ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR-46
Report No.:	50-298/96-30
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	November 4 through December 4, 1996
Inspector:	Terrence Reis, Sr. Project Engineer
Approved By:	Elmo Collins, Chief, Reactor Projects Branch C

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Cooper Nuclear Station
NRC Inspection Report 50-298/96-30

Operations

- Until October 1995, the licensee operated the facility inconsistent with the manner described in the Updated Safety Analysis Report in that residual heat removal (RHR) was not required to be available to service the spent fuel pool heat loading when performing full core offloads.

Engineering

- The licensee apparently failed to update the Updated Final Safety Analysis following a safety evaluation supporting a 1977 license amendment. Specifically, the license did not reflect the revised heat loading of the spent fuel pool nor licensing basis regarding maximum heat loading into Section X.5 of the USAR. This item, designated a unresolved in NRC Inspection Report 50-298/96-07 remains unresolved.

- The licensee failed to translate design basis information regarding heat loading of the spent fuel pool into operational procedures. Specifically, design assumptions and requirements indicated that an RHR intertie to the spent fuel pool was required to service the heat load associated with a full core offload and that the maximum fuel pool heat load would be associated with a full core discharged to the spent fuel pool 13 days after shutdown. The licensee did not procedurally control the availability of the RHR system and did not restrict the rate of core offload.

- The licensee identified that the reactor core isolation cooling system was not independent of ac power which caused it to be in violation of the station blackout rule. In assessing the safety significance of the reactor core isolation cooling design error, the licensee incorrectly credited high pressure coolant injection system operation for the coping period.

Details

I. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Reactor Core Isolation Cooling (RCIC) System Deficiency

a. Inspection Scope

The scope of the inspection was to assess the significance of the design deficiency described in Licensee Event Report (LER) 94-018 and its supplements and to determine if the corrective actions specified in the LER were sufficient.

b. Observations and Findings

In August 1994, the licensee identified during an extended shutdown that Valve RCIC-MOV-MO14, RCIC turbine trip/throttle valve, was inappropriately powered from an alternating current (ac) source. The licensee's compliance with the station blackout (SBO) rule, 10 CFR 50.63, "Loss of All Alternating Current Power," takes credit for automatic operation of Valve RCIC-MOV-MO14 in an SBO scenario.

As part of the agency's implementation of the NRC Manual Chapter 0350, "Staff Guidelines for Restart Approval," process, this design deficiency was identified as a restart item. In NRC Inspection Report 50-298/94031, it was documented that the deficiency had been satisfactorily corrected prior to restart.

In the original design of the plant, RCIC was assumed to initiate upon a low vessel level and automatically shutdown upon a high vessel level. No credit was taken for reinitiation of RCIC following the initial actuation and shutdown cycle. Accordingly, the turbine trip/throttle valve was designed as a manual valve which was normally open and mechanically tripped closed.

The valve was held open by a mechanical latching mechanism and closed by spring force when the latch was disengaged. The valve was equipped with an electromechanical (dc solenoid powered) trip device that would unlatch or trip the valve closed when specified physical parameters were exceeded. These included reactor vessel high water level, low suction pressure, high turbine exhaust, low turbine oil, and overspeed. A manual trip was also provided. As such, the functioning of the valve and the RCIC system were considered to be independent of ac power.

In August 1977, Minor Design Change 74-120 was implemented, which installed a motor operator on the turbine trip/throttle valve and a control switch in the control room. This motor operator was ac powered and was provided so that an operator did not have to be dispatched to the RCIC room to locally reset the valve. The motor operator was considered an enhancement and a convenience and was not considered as a requirement to meet its design function. Therefore, even though the motor operator was ac powered, RCIC was considered to be still capable of performing its safety function as described in

the Updated Safety Analysis Report (USAR). This description only required RCIC to be initiated and be capable of short-term (one cycle) operation independent of ac power.

