Nebraska Public Power District

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NLS2004128
October 12, 2004

Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Subject: Regulatory Conference - Request for Additional Information
Cooper Nuclear Station, Docket 50-298, DPR-46

Reference: Nuclear Regulatory Commission (NRC) Inspection Report 05000298/2004014, dated August 12, 2004

The purpose of this letter is for the Nebraska Public Power District (NPPD) to answer questions that resulted from a Regulatory Conference held with the NRC on September 27, 2004. The questions were posed after the caucus, and NPPD understands that the question responses will be used to help the NRC reach its final significance determination. This letter constitutes the formal submittal of the answers with only minor changes from the original responses. The answers are contained in the attachment and enclosures to this letter.

During the Regulatory Conference, NPPD presented information consistent with previously docketed information in the referenced inspection report, identified differences related to inputs and assumptions used by the NRC in reaching its preliminary significance determination, and provided new information developed from a full-scale confirmatory pump test. NPPD believes all of the information supports the position that the performance deficiency documented in the referenced inspection report should be characterized as having very low safety significance.

The questions were generally focused in the two areas of (1) predicted Service Water (SW) pump performance given the test results compared with in-situ conditions, and (2) station battery capability to support continued operation of the Reactor Core Isolation Cooling (RCIC) pump beyond what was assumed for compliance with the station blackout rule. With respect to SW pump performance, the colder river water temperatures experienced during the period that gland water was cross-connected, coupled with the inherent limitations on shaft heating due to bearing failure (which further limits shaft elongation) results in a significant margin to reaching a motor over-current trip condition. With respect to extended RCIC pump operation, NPPD determined that the Division 1 battery alone has sufficient capability to support continued RCIC operation beyond 8 hours. However, and as described in the answer regarding operations guidance for transferring the RCIC power source, the operators could shift the direct current power source from Division 1 batteries to Division 2 batteries within the first 90 minutes. If that were the case, then later in the event, operations would transfer RCIC back to the Division 1 battery source.

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Should you have any questions or require additional information, please contact me at 402-825-2774.

Sincerely,

Paul V. Fleming
Licensing Manager
Cooper Nuclear Station

/jrs

Attachment
Enclosures

U.S. Nuclear Regulatory Commission w/attachment and enclosures
Attn: Document Control Desk

Senior Project Manager w/attachment and enclosures
USNRC - NRR Project Directorate IV-1

Kriss Kennedy w/attachment and enclosures
Chief, Branch C
U.S. Nuclear Regulatory Commission
Region IV

Senior Resident Inspector w/attachment and enclosures
USNRC

NPG Distribution w/attachment and enclosures

Records w/attachment and enclosures

1. NRC Request

The most recent B and D Service Water (SW) Pump Vibration Data, including alert and action levels, prior to the condition that existed when the SW valves were mispositioned.

NPPD Response

The most recent vibration data for the B and D SW pumps prior to the condition was from the 1/5/04 performance of 6.2SW.101, "Service Water Surveillance Operation". The following is the data obtained:

Note: All readings are in inches per second.

	Recorded Value	Alert Level	Action Level
B SW Pump			
Location 1V	0.079	0.325	0.700
2H	0.116	0.325	0.700
4H	0.100	0.312	0.700
D SW Pump			
Location 1V	0.115	0.270	0.648
2H	0.105	0.275	0.660
4H	0.049	0.170	0.408

Locations can be seen in Figure 1 on Page 2 of this Attachment.

Basis:

This data is taken during quarterly Inservice Testing surveillance testing using temporary instrumentation consisting of a magnetic accelerometer and handheld data acquisition system. The SW pumps do not have permanently installed vibration monitoring capability.

The vibration data is taken to monitor equipment performance. By trending this data, adverse trends can be detected early and equipment performance issues can be proactively addressed.

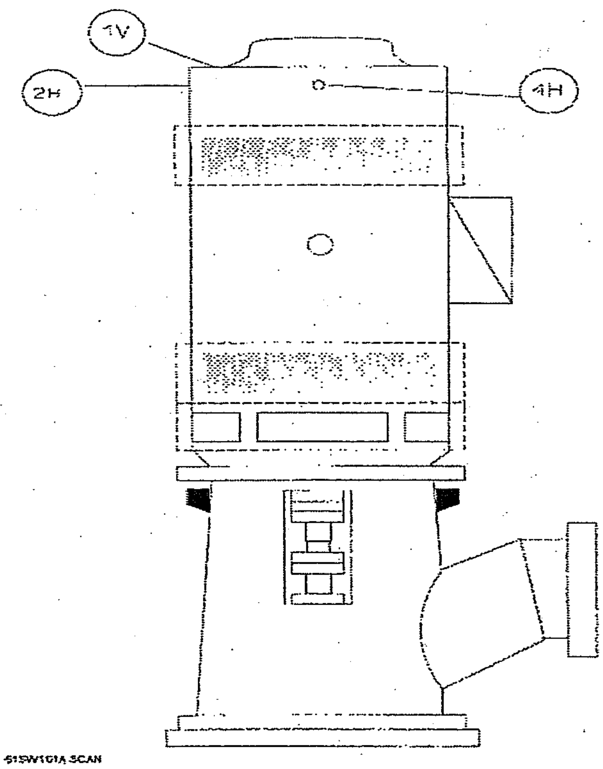

Figure 1

2. NRC Request

What is the difference in shaft expansion for the test conditions compared to the cooler water that existed during the 21 day period?

NPPD Response

The thermal expansion of the shaft during the test conditions and during the 21 day plant conditions will be approximately the same, with the plant case expansion slightly less than the test case. The expansion is nearly the same because the two main parameters that change from one case to the other (water temperature and starting pump shaft temperature) approximately cancel each other out when comparing the two cases.

Basis:

Hopper Elmore and Associates performed an analysis using an ANSYS thermal finite element model and a transient thermal analysis to calculate the differences between thermal expansion of the shaft at the test conditions and the plant conditions. The summary of this analysis is contained in the document entitled "Service Water Pump Thermal Growth Calculation Analysis Summary", which is provided in Enclosure 1. This summary document is further supported by a detailed model/calculation that is under final review at the offices of Hopper Elmore and Associates.

