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May 10, 2002

U S Nuclear Regulatory Commission
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Washington, DC 20555

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

Prairie Island Nuclear Generating Plant Response to Draft NRR Position on TIA 2001-04, "Design Basis Reliance on Non-Seismic and Non-Safety Related Equipment" (TAC Nos. MB1855 and MB1856)

By letter dated March 18, 2002, Prairie Island requested a meeting with the NRC staff to discuss the subject TIA and TIA 2001-02. The meeting was conducted on April 12, 2002. In this meeting we agreed to submit to the NRC our comments on the draft NRR staff position on TIA 2001-02 and TIA 2001-04. The attachment to this letter is our detailed comments on the draft NRR staff position on TIA 2001-04.

In this letter we have made no new Nuclear Regulatory Commission commitments. Please contact Jeff Kivi (651-388-1121) if you have any questions related to this letter.

M. K. Nazar

Mano K. Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC

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Attachment

Attachment

May 10, 2002

Prairie Island Response to Draft NRR Position on TIA 2001-04
"Design Basis Reliance on Non-Seismic and Non-Safety Related Equipment"

The two concerns addressed in the NRC draft response to TIA 2001-04 are:

- The Cooling Water (CL) system return headers are credited as a flow path as part of the resolution to USI A-46. These return headers are not seismically qualified.

- Non-safety related valve operation is relied on to reduce flow demand on the CL system in the evaluation for a loss of off-site power (LOOP).

This letter responds to the second item; i.e., credit for non-safety related valve operation. The response for the first item is included with the response to TIA 2001-02.

In response to the second item the draft NRC response states that it is outside the plant design basis to credit the operation of these non-safety related valves. This letter addresses the draft NRC response based on a review of the basis for our conclusion that the plant is operating within its licensing basis and a discussion of the significance if failure of these valves is assumed.

I. Licensing Basis

During the initial licensing of Prairie Island, a concern was raised by the AEC regarding the ability of the CL System to withstand a LOOP with a coincident single active failure. The specific event scenario and the related correspondence is described in detail in our letter to the NRC, "Response to Opportunity to Comment on TIA 2001-04, Design-Basis Reliance on Non-Seismic And Non-Safety Related Equipment," dated September 17, 2001. The evaluations documented in this correspondence (both NSP and AEC) credit the Turbine Generator Hydrogen Cooler Temperature Control Valve (referred to as 'TCV' in the remainder of this discussion) in each unit with automatically closing following the turbine trip and reducing the load on the CL system. From this correspondence it is evident which valves were being credited; e.g., by valve number, description and function.

The draft TIA response states "the staff found no explicit recognition or acknowledgement of this ('this' being the AEC awareness that the valves were non-safety related) during plant licensing by either the licensee or the NRC." *[underline added]* This statement is not inconsistent with our response in the letter, dated September 17, 2001. We recognize that the correspondence does not explicitly state that the valve is non-safety related. However, within this information, there are several instances that point to these valves being non-safety related, including:

- The valve numbers and valve description are explicitly discussed in the correspondence. These valves are shown on the CL system flow diagrams included in the FSAR; which show that these valves are non-safety related.
- The correspondence describes the valves as the temperature control valves for the turbine hydrogen coolers. Per FSAR, Appendix B, the turbine and auxiliaries are non-safety related.
- The correspondence also describes these valves as being part of the turbine building cooling water system. The turbine building cooling water system is non-safety related.

Furthermore, the correspondence makes reference to meetings held between NSP and the AEC to discuss this concern. A focus of one of the meetings was to discuss the function of these temperature control valves. Aside from the information documented in the AEC letter dated April 25, 1974, we do not have any additional notes from the meeting. However, at a typical meeting to discuss a technical issue or concern, there is in depth discussion exploring various relevant facets of the concern. Considering that the meeting documentation discusses the valve's function as reducing flow to the heat exchangers associated with the main steam turbine, we believe that it is unlikely that that the qualification status of the TCVs was not discussed at these meetings.

Therefore, based on the substantial amount of information provided during the licensing process it is reasonable to conclude that the AEC reviewer(s) recognized that the TCVs were non-safety related and determined that it was acceptable to credit operation of the valves in this scenario. This conclusion may have been based on the relatively low significance of these valves failing to close. The specific concern was that the AEC considered it unacceptable to credit operator action in the "short time intervals following an accident". The evaluations in the subject correspondence showed that adequate flow would be provided to the Emergency Diesel Generators (EDGs) even with the TCVs full open and no operator action in the short time interval following the accident.

II. Significance

As discussed above, it is within the plant's licensing basis to credit the TCVs with reducing the demand on the CL System in this event. The following items provide additional assurance for proper operation of the TCVs and a discussion of the margin built into the system.

(1) The reliability of the TCVs to perform this credited function.
(2) The plant response to a failure of the TCVs to close during a LOOP.
(3) The risk significance of this sequence of events including failure of the TCVs to close.

Each of these is addressed in more detail below.