In 1981, NUREG 0737, Item II.K.3.13, required that the RCIC system be modified such that the system would automatically restart upon a low reactor vessel water level signal after one cycle. To comply with this requirement the licensee modified the existing AC-powered motor operator for Valve RCIC-MOV-MO14 to automatically reset the turbine trip/throttle valve. Motor Operator MO14 would cycle closed, latch, and open. Minor Design Change 81-003 was implemented to accomplish this change. After implementation of the NUREG 0737 requirement, RCIC's safety function was not complete after one cycle and RCIC was now dependent on AC power to remotely reset Valve RCIC-MOV-MO14.

In 1991, the licensee responded to the SBO rule. As part of the coping requirements for an SBO, a system capable of operating without ac power was needed to maintain vessel water level. The licensee designated the RCIC system as being capable of fulfilling this function. This was done largely because the cognizant manager responsible for the licensee's response to the SBO rule failed to adequately research the design basis and configuration of the RCIC system and relied upon out-of-date information that indicated the RCIC system was ac independent.

Therefore, as a result of inadequate design and document control changes, the licensee relied upon incorrect information to respond to an NRC-imposed rule. The RCIC system was not ac independent and accordingly was incapable of performing its intended safety function in an SBO scenario. This deficiency existed from the time the licensee was required to comply with the SBO rule in 1991 until it was corrected prior to restart of the facility in March 1995. The dependence of the RCIC system on AC power is a violation of 10 CFR 50.63.

In reporting this deficiency to the NRC as required by 10 CFR 50.73, the licensee indicated the safety significance of the event was minimal since both the high pressure core injection (HPCI) system and local manual reset capability of RCIC would still be available for vessel level control during the SBO event. However, as a result of an issue identified by the NRC inspector, it was determined that the HPCI system would not necessarily be available for the duration of the coping period.

On May 23, 1995, the NRC inspector identified that plant procedural requirements were inconsistent with assumptions and commitments made to the NRC in a letter regarding SBO coping strategy dated September 30, 1991. The NRC had requested that the licensee perform a heatup calculation of the HPCI room to assess equipment operability. The licensee opted not to expend resources on such a calculation and instead committed to secure HPCI operation after one cycle of operation. This option was considered acceptable by the NRC.

This regulatory commitment was found not to be properly translated into procedures or instructions. The action taken as a result of this submittal was revision to CNS Emergency Procedure 5.2.5.1, "Loss of Off-Site AC Power," dated February 23, 1992. The actual

procedure direction that was implemented required that operators reduce dc loads to a minimum by shutting down the HPCI system if the RCIC system was maintaining reactor pressure and level. No clarification was provided restricting HPCI operation to one cycle.

The licensee relied on the information as stated in Procedure 5.2.5.1 in making the statement in LER 94-018, Revision 1, that the safety significance of the RCIC system not being AC independent was minimal. It indicated that HPCI would be available. However, HPCI was not analyzed to operate for longer than one cycle (approximately 10 minutes) in an SBO event and therefore could not be credited for the entire coping period.

The licensee failed to properly translate a regulatory commitment to secure HPCI operation after one cycle in an SBO scenario into station procedures. A Severity Level IV was cited in NRC Inspection Report 50-298/95-08 for Procedure 5.2.5.1 inadequacy.

10 CFR 50.63, Loss of all alternating current power, requires that each nuclear power plant must be able to withstand for a specified duration and recover from an SBO as defined in 10 CFR 50.2. It further states that the reactor core and associated coolant, control, and protection systems must provide sufficient capacity and capability to ensure that the core is cooled.

The dependence of the RCIC system on AC power for long-term operation following an SBO constitutes a violation of 10 CFR 50.63 (298/96030-01).

In addition to correcting the evaluation of safety significance in the LER, the licensee:

- Conducted a review of the RCIC system to ensure that other components required for RCIC to meet its safety function and ac independence did not exist.
- Developed an internal procedure, "Regulatory Correspondence Control," to ensure that the information transmitted to the NRC is complete and accurate.
- Held tailgate sessions with engineering personnel to stress the importance of validating information.