Hopper Elmore and Associates

Service Water Pump Thermal Growth Calculation

HOPPER ELMORE AND ASSOCIATES
ENGINEERS

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FAX: (310) 791-7308
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October 7, 2004
HEACNS-10/04-045

VIA ELECTRONIC MAIL AND U. S. MAIL

Mr. Dwight Vorpahl
Nebraska Public Power District
Cooper Nuclear Station
P. O. Box 98
Brownville, NE 68321-0098

Subject: Service Water Pump Shaft
Thermal Growth Evaluation
Results Summary Transmittal

References: Vorpahl, Elmore Technical Conversation on the Subject, October 7, 2004.

Dear Mr. Vorpahl:

In accordance with the reference we have prepared and enclose the subject results summary for your records. Our calculation package is being prepared and will be finalized and submitted next week.

It has been our pleasure assisting you with this interesting project. If you have any questions please do not hesitate to contact Mr. Mark Gielow or myself.

Very truly yours,

Kelley S. Elmore
Professional Engineer

cc: Mr. Todd Hottovy

HOPPER ELMORE AND ASSOCIATES
ENGINEERS

Service Water Pump Thermal Growth Calculation
Analysis Summary

Purpose: Calculate SW pump shaft thermal expansion due to heat input at the dry shaft bearings for the test operating conditions and for plant operating conditions.

Analysis Methodology: An ANSYS thermal finite element model of a portion of the pump shaft length, bearing, and outer enveloping tube is constructed. The model is axisymmetric and includes approximately one span length between bearings. A heat generation load is applied at the interface between the bearing and the shaft. An initial heat generation rate is calculated based on the pump RPM, bearing material coefficient of friction, and assumed bearing operating pressure load. A convection cooling load is applied to the outer surface of the enveloping tube to model the cooling effect of the flowing water. A transient thermal analysis is performed to calculate the shaft temperature as a function of time. The shaft temperatures are then input to a structural model of the shaft which calculates the thermal expansion. To obtain the total expansion of the full shaft length, the expansion for one bearing span length calculated by the model is multiplied by five. The heat generation rate is adjusted to match the results of the pump test, i.e., to find the rate that causes closure of the gap between the impeller and pump bowl after approximately 37 minutes of dry operation. Using the adjusted heat generation rate, the model is reanalyzed for the plant operating conditions.

Assumptions:

1. Heat is generated at only the five upper shaft bearings; there is minimal heat generation at the other bearings based on the test results that show no damage at these bearings.
2. The nominal gap between the impeller and pump bowl is 0.08".
3. The pump shaft deflects downward 0.03" due to hydraulic downthrust on the impeller, reducing the gap to approximately 0.05".

Evaluation Conditions:

Test Case

Water temperature	=	63°F
Starting pump shaft/tube/casing temperature	=	63°F
Flow rate	=	6030 GPM

Plant Case

River water temperature	=	38°F
Pump bay temperature	=	43°F
Starting pump shaft/tube/casing temperature	=	43°F
Flow rate	=	6030 GPM

Analysis Results:

Test Case: Using the initial estimated heat generation rate at the shaft/bearing interface, the shaft was found to heat up and thermally expand 0.05" in approximately 15 minutes of dry operation. Therefore the impeller would contact the bowl in 15 minutes, whereas during the test, contact occurred after approximately 37 minutes. To calibrate the model with the test results, the heat generation rate was reduced to cause the shaft to expand 0.05" in approximately 37 minutes. The pump shaft expansion as a function of time is shown in Figure 1. Using the calibrated heat generation rate, the shaft axial expansion after 37 minutes was 0.050".

Plant Case: The model was re-analyzed for the extreme plant operating conditions (bay temperature of 43°F and water temperature of 38°F due to loss of de-icing flow into the river water). The starting temperature of the pump components and the water temperature were changed to the plant conditions. The calibrated bearing heat generation rate was used in the analysis. The results of the analysis are shown in Figure 1. The results in Figure 1 show that the shaft expansion for the test case and the plant case are very similar - the two curves lay almost on top of each other. The pump shaft thermal expansion after 37 minutes was 0.049", about 2% less than the expansion calculated for the test case.

Conclusion / Discussion of Results: The thermal expansion of the shaft during the test conditions and during the extreme plant conditions will be approximately the same, with the plant case expansion slightly less than the test case. The expansion is nearly the same because the two main parameters that change from one case to another (water temperature and starting temperature) approximately cancel each other out when comparing the two cases. This effect can be better understood by examining the heat transfer phenomena that occurs while operating the pump with dry bearings.

The heat generated by friction at the interface between the shaft and the dry bearings is dissipated in primarily two ways: 1) Conduction heat flow in the axial direction up and down the shaft away from the bearing; 2) Conduction heat flow in the radial direction through the bearing and enveloping tube and then into the pump water flow via convection at the outer surface of the tube. (Some additional heat is dissipated into the air gap around the dry length of shaft, but this heat loss is minimal and is conservatively neglected in the analysis.) The amount of heat generated by bearing friction is the same for both the test and the plant cases. Therefore, the only parameter that changes is the relative amount of heat that is dissipated in the two directions. The relative amount of heat that is dissipated in each direction is proportional to the temperature difference between the hot zone near the bearing and the temperature of the area the heat is flowing towards. For example, consider a point in time shortly after the start of dry pump operation when the bearing area has heated up by 50°F. For the test case, the bearing area is now at a temperature of 63 + 50 = 113°F. At this time, the rest of the shaft away from the bearings is still close to its start temperature of 63°F. The axial conductive heat flow into the shaft is proportional to

$$\Delta T_{axial} = T_{hot} - T_{shaft} = 113°F - 63°F = 50°F.$$

The radial heat flow through the bearing and tube and then via convection into the 63°F pump water flow is proportional to

$$\Delta T_{radial} = T_{hot} - T_{water} = 113°F - 63°F = 50°F.$$

The ratio $\Delta T_{axial}/\Delta T_{radial} = 50/50 = 1$. (Note: this is not meant to imply that the heat flow is equally divided in the two directions, it only indicates that for the purpose of comparison to the plant case, the driving ΔT ratio is equal to one.)

For the plant case, the bearing area is now at a temperature of 43 + 50 = 93°F. The axial conductive heat flow into the shaft is proportional to

$$\Delta T_{axial} = T_{hot} - T_{shaft} = 93°F - 43°F = 50°F$$

The radial heat flow through the bearing and tube and then via convection into the 38°F pump water flow is proportional to

$$\Delta T_{radial} = T_{hot} - T_{water} = 93°F - 38°F = 55°F.$$

The ratio $\Delta T_{axial}/\Delta T_{radial} = 50/55 = 0.91$.

