant (PINGP), Unit Nos. 1 and 2. The meeting was held at the plant site for the purpose of resolving the remaining open issues dealing with our review of the Prairie Island DCRDR and to inspect the mock-up control room, the plant simulator, and the main control room as these facilities relate to this licensing issue.

The attendees were as follows:

John Goldsmith	NSP/NTS (PINGP)
Dayle Althaus	NSP/NTS (PINGP)
William R. Waldron	NSP/NTS (PINGP)
Ray Rogers	NSP/NTS (Monticello)
Christopher Koch	Honeywell (Tech Strategy Center)
Stephen Metz	Honeywell (Tech Strategy Center)
Saba N. Saba	NRR/PAD-1 NRC
Dom DiIanni	NRR/PAD-1 NRC PM

The Technical Evaluation Report (TER) No. SAIC-86/1076 prepared by Science Applications International Corporation (SAI), the NRC's consultant, covered the review of the licensee's submittals including the licensee's summary report transmitted by letter dated December 31, 1985. The TER contains open issues that required further information so that review of this licensing action can be completed. The licensee by letter dated June 12, 1986 provided a supplemental response to our concerns and this meeting served to further clarify this latest response by the licensee. The licensee addressed our concerns as follows:

1. System Function and Task Analysis (SFTA)

<u>The NRC Concern:</u>

° A review of the Element Tables and the Instrumentation and Control Requirements Tables determined there is inadequate documentation for the reviewers to identify how plant specific information and control characteristics were derived from background documentation of Revision 1 Emergency Response Guidelines (ERGs) or from plant specific information. NUREG-0800, Appendix A, Section 2.2(3), titled, "Use of Function and Task Analysis" is clear in describing the level of detail beginning with, "Analyze the operator tasks to determine the characteristics of information or control capability needed to perform the task." PINGP should provide the above information in a supplement to the Summary Report or be prepared to discuss the SFTA process at a meeting with the NRC.

- The reviewers require an explanation of how needed plant specific characteristics for operator information and control needs were developed and recorded (an auditable record) at the task and subtask level.

° The Licensee's Responses:

The needed operator information and characteristics of Prairie Island Control Room instrumentation were determined independently of the existing control room by using the Revision 1 of Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs) as background information in the following manner:

a. A study was performed to compare plant specific instrumentation and control characteristics with the generic plant instrumentation and control characteristics. This comparison involved plant systems and instrumentation and controls.

b. The chronology of the review of instrumentation availability was as follows: Information and control requirements from the WOG Generic Plant Step Description Tables and Element Tables were used to define the characteristics and criteria for required instrumentation at Prairie Island. Those criteria were then used to generate the plant specific element tables. Finally, the control room was then checked to determine that the existing plant instrumentation and controls meet the required characteristics.

c. The existing Emergency Operating Procedures (EOP's) were reviewed to determine if additional instrumentation and controls not present in the generic plant requirements would be necessary to support emergency plant conditions at Prairie Island.

d. The existing control information available was not used in the development of the element tables related to the instrumentation and control requirement.

° The NRC's Position

The licensee confirmed that their consultant independently established the information and instrumentation needs without using (or being in the) control room to find what or where instrumentation existed at this stage of the DCRDR. The NRC discussed and the licensee clarified the Figures of Appendix B of the June 12, 1986 submittal.

Based on the discussions, clarifications, review of documents at the meeting and the above response, the licensee has adequately satisfied NRC concerns dealing with SFTA and whether it was done independently.

2. Control Room Validation

The NRC Concern:

° During the validation of control room functions, several human engineering discrepancy (HEDs) were generated involving inadequate personnel staffing, extreme interpanel movement between tasks, and inadequate display/control relationships. In addition, these HEDs involved three tasks in the EOPs (See Table 3-17 of the Summary Report). A total of 31 HEDs were identified. Twenty-one of these HEDs were identified as requiring "no corrective action" and seven HEDs were corrected by future SPDS implementation, which has been deemed an inappropriate solution. PINGP is to reassess the three EOP tasks under real-time dynamic conditions to evaluate fully the cumulative and interactive effects of these HEDs. All 31 HEDs should be resubmitted with appropriate reassessment documentation and justifications for corrective actions.