These corrective actions were discussed in a public meeting on May 4, 1995, and were deemed appropriate.

c. Conclusions

The licensee identified that the RCIC system was not independent of AC power which caused it to be in violation of the SBO rule.

In addressing the safety significance of its failure to comply with the SBO rule, the licensee improperly credited the HPCI system as being available for SBO coping.

The licensee's corrective actions were appropriate.

E2.2 Performance of Full Core Offloads During Refueling Outages

a. Inspection Scope

The scope of the inspection was to determine if the licensee's practice of routinely performing full core offloads to the spent fuel pool was consistent with its licensing basis and regulatory requirements.

b. Observations and Findings

In NRC Inspection Report 50-298/96-007, Section 7.3, the staff documented that it considered the licensee's practice of performing full core offloads during refueling outages on a routine basis to be inconsistent with the licensing basis of the plant that existed on October 19, 1995, when the NRC first questioned this practice. The performance of routine full core offloads during past refuelings outages without performing 10 CFR 50.59 analyses and the failure to update the USAR following a licensing amendment were left as unresolved in the inspection report.

The observations, findings, and conclusions from NRC Inspection Report 50-298/96-07, Section 7.3, are restated below:

> During a recent evaluation of spent fuel pool decay heat removal and refueling practices, the NRC staff reviewed licensing basis documents for Cooper Nuclear Station. The documents included the USAR and documents associated with Amendment 52 to the Cooper license dated September 29, 1978 (rerack amendment). In these documents the inspector found that the routine core offload for these plants was described as being a partial offload.
>
> In USAR Section 5.5, the licensee describes two spent fuel assembly offloads, normal and emergency. The USAR states:
>
> > Case 1 Normal Heat Load
> >
> > The normal heat load case consists of one freshly discharged batch (approximately 160 bundles) in addition to the fuel (approximately 160 bundles) from each of the previous refueling outages.
> >
> > Case 2 Emergency Heat Load
> >
> > The emergency heat load case which produces maximum heat load results from the unscheduled discharge of the entire core just prior to a scheduled refueling outage. The freshly discharged fuel contributes the major portion of the heat load.

> This makes the total heat load quite insensitive to the number of assemblies cooled for more than a few years in the pool.

The maximum normal heat load is listed in USAR Table X-5-1 as 6.38 x 10^6 Btu/hr with a fuel pool temperature of 125°F, and the emergency heat load is listed as 16.6 x 10^6 Btu/hr with a fuel pool temperature of 150°F. This table lists the heat exchanger heat removal capacity as 3.19 x 10^6 Btu/hr for each of the exchangers.

Section 5.6 of the USAR describes use of the RHR system to remove heat from the spent fuel pool. Specifically it states:

> The maximum possible heat load is the decay heat at the full core load of fuel at the end of the fuel cycle plus the remaining decay heat of the spent fuel discharged at previous refuelings. The residual heat removal system is operated in parallel with the fuel pool cooling and demineralizer system to remove this heat load . . .
>
> If it appears that the pool water temperature will exceed 150°F, the fuel pool cooling and demineralizer system can be connected by operator action to the residual heat removal system. This increases the cooling capacity of the fuel pool cooling and demineralizer system so that a water temperature below 150°F is maintained.

In the rerack amendment request dated July 22, 1977, the licensee also describes the normal and emergency heat loads. The normal heat load is associated with a partial fuel offload. The emergency heat load is associated with an "unscheduled discharge of the entire core just prior to a scheduled refueling outage," as stated in Section 3.3 of the submittal. Based on the reracked conditions, the heat loads would increase to 7.7 x 10^6 Btu/hr for the normal offload, with an assumed cooling time of 7 days after shutdown, and to 19.8 x 10^6 Btu/hr for the emergency offload, with a cooling time of 13 days. The licensee also states in the application that:

> For the maximum emergency case, . . . it is expected that the RHR system will be capable of handling the expanded maximum heat generation of 19.8 x 10^6 Btu/hr by either increasing the flow rate or by allowing the pool temperature to increase to not more that 160°F (the current maximum is 150°F) . . . .
>
> In both cases, the anticipated maximum heat loads can be accommodated without any modifications to the present pool cooling system, with minor changes in design values of the

> maximum allowable pool temperature (no more than 10°F increase) for relatively short periods (maximum of 8 days).