Therefore the ratio of the two ΔT's for the test and the plant cases is approximately the same. This indicates that the relative heat flow, i.e., the percentage of total heat generated that flows axially into the shaft versus the percentage that flows radially outward and into the water flow, will be approximately the same for the test case and the plant case. The amount of shaft expansion is a function of the amount of heat that flows into the shaft; therefore if the heat input is the same, the thermal expansion will also be the same. Therefore the ANSYS model result is reasonable and it can be concluded that the shaft thermal expansion will be approximately the same for the test case and the plant case. The slightly lower expansion for the plant case is expected considering the ΔT ratios above. The lower ratio for the plant case indicates that a greater proportion of the heat flows radially outward than axially into the shaft than for the test case, which means less heating of the shaft and less shaft thermal expansion. Note that the shaft expansion would also be approximately the same as the test case for the plant condition with no loss of de-icing flow and a water temperature of 43°F instead of 38°F.

Shaft Expansion After Impeller Impact

The results in Figure 1 show the shaft thermal expansion up to the point where the impeller makes contact with the pump bowl. After that point, the expansion will follow a steeper curve as a function of time. This occurs because impeller contact causes compressive and lateral loading of the shaft, which results in increased shaft orbiting, much larger bearing loads, and greater heat generation. The increased bearing loads and heat generation caused the severe bearing damage observed on the test pump. Also during this period, impeller and bowl material are being removed. Finally an equilibrium point is reached after about 3.5 minutes, when bearing heat generation has peaked and no additional shaft expansion occurs. After approximately 7.5 minutes, stable pump operation resumes. This series of events, including the total shaft expansion and amount of material removed by the impeller scraping the bowl, will be essentially the same for the test case and for the plant case.

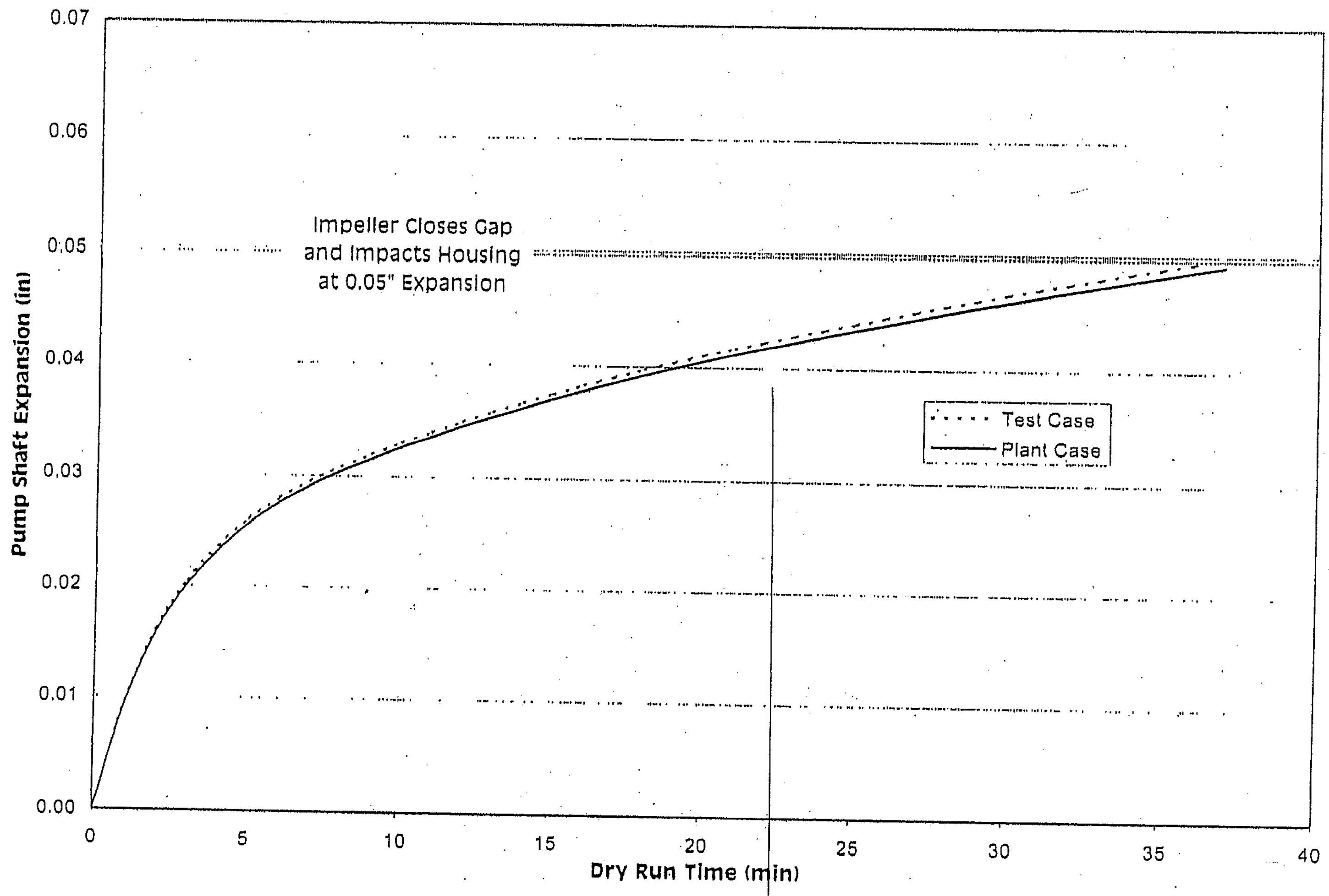

0.07
0.06
0.05
0.04
0.03
0.02
0.01
0.00
0
5
10
15
20
25
30
35
40
Pump Shaft Expansion (in)
Dry Run Time (min)
Impeller Closes Gap
and Impacts Housing
at 0.05" Expansion
Test Case
Plant Case

Figure 1

3. NRC Request

What were the highest and lowest river temperatures during the 21 day period?

NPPD Response

River temperatures are not taken directly at Cooper Nuclear Station. Only intake bay temperatures and plant component inlet temperatures are recorded at the site. However, river temperatures are recorded at Nebraska City, NE, by the U.S. Army Corps of Engineers, which is approximately 30 river miles upstream of Cooper Nuclear Station. The following temperatures were recorded during the period from 01/21/04 to 02/11/04.