1. TCV Reliability

The TCVs require an air supply to close; i.e., the valves fail open on a loss of air. The air supply comes from the Instrument Air System. The Instrument Air Compressors are non-safety related; however, the air compressors are automatically loaded on the EDGs and would be available during a LOOP. The electrical supply to the control valve actuator is from an electrical panel that is powered from a non-safeguards non-interruptible power supply that is battery backed. The temperature element (that controls valve position) is powered from a safeguards instrument panel that would receive power from a safeguards battery. Diagnostic testing is performed on the TCVs every refueling outage. As necessary, any problems are identified and corrected using the Work Control process. A search was conducted to determine the reliability of the subject valves. This search reveals that the valves have been very reliable, with only one instance of a valve failing to completely close (in 1979). This reliability is consistent with the importance of the valve functioning to support normal plant operations. Therefore, although the TCVs are non-safety related; there is reasonable assurance that the valves will close in this event.

2. Plant and Operator Response

The specific event scenario addressed is as follows (the only difference between this and that identified in a letter from the AEC to NSP, dated

February 27, 1974, is that credit is not taken for the TCVs closing in 10 minutes):

- Prior to the LOOP, it is assumed that two safeguards CL Pumps are operable. This is the minimum Technical Specification Limiting Condition of Operation (LCO).
- The LOOP is assumed to occur with no other coincident accident. That is, there is no safety injection (SI) signal to split the CL headers (with the SI signal the headers split and this issue is not a concern).
- Both Units trip due to the LOOP.
- One of the safeguards CL Pumps fails to start (single active failure), resulting in one CL Pump supplying both CL headers.

Hydraulic evaluations indicate that adequate flow would be provided to the EDGs and that the CL pump could be operating in a limited operating region of the pump curve; pump manufacturer recommended operation in this region no more than 1 to 2 hours. This is very similar to the results of the evaluation documented in the 1974 correspondence.

With the reactor trip, the control room operators would enter Emergency Operating Procedure (EOP) E-0 (Reactor Trip). The first four steps of E-0 are performed from memory; verify reactor and turbine trip, verify safeguards electrical buses are energized and check if SI is actuated. After determining that SI is not actuated, the operators would transition to EOP ES-0.1 (Reactor Trip Recovery).

Early in ES-0.1, the operators verify Cooling Water Header pressure is greater than 75 psig. In this case, the pressure would not be greater than 75 psig and the operators would be directed to the procedure for restoring Cooling Water pressure (C35 AOP2). In addition, there is an alarm in the control room for the low CL header pressure (setpoint of 75 psig) that will also direct the operators to C35 AOP2 to restore CL system pressure.

C35 AOP2 reduces the total demand within the continuous operating region for the pump. The valve manipulations performed to reduce the flow demand are all performed from the Control Room (no out-plant actions are required). Personnel outside the control room are directed to help in the investigation of the cause of the high flow demand; however, the flow demand can be reduced independent of these investigations.

Operators are trained on these procedures and the actions are practiced using the plant simulator. Based on the relatively few number of steps required, the relative ease of performing these steps, and that a LOOP is a condition II event (which is less stressful than other Design Basis Accidents) 30 minutes is a conservative estimate of the time required to complete the actions necessary to reduce the demand on the system to within the pump continuous operating region. When similar scenarios have been used on the simulator the operators were able to perform the necessary actions well within 30 minutes. Thus, there is substantial margin between the time required for the operators to reduce the flow demand (less than 30 minutes) and the time that the pump can operate in this limited operating region (1 to 2 hours). This time frame for operator action is considered as recovery actions and outside the concern for crediting operator actions in the "short time intervals" following an accident.

3. Risk Significance

A determination was made of the annual frequency of occurrence of conditions that would lead to this requirement for one remaining safeguards CL pump to supply the CL system following a LOOP initiating event for greater than one hour. Note that this is determining a frequency of occurrence and not the same as the core damage frequency typically used as a measure of risk in Probabilistic Risk Assessment (PRS) analyses. Other failures would be required in these sequences in order to produce a core damage event.

Failures modeled include those leading to single pump operation, random and common-mode failures of the Turbine Hydrogen Cooler TCVs (including coincident loss of Instrument Air), failure of off site power recovery within one hour and failure of operator action to reduce CL system load within one hour. From this determination, the overall event frequency is approximately 1E-8/year. This demonstrates that the likelihood of occurrence of the sequence of events leading to the loss of CL is very low and, furthermore, the expected Core Damage Frequency (CDF) related to this scenario would be even lower. The total core damage frequency from the Individual Plant Evaluation (IPE) and from other recent PRA analyses are greater than 1E-5/yr. Also, again for comparison, the most widely accepted (through the NRC Significance Determination Process (SDP) and other measures) general delta-CDF threshold, above which an issue is considered to have some risk significance, is 1E-6/yr. The calculated frequency for this sequence of events is nearly 2 orders of magnitude below that threshold.

III. Overall Conclusions

In conclusion, based on the above discussion, the plant is operating within the licensing basis. Furthermore, our evaluation shows that failures beyond the licensing basis (failure of the TCVs in this case) can be tolerated. In addition, such failures are not significant from a risk assessment perspective.