° The Licensee's Response:

The licensee addressed all 31 HEDs in the supplemental response (Appendix B) submitted on June 12, 1986. Based on this supplemental information the licensee was requested to clarify the response on 9 of the 31 HEDs which were reassessed. The clarification on the 9 HEDs are as follows:

a. HED 048X

The licensee established a priority 2 indicating that the correction must be completed as soon as possible. The problem associated with excessive operator movement will be completed as part of SPDS installation by December 31, 1986. The NRC finds the schedule for resolving this issue acceptable.

b. HEDs 015X and 016X

These HEDs deal with the control room work space, work station design, and environment in the control room. The inspection of the control room by the NRC revealed that a problem does not exist with instrument readouts and that the workspace appears adequate.

c. HED 004X

The concern is that the control/display are too far away to be seen by operator from panels C and B. This was reviewed at the simulator. The locations of the SI push button on Panel B and the reset status light on Panel C are adequate and acceptable.

d. HEDs 005X, 017X and 027X

The NRC concern is how far away the displays are from the controls. The on-site checking showed that the seal flow and charging flow indications are adequate and in the line of sight of the operator.

e. HED 024X

The two components involved in this HED are related to cooldown (Priority 3). The operator movements during cooldown should be minimal. The locations of the reactor coolant system (RCS) and steam generator (SG) 1A and 1B pressure controllers are marginal. On-site checking showed that the RCS temperature and steam pressure controllers are adequate and in the line of sight of the operator.

f. HED 021X

To resolve this HED, the turbine trip push button will be relocated during the "E" panel modification in July 1989 for Unit 1 and in March 1990 for Unit 2. These dates were coordinated with future reactor refueling outages and were found acceptable by the NRC.

3. Control Room Survey

The NRC concern can be summarized in that PINGP used survey guidelines other than NUREG-0700 for the control room which could have resulted in a less complete survey. The licensee confirmed that NUREG-0700 guidelines (Section 6) were used. The discussion of this item clarified the early use of NUTAC documents as a reference for the technique in conducting the surveys.

The document prepared by Honeywell Inc. for PINGP in November 1984 "Evaluation of Design Conventions Specifications Against NUREG-0700 Guidelines" identified 14 deviations from NUREG-0700. Of the 14 deviations, five of them were of concern to the NRC. The following is clarification, justification, and resolution of these five items.

a. Item 3 of the deviation: NUREG-0700, Guideline 6.4.4.5d(1) is related to two and three position switch functions and position indications. At Prairie Island, the Westinghouse OT-2 switches are used and the operators have been trained to be fully aware of the functions and positions of these control switches. These switches were examined by the NRC at the control board and we found the justification acceptable.

b. Item 5 of the deviation: NUREG-0700, Guideline 6.4.5.1(2)b is concerned with the trough distances for manipulations of thumbwheels on the controllers. The licensee's explanation and the convention that exists at Prairie Island is that the trough distance of 1-1/8 inch allows adequate space for thumbwheel operation. The NRC examined the trough distance of controllers in the control room and found the convention used at Prairie Island acceptable.

c. Item 9 of the deviation: NUREG-0700, Guideline 6.5.3.1c(1) is related to monitoring permissive indication of starting electric motors. The "Large Motor Monitor" system at Prairie Island to alert the operator on starting 250HP and larger motors is of the "permissive" indication and not alarm condition and is acceptable.

d. Item 11 of the deviation: NUREG-0700, Guideline 6.6.2.4c. This issue deals with labels visibility during actuation of controllers. Label visibility was observed in the control room. The convention employed at Prairie Island is consistent with industry standards and the design of the controllers is such that the operators are not confused with the position indicator even if the label "auto" is momentarily covered by the finger. Thus, the NRC finds the label visibility acceptable.

e. Item 12 of the deviation: NUREG-0700, guideline 6.6.3.8a is related to Control Position Labeling. It states, that "all discrete functional control positions should be identified." The subject of this convention is the Westinghouse OT-2 rotary selector controls that have two discrete functional control positions which are labeled and a spring-loaded center position which is unlabeled. (An example of this control is diagrammed in the "Control Board Standards" on page 2-17). For these controls, the center position is not a discrete functional position--it indicates only the absence of an "open" or "close" signal for motor valves. If the valve may be activated automatically, then the center position is functional. For such valves, the "Control Board Standards" directs that the center position be labeled "AUTO" (e.g., page 2-15). The convention as currently stated in the "Control Board Standards" is appropriate. In conclusion, the justifications for deviations from NUREG-0700 guidelines and the use of NSP "Control Board Standards" in the survey are acceptable.

4. Human Engineering Discrepancies HEDs Appearing in the Report Issued by the Licensee Entitled "Detailed Control Design Review Summary Report" December 1985

The NRC had additional concerns on the HEDs appearing in the Summary Report. These concerns on the specific HEDs were discussed with the licensee and the results are summarized below.

a. HED 006V, (Items 46316 and 46317)

HED 006V deals with establishing distinctive enhancement for rotary selector knob used for emergency controls. The concern related to this HED is a matter of priority 2 and 3 for motor operated valves in the auxiliary feedwater system. The licensee acknowledged that there is an error in the report and the priority should be consistent for all components of the auxiliary feedwater system. Priority level 2 should be applied to item 46316 which would be consistent with item 46317. The licensee acknowledged the error and correction will be made to to the internal working documents. The licensee confirmed that the reactor and turbine trip switches will have blaze orange backplates as an enhancement.