River Temperature (deg F)

Date	Omaha	Nebraska City
1/21/2004	32.11	38.25
1/22/2004	32.50	37.85
1/23/2004	30.79	37.71
1/24/2004	32.58	37.71
~~1/25/2004~~	~~32.34~~	~~37.85~~
1/26/2004	32.94	37.72
1/27/2004	32.90	37.45
1/28/2004	32.70	37.08
1/29/2004	32.23	37.42
1/30/2004	32.58	37.18
1/31/2004	32.63	37.18
2/1/2004	32.49	36.82
2/2/2004	32.41	36.77
2/3/2004	32.29	36.88
2/4/2004	32.68	36.73
2/5/2004	32.85	36.82
2/6/2004	32.61	36.68
2/7/2004	32.99	36.64
2/8/2004	32.99	36.70
2/9/2004	33.17	36.48
2/10/2004	32.83	36.72
2/11/2004	32.88	36.81

Basis:

As noted above, the river temperature is obtained from instrumentation near Nebraska City, NE which is considered to be representative of the temperature of the river at Cooper Nuclear Station. However, because Cooper Nuclear Station is approximately 30 river miles downstream of Nebraska City, there is a slight heating effect as the water picks up some ground heat as it travels South during the winter months. Taking this added heat into account, it can be assumed that the river temperature at Cooper Nuclear Station ranged between 38 and 39 degrees F during the 21 day period.

4. NRC Request

How much more force would it take to cause a pump/motor breaker trip (overcurrent)?

NPPD Response

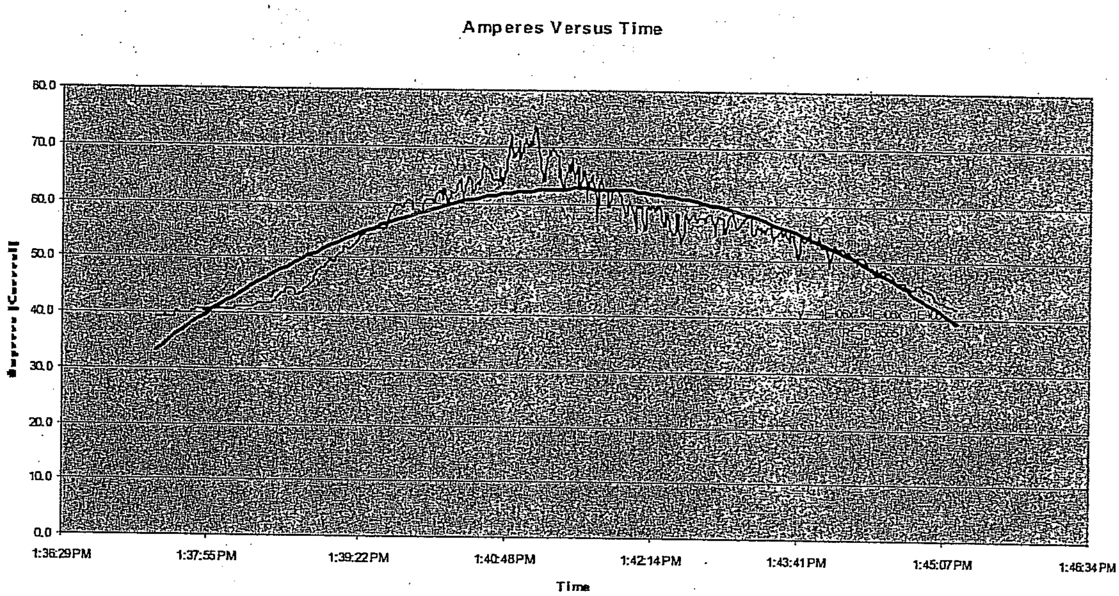
In order for the pump motor to trip on overcurrent, the force would have to nearly double (increase approximately 85%). However, as supported by the response to the question on shaft elongation, there is no mechanism present that would cause the heat-up rate of the shaft to increase.

Basis:

The scraping effect of the impeller upon the bowl is no different in principle than the process of machining. In that regard, the rate of material removal, usually measured in cubic inches per minute, is directly proportional to the power input at the motor that powers the machining process. If the rate of material removal is doubled, and all other factors about the process remain the same, the power needed to do this is also doubled.

The rate of shaft expansion in the period prior to contact between the impeller and bowl liner when the shaft was being heated by contact with the bearings was approximately 50 mils in 37 minutes, or an average expansion rate of about 1.35 mils per minute. However, due to the heat up process following an exponential curve, as defined by Newton's Law of Cooling, the rate of expansion will initially be higher, and then slows down as the temperature approaches it's peak.

During the impeller/bowl grinding period, which was about 7 minutes, slightly less than 38 mils of material was removed. This is a removal rate of approximately 5.4 mils per minute. The reason for the increased expansion rate during the impeller/bowl grinding period is due to the shaft being under compression. This results in the shaft orbiting, creating higher lateral bearing loads, which in turn causes more friction between the bearings and shaft, causing the shaft to heat up at a faster rate. However, once the temperature of the bearing approaches 300 degrees F, the rubber breaks down and portions peel away, thus limiting the mechanism (i.e., resistance between the shaft and bearing) for further increase in temperature. Once again, during this 7 minute period of time where grinding occurred, the heat up and cooldown process follows an exponential curve based on Newton's Law of Cooling. This is noted in the following graph, which has a parabolic curve fitted to the ampere data.



Thus, the power needed by the motor to sustain the machining rate between the impeller and bowl would not have been much more than that already recorded in the test. Note that the parabolic curve fit in the graph has a fit "R" value of 93%, which is a good correspondence.

When the pump was just pumping water, it used approximately 39 amperes. The maximum current used by the motor was about 73.6 amperes during grinding of the impeller into the bowl liner. Thus, approximately 39 amperes was being used to pump water, and about 34.6 amperes was being used to remove metal at a rate of 5.4 mils per minute.

In order to reach a current consumption of 103 amperes, which is the trip setpoint, the downward force due to shaft elongation would need to be increased such that the current consumption is increased from 34.6 amperes to 64 amperes. This would require the force be increased by approximately 85%.