The inspectors observed that the USAR, through revisions dated July 22, 1994, did not reflect the revised heat loads described in the rerack amendment application. An October 1995 change to the USAR, performed in response to NRC concerns, describes the heat removal capacity of the spent fuel pool cooling and RHR fuel pool in a manner that is consistent with the information included in the rerack application.

In the NRC staff's safety evaluation for the rerack amendment, the staff stated that it had focused on the higher fuel pool heat loads that would be generated by the larger number of fuel to be stored. The NRC staff commented in the safety evaluation about offload practices:

> ". . . NPPD [Nebraska Public Power District, the licensee] states that the maximum possible heat load in the spent fuel pool due to annual refueling will be 7.7×10^6 BTU/hr and that the heat load due to a full core offload will be 19.8×10^6 BTU/hr."

Further, the NRC staff calculated higher estimated heat loads for the offload cases based on the additional heat generated by the successive offloads along with higher potential fuel pool temperatures than that calculated by the licensee, specifically, 9.1×10^6 BTU/hr and 138°F for the normal offload and 21.9×10^6 BTU/hr for the full core offload with temperature still maintained at 150°F based on the use of the residual heat removal system.

The inspector noted that the definition of normal versus emergency offloads was clear in the USAR and, further, the use of the term "unscheduled' in the USAR supports the conclusion that partial offloads were to be the normal or routine refueling outage practice. The NRC staff considers that the practice of routinely conducting full core offloads and introducing an off-normal heat load into the spent fuel pool to be a change to the normal practice defined in the Cooper Nuclear Station licensing basis. Such a change to the operation of the facility should have been reviewed pursuant to the requirements of 10 CFR 50.59, including clear discussion of the systems relied on to remove the spent fuel pool decay heat for the duration of the outage, prior to implementing the practice of routinely offloading the full core.

The inspector noted that no 10 CFR 50.59 safety evaluation was done regarding this practice of performing full core offloads on a routine basis until October of 1995. In October 1995, after significant discussion with the NRC staffs, the licensee revised the USAR to reflect temperature limits and heat load during refueling and the systems that are relied on to maintain the spent fuel pool bulk temperature within those limits.

> The performance of routine full core offloads during past refueling outages without performing appropriate 10 CFR 50.59 reviews and the untimely updating of the USAR to reflect information submitted to support the 1977 rerack application is an unresolved item (Update of 298/9514-01).

Since NRC Inspection 50-298/96-07 was completed, the NRC has revised NUREG-1600, "NRC Enforcement Policy." This revision provides guidance on the disposition of concerns arising from departures from the USAR. The guidance was promulgated in Enforcement Guidance Memorandum (EGM) 96-005 dated October 21, 1996.

Based on this guidance, the NRC has <u>not</u> found that performing full core offloads on a preplanned and routine basis constitutes a "defacto change" to the facility as described in the USAR and has <u>not</u> found that a 10 CFR 50.59 violation occurred due to the licensee's practice of performing full core offloads. However, the NRC has determined that violations of NRC requirements did occur in that:

- The licensee performed full core offloads without ensuring RHR remained available to service the heat load. This does represent a "defacto change" to the facility for which a 10 CFR 50.59 evaluation was not performed.

- The licensee failed to translate design basis information regarding heat loading into operational procedures. Specifically, there were no procedural controls in place to ensure the full core offload was not completed prior to a 13-day cooling period.