b. HED 034V

HED 034V deals with the adequacy of the range scales of auxiliary feedwater (AFW) flow rate readout instrumentation to assure that the expected operations range is covered by an appropriate scale range of the readout instrumentation. The NRC questioned the adequacy of the scale range AFW flow to steam generators 11 and 12. The licensee indicated that HED No. 034V pertaining to AFW flow is superseded by HED 044V that takes into account the requirements of Reg. Guide 1.97. The NRC finds that HED 044V resolves the concern of scale adequacy for AFW flow and, therefore, the resolution is acceptable.

c. HED 043V-5

This HED is concerned with the condition of the reactor coolant pump, that is, what information is required in emergency operating procedures pertaining to lower bearing water temperature, seal outlet temperature, labyrinth seal differential pressure, and seal injection temperature that are the parameters measured to determine pump condition. The NRC considers the justification for correcting this HED was not adequate since the justification did not address the parameters ranges for off-normal conditions. The licensee indicated that the operators are instructed to refer to the plant operating procedures that give normal and alert ranges for these normal operating parameters. This information appearing in the operating procedures was obtained from the pump manufacturer. On this basis, the licensee considers adequate justification exists for not correcting this HED. The NRC agreed with the position taken by the licensee.

d. HED 007-R

This HED raises the concern that the computer alarm auditory and visual display may not be adequate to get the operators attention. The NRC considered the justification given in the summary report was not adequate. The licensee indicated that if the alarm is not acknowledged or is overlooked, there is not a safety consequence since other backup indications are available to alert the operator. The NRC considered this additional justification as adequate for not correcting this HED.

e. HEDs 128C and 129C

These two HEDs were concerned with the possible confusion arising in the control position of the rotary "T" handle control switches relative to the position markers. The NRC, after inspecting the "T" handle control switches in the control room, judged that the marker locations for the various positions of the "T" handle control switches are adequate to avoid confusion by the operator. In addition, operators do receive training in the operation of these switches. On this basis, the NRC agreed with the licensee that correction of these HEDs are not necessary.

The above HEDs were resolved as well as other HEDs that were resolved by NSP and were found acceptable by NRC (i.e., Appendix A and B of April 25, 1986 TER).

5. NSP's Response, in Letter Dated June 12, 1986 "Appendix B"

By letter dated June 12, 1986, the licensee submitted Appendix B that esponded to the open issues appearing in the draft TER prepared by Science Application International Corporation. The licensee's response (Appendix B) to the open issues appearing in the TER was reviewed by NRC and was found adequate except for the following areas in the Conclusion and Recommendations section that needed further clarification.

a. Item 5, Appendix B, page 8 "Selection of design improvement"- Concerns were raised on how PINGP dealt with possible cumulative effects from HEDs, some of which might be low priority. Based on the licensee's response appearing in Appendix B, the procedure to review the cumulative effects by walk-throughs and on the NRC's inspection of the mock-up control room, the simulator, and the reactor control room; the NRC has judged that the cumulative effect from HEDs covering topics of annunciator system design, legends, and pushbuttons and label locations has been adequately addressed, and that the selection of design improvement is adequate.

b. Item 6, Appendix B, page 9 "Coordination of DCRDR Efforts"- Item 6 deals with the description of the licensee's coordinated efforts and methods used to develop the Integrated Plan in response to Generic Letter 82-33 for all the control room review elements. The NRC found the licensee's response acceptable after obtaining minor clarification during the meeting.

c. Item 7, Appendix B, page 9 "Schedules for Implementation of HED Corrections"-Item 7 requested the licensee to submit implementation schedules and a summary justification for HEDs having safety significance that remain uncorrected. The NRC found the licensee's response for this item is acceptable after minor comments were clarified during the meeting.

6. Licensee's Commitments

a. The licensee committed to have all 128 HEDs that relate to the control board standards evaluated as being safety or non-safety related. In addition, the licensee committed to complete modifications to meet requirements for labeling, sealing, and standard abbreviations as defined in the control board standards. This work will be completed by December 31, 1990. The NRC found this scheduled completion date acceptable.

b. The licensee committed to have the Control Board Standard (dated March 24, 1986) as a living and controlled document to be used at Prairie Island as a guide for future control room modifications.

c. The licensee committed to have on file the NUREG-0700, "Guidelines References" for the HEDs appearing in Section D of the summary report where applicable.

Conclusion

Based on the above the NRC concluded that all open issues for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 have been resolved for DCRDR. The resolution of some of the open items are based on commitments by licensee as described above.

Summary Prepared by: Dominic C. DiIanni and Dr. Saba.