64 amps/34.6 amps = 1.85

An 85% increase in the metal removal rate will not occur for four reasons.

1. The net expansion of the shaft will be less when the water is colder, as discussed in Enclosure 1. There will be less total interference contact between the impeller and bowl liner when the water is colder.

2. The rate of heat up of the shaft, which is the factor that controls the "feed" of the impeller into the bowl, will be less when the water is colder. The colder water will slow the heat up rate.

3. The peak rate of material removal had already been reached. The data in the previous graph indicates that "topping off" of the current load had already begun when the shaft began to cool and the impeller started to disengage from the bowl liner. In other words, the rate of feed and the rate of material removal was close to the equilibrium or maximum point and would remain close to the 73.6 amperes recorded.

4. The physical process needed to nearly double the amperage consumed in material removal, to increase it from 34.6 amperes to 64 amperes, is not present. In the water test, the rate of feed of the impeller into the bowl was approximately 5.4 mils per minute. To increase the amperage by 85%, the feed rate would have to be increased from approximately 5.4 mils per minute to about 10 mils per minute. This would require an 85% increase in the heat input rate. This additional amount of heat being produced by contact between the rotating shaft and the rubber bushings is not likely. The heat source is finite and limited (per the discussion concerning 300 degrees F on Page 5 of this Attachment).

5a. NRC Request

What is the basis (analysis) for 8 hours of RCIC? The focus of this question lies in battery capability.

NPPD Response

A calculation was performed to show the following:

1) The capacity of the Division 1 125VDC battery will support RCIC operation for 9.5 hours;
2) The capacity of the Division 2 125 VDC battery with both divisions transferred to the Division 2 batteries will support RCIC operation for approximately 5 hours.

This results in the two bounding cases:

1) If the DC loads were not transferred, the single Division I bus would continue to support RCIC operation greater than 8 hours.
2) If the loads were transferred, the worst case that would maximize the discharge of the Division I battery would be a transfer just prior to the failure of Diesel Generator (DG) 2 at 1.5 hours. Once all loads are transferred to the Division 2 battery, the Division 1 battery is no longer discharging and the remaining capacity of 8 hours could be used. This would result in 1.5 hours of operation on Division 1, plus approximately 5 hours of operation on Division 2, plus 8 hours of operation back on Division 1 which is greater than the required 8 hours.

Therefore, since both of these cases exceed the 8-hour RCIC operation and they bound the spectrum of potential operator actions, the 8-hour assumption of RCIC operation remains valid.

Basis:

NEDC 04-047 (Enclosure 2) calculated the capacity of the 125VDC batteries using the following assumptions:

1. Batteries were assumed to be at end of life (i.e., 90% capacity, plus 5% design margin).
2. Automatic Depressurization System capability included.
3. DG start is supported at the end of battery capacity.

NEDC 04-047
125 VDC DIV I/II Sensitivity Study, SBO

Title: 125 VDC DIV I / II Sensitivity Study, SBO Extended Voltage Profile	Calculation Number: NEDC 04-047
	CED/EE Number: N/A
System/Structure: EEDC	Setpoint Change/Part Eval Number: N/A
Component: Various	Discipline: Electrical
Classification: [X] Essential; [ ] Non-Essential	SQAP Requirements Met? [X] Yes; [ ] N/A

Proprietary Information Included? [ ] Yes; [X] No

Description:

This "Information Only" sensitivity study evaluates the sequence of events on the Division I and II 125 volt DC system for the Station Blackout (SBO) scenario and determines bus voltages beyond T= 4 hours. The study was performed for a specific scenario and was derived from the analyses within calculations NEDC 87-131C Revision 9 and NEDC 87-131D Revision 8, and with input from CNS Operations.

Conclusions and Recommendations:

The sensitivity study determined voltages at the 125VDC RCIC bus, HPCI bus, SRVs, and other safety related equipment remain above minimum levels for the extended time considered in the scenario.

Rev. Number	Status	Prepared By/Date	Reviewed By/Date	IDVed By/Date	Approved By/Date
1	2	Dustin Kiekel/Marshall Van Winkle Dustin Kiekel 10-7-04 Marshall Van Winkle 10-7-04	Scott Bunting Scott Bunting 10-7-04	N/A	for M. McCormick 10-7-04
0	2	Dustin Kiekel	Scott Bunting	N/A	Dan Buman

Status Codes

1. Active
2. Information Only
3. Pending
4. Superseded or Deleted
5. OD/OE Support Only
6. Maintenance Activity Support Only
7. PRA/PSA

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	DESIGN INPUTS	REV. NO.	PENDING CHANGES TO DESIGN INPUTS
1	NEDC 87-131C	9	CCN 9C1, 9C2
2	NEDC 87-131D	8	CCN 8C1, 8C2, 8C3, 8C5
3	Procedure 2.2.25A	4	N/A
4	Procedure 5.3SBO	6	N/A
5	DI 2650 (Report No. EGS-TR-096600-02)	0	N/A
6	Procedure 5.3AC480	7	N/A

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	AFFECTED DOCUMENTS	REV. NUMBER
	None	

The purpose of this form is to assist the Preparer in screening new and revised design calculations to determine potential impacts to procedures and plant operations.®[1]

	SCREENING QUESTIONS	YES	NO	UNCERTAIN
1.	Does it involve the addition, deletion, or manipulation of a component or components which could impact a system lineup and/or checklist for valves, power supplies (breakers), process control switches, HVAC dampers, or instruments?	[ ]	[X]	[ ]
2.	Could it impact system operating parameters (e.g., temperatures, flow rates, pressures, voltage, or fluid chemistry)?	[ ]	[X]	[ ]
3.	Does it impact equipment operation or response such as valve closure time?	[ ]	[X]	[ ]
4.	Does it involve assumptions or necessitate changes to the sequencing of operational steps?	[ ]	[X]	[ ]
5.	Does it transfer an electrical load to a different circuit, or impact when electrical loads are added to or removed from the system during an event?	[ ]	[X]	[ ]
6.	Does it influence fuse, breaker, or relay coordination?	[ ]	[X]	[ ]
7.	Does it have the potential to affect the analyzed conditions of the environment for any part of the Reactor Building, Containment, or Control Room?	[ ]	[X]	[ ]
8.	Does it affect TS/TS Bases, USAR, or other Licensing Basis documents?	[ ]	[X]	[ ]
9.	Does it affect DCDs?	[ ]	[X]	[ ]
10.	Does it have the potential to affect procedures in any way not already mentioned (refer to review checklists in Procedure EDP-06)? If so, identify:	[ ]	[X]	[ ]

____

If all answers are NO, then additional review or assistance is not required.