As stated, the licensee demonstrated in October 1995 that the practice was not inconsistent with its licensing basis and did not constitute an unreviewed safety question. However, in performing their investigation, the licensee did identify that there were limitations on the evolution that were not adequately controlled in the licensee's programs and procedures.

The current licensing basis for the heat load capacity of the spent fuel pool was documented in the NRC staff's Safety Evaluation Report (SER) associated with Amendment 52 to the operating license. In the licensing submittal, the licensee indicated that the heat load associated with the full core offload assumed a 13-day (312 hour) interval between a reactor shutdown and the time a full core offload was completed. The staff's SER documented this time interval as part of the licensing basis. The licensee had no administrative controls in place to ensure that the offload rate was not exceeded. During Refueling Outage 16, which began in October 1995, the offload rate was exceeded.

Additionally, the staff's SER documented that the licensee would use the RHR system for cooling the spent fuel pool when a full core offload was performed. The purpose of this licensing basis requirement was to ensure the temperature of the spent fuel pool did not exceed the design temperature of 150 F. In October 1995, while in the process of performing a full core offload, RHR Train B, which is the only train that can physically service the fuel pool, was not available. However, the situation was corrected prior to the licensee discharging more than 1/3 of the core. The SER documents that the normal spent

fuel pool cooling system is capable of handling this heat load. It is not known if RHR Train B was available to serve the spent fuel pool during previous outages, but the licensee acknowledges that it had no outage plans or administrative procedures to ensure RHR Train B remained available.

10 CFR 50.71(e) requires that each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report to assure that the information included in the final safety analysis report contains the latest material developed. The updated final safety analysis report (USAR) shall be revised to include the effects of all changes made in the facility or procedures as described in the final safety analysis report (FSAR) and all safety evaluations performed by the licensee in support of requested license amendments.

In its July 22, 1977, licensing submittal supporting License Amendment 52, the licensee designated the maximum possible heat load in the spent fuel pool due to an annual refueling as 7.7E6 BTU/hr and the heat load associated with a full core offload as 19.8E6 BTU/hr.

In its July 22, 1977, licensing submittal supporting License Amendment 52, the licensee designated a 13-day cooling time interval between a reactor shutdown and the time a full core offload is completed.

Nebraska Public Power District, the licensee for Cooper Nuclear Station, failed to revise the USAR to include the effects of:

1. A safety evaluation performed by the licensee dated July 22, 1977, in support of License Amendment 52 in that on October 20, 1995, USAR Table X-5-1 designated the maximum normal heat load of the spent fuel pool as 6.38E6 BTU/hr and emergency heat load (full core offload) as 16.6E6 BTU/hr.

2. A safety evaluation performed by the licensee dated July 22, 1977, in support of License Amendment 52 in that on October 20, 1995, Section 5 of the USAR did not address that maximum heat loads were associated with a 13-day cooling time between a reactor shutdown and the time a full core offload is completed.

This aspect of Unresolved Item 298/95014-01 to update the USAR to reflect Licensee Amendment 52 remains open.

10 CFR 50.59(a)(1) requires that a licensee may make changes in the facility as described in the safety analysis report without prior Commission Approval unless the change involves a change in the Technical Specifications incorporate in the license or an unreviewed safety question. 10 CFR 50.59(b)(1) states, in part, that the licensee shall maintain records of changes in the facility, to the extent that these changes constitute changes in the facility as described in the safety analysis, and that these records must include a written safety evaluation which provides the basis for the determination that the change did not involve an unreviewed safety question.

On October 20, 1995, the licensee's FSAR, Section 8.5.6, stated, in part, that "the residual heat removal (RHR) system can be intertied with the Fuel Pool cooling system if required. This capability increases the spent fuel pool cooling capacity in the event that such additional capacity is necessitated by removal from the core of an unusually large number of fuel elements. The RHR system - fuel pool cooling system intertie is sized to remove an emergency heat load . . . from the fuel pool which corresponds to full core off-loading plus the batch of spent fuel discharged at the previous refueling outage.