If any answers are YES or UNCERTAIN, then the Preparer shall obtain assistance from the System Engineer and other departments, as appropriate, to determine impacts to procedures and plant operations. Affected documents shall be listed on Attachment 2.

PURPOSE:

NEDC 04-047 is an "Information Only" sensitivity study for event specific load profiles to address available bus voltages at a predetermined time after the scenario described herein. These load profiles are for both the 125 VDC Division I and II systems.

ASSUMPTIONS:

1. This sensitivity study was derived from the data and assumptions contained within calculations NEDC 87-131C Revision 9 and NEDC 87-131D Revision 8. These calculations should be referenced for Station Blackout loading and timing assumptions, except those listed herein.

2. Certain time steps were altered (combined) from those used in NEDC 87-131C and 87-131D due to the limitations of the EDSA software. These changes did not alter the assumed sequence of loads and thus are not an adverse change to the calculation methodology. Additionally, the Division II scenario assumes all Division I loads are transferred to Div II, the limitations of the EDSA models necessitate using the Division II time steps for the Division I loads. The time steps within NEDC 87-131C and 87-131D are very similar and therefore these changes do not adversely affect the results of the Division II scenario.

3. Appendix E describes the Division I scenario load assumptions, as they deviate from NEDC 87-131C Revision 9.

4. Appendix F describes the Division II scenario load assumptions, as they deviate from NEDC 87-131C Revision 9 and NEDC 87-131D Revision 8.

METHODOLOGY:

The following scenarios were used to establish voltage profiles with the CNS EDSA DC load flow models contained in NEDC 87-131C and NEDC 87-131D.

<u>125VDC Div I Scenario:</u>
A loss of offsite power (LOOP) is assumed to occur at T=0 seconds which initiates a scram. DG1 and DG2 are signaled to start, DG2 is assumed to start but DG1 is assumed to fail. Division 1 125 VDC battery picks up required loads, Division II 125 VDC system continues to be maintained by battery charger 1B (as fed by DG2). Station Blackout loading is assumed on Division 1 battery through T=9.5 hours.

Appendix A contains graphical representations of the specific bus voltage profiles through 9.5 hours. Appendix B contains the 125VDC Division I load flow results as obtained from the modified NEDC 87-131C EDSA model.

125VDC Div II Scenario:
A loss of offsite power (LOOP) is assumed to occur at T=0 which initiates a scram. This is sensed at T=1 second which causes various 4kV switchgear breakers to trip. At T=1 second the EDGs are signaled to start but are assumed to fail. A Station Blackout event begins. It is assumed that unique event timing has lead to the transfer the DIV I Panel A and DIV I RCIC Starter Rack to the emergency feed (SWGR 1B) prior to start of scenario, in accordance with Procedure 5.3AC480 step 1.16. Both Division I and Division II Station Blackout loading is assumed on the Division II battery through T=5 hours.

Appendix C contains graphical representations of the specific bus voltage profiles through 5 hours. Appendix D contains the 125VDC Division II load flow results as obtained from the modified NEDC 87-131D EDSA model.

Acceptance Criteria
NEDC 87-131C Revision 9 contains the minimum voltage criteria for the Division I loads. Specifically, the minimum acceptable voltage for the RCIC bus is 98 Volts and the minimum acceptable voltage for the Safety Relief Valves (MS-SOV-SPV71A - H) is 95.1 Volts DC (accounting for cable voltage drop).

NEDC 87-131D Revision 8 contains the minimum voltage criteria for the Division II loads. Specifically, the minimum acceptable voltage for the HPCI bus is 98 Volts and the minimum acceptable voltage for the Safety Relief Valves (MS-SOV-SPV71A - H) is 94.4 Volts DC (accounting for cable voltage drop).

NEDC 87-131C Appendix R, note 42, states the RCIC inverter (RCIC-IVTR-1A), powered from Panel AA2 circuit 11, rated minimum voltage is 105 VDC. The voltage drop from the board to the inverter is 0.179 volts, therefore the required minimum voltage was stated as 105.179 VDC. However, baseline functional tests performed under Report No: EGS-TR-096600-02 (ref. DI 2650) determined the inverter drop out voltage was 102.8 VDC. Including voltage drop between the board and the inverter, the minimum required voltage is 102.979 VDC.

CONCLUSION:

Based on the acceptance criteria discussed above and the results provided in Appendices A thru D, the 125 VDC Division I and II batteries have adequate capacity to maintain the minimum required voltage at the RCIC bus, HPCI bus and at the terminals of the Safety Relief Valve SOVs for the durations specified in the described scenario.

The results of the Division II scenario, Appendix D, determined inadequate voltage will be present at the RCIC inverter (<103VDC) for a short period of time (last 25 seconds) of the 5 hour scenario. The voltage is below the acceptance criteria due to the starting inrush current to the Fuel Oil Booster Pump Motor during the start of DG2. At this point in the scenario, the voltage recovers after the initial inrush of the booster pump motor and again after DG2, via battery charger 1B, takes the bus. Therefore the RCIC inverter could be reset almost immediately.

REFERENCES:

1. NEDC 87-131C Revision 9
2. NEDC 87-131D Revision 8
3. Emergency Procedure 5.3SBO Revision 6
4. Drawing 3058 Revision N47
5. Drawing 3059 sheet 4, Revision N06
6. Drawing 3059 sheet 5, Revision N09
7. DI 2650, EGS-TR-096600-02

APPENDICES:

A - Division I Scenario Voltage Profile Graph
B - Division I Scenario EDSA Output
C - Division II Scenario Voltage Profile Graph
D - Division II Scenario EDSA Output
E - Division I Scenario Load Assumptions
F - Division II Scenario Load Assumptions

LOAD DESCRIPTION PANEL/CIRCUIT	BUS ID	DESCRIPTION
RCIC-MO-131(CNTRL)	RCIC131C	Constant load except when valve is stroking, reference valve assumptions.
RCIC-MO-131	RCIC-131	The normally closed steam admission valve starts to open at T=2 sec. on low reactor water level. Based on a maximum design open stroke time of 18 seconds (Ref. IST Program) the valve is fully open at T=20 seconds but is modeled to end of the time step at T=21 seconds. The SBO coping analysis requires that RCIC cycle 6 times per hour after an initial cycle initiatedby low reactor level and tripped by high reactor water level. The cycling of this valve is modeled at its inrush value for the time T=138 seconds to T=144 seconds and then at its full load current for the time T=144 seconds to T=14400 seconds. Assuming full load current as a constant load for 14262 seconds (3.96hrs) bounds valve operation because load diversification based on stroke time, number of strokes per hour, number of hours, total time and inrush current yields a load value far less than the 6.5amp full load current, see below, (18sec/cycle)*(6 cycles/hr)*(4 hrs)*(1/14262sec)*(36amps) = 1.09amps. The valve loads are set to zero after T=14400 seconds because further intermittent use of RCIC is bounded by the conservative assumptions of the first 4hours.
RCIC-MO-27(CNTRL)	RCIC27C	Constant load except when valve is stroking, reference valve assumptions.
RCIC-MO-27	RCIC-27	The normally closed minimum flow bypass valve starts to open at T=2 sec. on low reactor water level and low RCIC pump flow to provide pump minimum flow protection. The valve starts to close when flow is about 20% of full flow (80 gal/min). It is conservatively assumed that the valve opens and immediately re-closes to create an overlap of the closing stroke inrush with othersystem loads at T=12 sec. and be fully closed at T=20 seconds but modeled to the end of the period at T=21 seconds. When the RCIC turbine is tripped by high reactor water, this valve will remain closed. The SBO coping analysis requires that RCIC cycle 6 times per hour, initiated by low reactor level and tripped by high reactor water level. The cycling of this valve is modeled at its inrush value for the time T=138 seconds to T=144 seconds and then at its full load current for the time T=144 seconds to T=14400 seconds. Assuming full load current as a constant load for 14262 seconds (3.96hrs) bounds valve operation because load diversification based on stroke time, number of strokes per hour, number of hours, total time and inrush current yields a load value far less than the 9.4amp full load current, see below, (11sec/cycle)*(6 cycles/hr)*(4 hrs)*(1/14262sec)*(58.75amps) = 1.09amps. The valve loads are set to zero after T=14400 seconds because further intermittent use of RCIC is bounded by the conservative assumptions of the first 4 hours.

LOAD DESCRIPTION PANEL/CIRCUIT	BUS ID	DESCRIPTION
AA3/8 RR SET 2-184-11	AA3-FU08	RRMG field breakers TCA1 and TCA2 energize momentarily at T=10 sec. to delay RPT on ATWS low level initiation. TCA1 and TCA2 are conservatively modeled as energized from T=5-19 sec. ARI SOVs are modeled as energized through 4 hours. After T=14400 seconds, the panel is assumed to be manually secured.

DIV II Scenario Loading Assumptions (for Division I loads)

LOAD DESCRIPTION PANEL/CIRCUIT	BUS ID	DESCRIPTION
RCIC-MO-131(CNTRL)	RCIC131C	Constant load except when valve is stroking, reference valve assumptions.
RCIC-MO-131	RCIC-131	The normally closed steam admission valve starts to open at T=2 sec. on low reactor water level. Based on a maximum design open stroke time of 18 seconds (Ref. IST Program) the valve is fully open at T=20 seconds but is modeled to end of the time step at T=21 seconds. The SBO coping analysis requires that RCIC cycle 6 times per hour after an initial cycle initiatedby low reactor level and tripped by high reactor water level. The cycling of this valve is modeled at its inrush value for the time T=138 seconds to T=144 seconds and then at its full load current for the time T=144 seconds to T=14400 seconds. Assuming full load current as a constant load for 14262 seconds (3.96hrs) bounds valve operation because load diversification based on stroke time, number of strokes per hour, number of hours, total time and inrush current yields a load value far less than the 6.5amp full load current, see below, (18sec/cycle)*(6 cycles/hr)*(4 hrs)*(1/14262sec)*(36amps) = 1.09amps. The valve loads are set to zero after T=14400 seconds because further intermittent use of RCIC is bounded by the conservative assumptions of the first 4hours.
RCIC-MO-27(CNTRL)	RCIC27C	Constant load except when valve is stroking, reference valve assumptions.
RCIC-MO-27	RCIC-27	The normally closed minimum flow bypass valve starts to open at T=2 sec. on low reactor water level and low RCIC pump flow to provide pump minimum flow protection. The valve starts to close when flow is about 20% of full flow (80 gal/min). It is conservatively assumed that the valve opens and immediately re-closes to create an overlap of the closing stroke inrush with othersystem loads at T=12 sec. and be fully closed at T=20 seconds but modeled to the end of the period at T=21 seconds. When the RCIC turbine is tripped by high reactor water, this valve will remain closed. The SBO coping analysis requires that RCIC cycle 6 times per hour, initiated by low reactor level and tripped by high reactor water level. The cycling of this valve is modeled at its inrush value for the time T=138 seconds to T=144 seconds and then at its full load current for the time T=144 seconds to T=14400 seconds. Assuming full load current as a constant load for 14262 seconds (3.96hrs) bounds valve operation because load diversification based on stroke time, number of strokes per hour, number of hours, total time and inrush current yields a load value far less than the 9.4amp full load current, see below, (11sec/cycle)*(6 cycles/hr)*(4 hrs)*(1/14262sec)*(58.75amps) = 1.09amps. The valve loads are set to zero after T=14400 seconds because further intermittent use of RCIC is bounded by the conservative assumptions of the first 4 hours.