In the NRC's safety evaluation supporting License Amendment 52 dated September 29, 1978, it was stated in Section 2.2 that the licensee would use RHR cooling when performing full core offloads.

On October 20, 1995, the licensee changed the facility as described in the safety analysis report in that the facility was not operated as described in the FSAR and a written safety evaluation of the change from the FSAR had not been performed to determine whether this change involved an unreviewed safety question. Specifically, the licensee was in the process of performing a full core offload, and the RHR system was not available to assist the fuel pool cooling system in removing what the FSAR characterized as an emergency offload. This is a violation of 10 CFR 50.59 (298/96030-02).

Criterion III of Appendix B to 10 CFR Part 50 requires that regulatory requirements and the design basis, as defined in 10 CFR 50.2 and as specified in the license application, for those structures systems and components to which the appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

In the safety evaluation report which accompanied Amendment 52 to the facility operating license, the NRC staff acknowledged that the licensee's spent fuel pool and cooling systems were capable of handling the heat load associated with a full core discharge. However, this acknowledgement was based on certain design assumptions. In the Safety Evaluation Report, the staff stated that the maximum fuel pool heatload was associated with an offload that would occur 13 days after shutdown.

The design basis assumption that the maximum heat load was associated with full core discharge which was completed in 13 days was not translated into procedures. Procedure 2.3.2, "Fuel Pool Cooling and Demineralizer System," contained no administrative controls to ensure that fuel was not loaded at a rate that would exceed the 13-day assumption. In October 1995, the licensee did exceed this offload rate before the full core offload was halted at 1/3 of the core. This is a violation (298/96030-03).

The two violations (298/96030-02,03) disposition the spent fuel pool operation aspects of Unresolved Item 298/95014-01 (Update).

c. Conclusions

The licensee apparently failed to update the USAR, following a safety evaluation supporting a 1977 license amendment.

The licensee operated the facility inconsistent with the manner described in the USAR in that RHR was not required to be available to service the spent fuel pool heat loading when performing full core offloads.

The licensee failed to translate design basis information regarding heat loading of the spent fuel pool into operational procedures. Specifically, design assumptions and requirements indicated that an RHR intertie to the spent fuel pool was required to service the heat load associated with a full core offload and that the maximum fuel pool heat load would be associated with a full core discharged to the spent fuel pool 13 days after shutdown. The licensee did not procedurally control the availability of the RHR system and did not restrict the rate of core offload.

II. Management Meetings

XI Exit Meeting Summary

The inspector met with licensee management on November 7, 1996, and presented the findings. At that time, one of the violations involving operation of the spent fuel pool was characterized as a "defacto change" to the facility for performing full core offloads when wording in the USAR indicated that the performance of a full core offload was an emergency condition. The licensee objected to that characterization.

On December 4 and 20, 1996, the inspector formally exited with the Manager, Nuclear Safety and Licensing. At this time, the issue discussed above was characterized as presented in this report, which is an unresolved item.

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Brad Houston, Nuclear Safety and Licensing Manager
Robert Godley, Manager, Plant Engineering
Mike Peckham, Plant Manager
Rick Gardner, Operations Manager
Jack Dillich, Maintenance Manager

INSPECTION PROCEDURES USED

IP 92903: Followup - Engineering

ITEMS CLOSED

298/94-018	LER	RCIC System Inoperable in Station Blackout Scenario
298/96030-01	VIO	Failure to Comply with Station Blackout Rule
298/95-013	LER	Plant Procedural Requirements Inconsistent with Station Blackout Assumptions

ITEMS OPENED AND UPDATED

298/96030-01	VIO	Failure to Comply with Station Blackout Rule
298/96030-02	VIO	Failure to Comply with 50.59 in that RHR System not Available
298/96030-03	VIO	Failure to Translate Design Basis information into operational Procedures
298/95014-01	URI	(UPDATE) The aspects of this URI regarding the operation of the spent fuel pool are closed. The apparent failure to update the USAR remains unresolved.