DIV II Scenario Loading Assumptions (for Division I loads)

LOAD DESCRIPTION PANEL/CIRCUIT	BUS ID	DESCRIPTION
AA3/8 RR SET 2-184-11	AA3-FU08	RRMG field breakers TCA1 and TCA2 energize momentarily at T=10 sec. to delay RPT on ATWS low level initiation. TCA1 and TCA2 are conservatively modeled as energized from T=5-19 sec. ARI SOVs are modeled as energized through 4 hours. After T=14400 seconds, the panel is assumed to be manually secured.
DG1/1 FOBP	DG1FU01C	Control Power for the DG1 Fuel Oil Booster Pump. The DGDO Fuel Oil Booster Pumps start on the DG start signal. At the end of the Divison II scenario, DG2 is assumed to start and DG1 remains secured. Therefore the DG1 Fuel Oil Booster Pump remains secured.
DG1/1 DGDO-MOT-FOB1	DG1-FU01	The DGDO Fuel Oil Booster Pumps start on the DG start signal. At the end of the Divison II scenario, DG2 is assumed to start and DG1 remains secured. Therefore the DG1 Fuel Oil Booster Pump remains secured.
DG1/8 DG1 EXCITER PNL	DG1 –FU08	At the end of the Divison II scenario, DG2 is assumed to start and DG1 remains secured. Therefore the Exciter Panel is assumed at the minimum load
DG1/6 4160V SWGR EG1	DG1-FU06	At the end of the Divison II scenario, DG2 is assumed to start and DG1 remains secured. Therefore 4160 breaker EG1 is not energized for closing.

DIV II Scenario Loading Assumptions (for Division II loads)

LOAD DESCRIPTION PANEL/CIRCUIT	BUS ID	DESCRIPTION
		Division II Assumptions do not vary from NEDC 87-131D Revision 8.

5b. NRC Request

What are the Operator procedures/steps that would be used to transfer the 125 VDC RCIC Starter Rack to support continued RCIC operation (loss of all AC power, Station Blackout)?

NPPD Response

The Operator will transfer the RCIC Starter Rack per Procedure 5.3AC480, Revision 6, "480 VAC Bus Failure", prior to a Station Blackout (SBO), or upon an SBO, will recognize battery degradation and transfer, or be prompted to transfer by the Technical Support Center. Note that the procedure revisions indicated were those in effect during the 21 day period.

Basis:

Initial assumptions; Loss of Off-Site Power event, DG1 fails, DG2 starts and runs for approximately 90 minutes. A loss of off-site power results in an immediate reactor scram and turbine trip. Operations personnel would enter Emergency Operating Procedures for responding to the reactor scram and turbine trip. Due to no off-site power, Emergency Procedure 5.3EMPWR, "Emergency Power", Revisions 5 and 6, would be entered with a priority placed on restoration of off-site power, DG 1, SW, Reactor Equipment Cooling, and the plant air system. After plant conditions have stabilized, Operations personnel would begin to address other off normal conditions and alarms. Various 4160/480 VAC bus alarms would prompt entry into Procedure 5.3AC480, Revision 6, which, per Attachment 8 of the procedure, provides guidance to transfer the 125 VDC RCIC Starter Rack. If/when DG2 is lost, the station is in a blackout condition with only DC power available.

During a SBO, the operating crew will be in Procedure 5.3SBO, "Station Blackout", Revision 5. Depending on plant conditions and/or sequence of events, the 125 VDC RCIC starter rack will be powered from either Division 1 or Division 2. The starter rack can be transferred to the other division during the event, extending RCIC availability. The guidance for transfer is in Procedure 2.2.25.1, "125 VDC Electrical System (DIV 1)", Revision 4, Section 34, or Procedure 2.2.25.2, "125 VDC Electrical System (DIV 2)", Revision 4, Section 39. It is understood and trained on by operators that 125 VDC batteries will supply loads for a period of 4 hours without load stripping (4.5 hours on Division 2 for 10CFR50 Appendix R). There are two most likely success paths during SBO that will transfer the RCIC 125 VDC Starter Rack.

1) During the event, the operations crew will be monitoring both 125 and 250 VDC battery amperage and voltage. The 125 and 250 VDC bus voltage indication is available in the Main Control Room on Board C. The 125 VDC Bus 1A/1B meter green band is 125 to 131 VDC and the 250 VDC Bus 1A/1B meter green band is 265 to 272 VDC. The green band signifies the normal operating voltage range of the associated bus. If the 125 VDC bus voltage were to be out of band low, the meter banding would be a visual cue to the operator that bus voltage is not within normal specifications. The instruments meter banding is controlled by Engineering Procedure 3.26.1, "Meter Banding Control", Revision 0. The Control Room operations crew also records the voltage on these buses each shift,

as required by Procedure 2.1.12, "Control Room Data", Revision 31 and 32, so they are familiar with the normal bus voltages.

Control Room annunciation alarms C1/C-1 (C4/C-6), 250 VDC BATT CHARGER 1A(1B) TROUBLE and C1/C-2 (C4/C-7) 125 VDC BATT CHARGER 1A (1B) TROUBLE would also alert the Control Room operations crew of problems with the respective buses. Specifically, these alarms are received for the following problems: AC voltage failure, DC voltage high, DC voltage low, and output breaker tripped. In addition, Control Room annunciation alarm C1/C-1 (C4/C-6), 250 VDC BATT CHARGER 1A(1B) TROUBLE directs the operations crew to monitor 250 VDC bus voltage when the battery is suppling the bus alone. These trouble alarms in conjunction with the meter banding should prompt the operations crew to transfer the affected 125 VDC bus to the other division when required.

If the bus transfer was not previously performed, eventually 125 VDC battery degradation would result in the loss of 125 VDC loads including RCIC. Loss of RCIC and other 125 VDC loads would be easily recognized by the operations crew. Upon recognition, the operations crew would transfer the bus per Procedure 2.2.25.1, Revision 4, Section 34 or Procedure 2.2.25.2, Revision 4, Section 39 to the other division, and if required, restart RCIC.

2) The Emergency Response Facilities will be manned no later than1 hour into the event. Per Procedure 5.7.7, "Activation of TSC", Revision 29, the function of the Technical Support Center (TSC) is to:

> Provide facilities, communications, and technical data to support CNS Emergency Response Organization. TSC personnel shall research drawings, specifications, test data, and other Engineering data as required to provide technical support to Control Room Operations personnel by:
>
> Recommending courses of action which may be taken to mitigate consequences of the event.
>
> Evaluating effects of abnormal system configuration on future operational evolutions and to assure such evolutions are properly planned.
>
> Diagnosing station conditions and performing trending of key parameters to ensure technical evaluations are being conducted with the most current information.

Based on the function of the TSC, priorities for that facility during this event will be restoration of AC power, and the trending and status of the operating DC systems. Based on diagnosis of 125 VDC battery condition and trending those parameters, pending degradation can be determined, and transfer or removal of loads will be recommended to the Control Room.

Correspondence Number: NLS2004128

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing & Regulatory Affairs Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	