PrairieIslandNPEm Resource

From: Vincent, Robert [Robert.Vincent@xenuclear.com]
Sent: Friday, November 21, 2008 10:56 AM
To: Nathan Goodman; Richard Plasse
Cc: Eckholt, Gene F.; Davis, Marlys E.
Subject: SAMA RAI Response Letter Pages 57-74
Attachments: 20081121 Response to RAI Letter dtd 10-23-2008 Pages 57-74.pdf

Here's the final part

Bob Vincent
X7259

Hearing Identifier: Prairie_Island_NonPublic
Email Number: 165

Mail Envelope Properties (9FA1D9F2F220C04F95D9394E3CF02DABB2E5D8)

Subject: SAMA RAI Response Letter Pages 57-74
Sent Date: 11/21/2008 10:56:11 AM
Received Date: 11/21/2008 11:05:51 AM
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Files	**Size**	**Date & Time**
MESSAGE	57	11/21/2008 11:05:51 AM
20081121 Response to RAI Letter dtd 10-23-2008 Pages 57-74.pdf	3092164	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

RAI SAMA 8.a

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER would be viable and potentially cost-beneficial. Evaluate the following SAMAs or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at PINGP.

a. Procedure for manually controlling the degree of SG depressurization and reclosing the SG PORVs in the event core damage is imminent, in order to prevent or reduce the challenge to SG tube integrity.

NSPM Response to RAI SAMA 8.a

Procedural guidance similar to that suggested in this SAMA is already in place for events involving extreme damage to the plant (such as may occur during a security-related incident). The Extreme Damage Mitigation Guideline (EDMG) for injecting water into the steam generators includes the following direction for Technical Support Center (TSC) personnel:

- Monitor conditions and be prepared to recommend closure of the Steam Generator (SG) Power Operated Relief Valves (PORVs) in the event core damage is imminent in order to prevent a challenge to SG tube integrity.

Also, the Severe Accident Management Guideline (SAMG) procedure for injecting water into the steam generators provides direction to the plant staff in re-establishing water flow to the SGs following a core damaging event. The procedure requires that the negative impacts of injecting water into the SGs be identified and evaluated. The procedure includes a table of negative impacts to consider and a listing of actions that can be taken to reduce or mitigate these impacts, if the decision is made to use this strategy. The negative impacts are described in detail, including how depressurization of the SGs (to allow injection with lower pressure systems) can increase the potential for tube failure due to the higher differential pressures across the tubes.

The PINGP emergency response personnel (TSC and Operations staff) that respond to plant events requiring use of the Emergency Operating Procedures are the same personnel that respond to events requiring implementation of the SAMGs and EDMGs. These personnel are trained in the use of these procedures in response to an event similar to that described above.

Due to the guidance to the operations and emergency response staff already in place, implementation of this proposed SAMA would have no beneficial impact.

RAI SAMA 8.b

b. Procedure for enhancing manual operation of turbine-driven Auxiliary Feedwater (AFW) pumps including alternate water sources, and operator aids for using local flow indication to maintain SG level.

NSPM Response to RAI SAMA 8.b

The use of alternate water sources is already addressed in the post-accident procedures requiring operation of the AFW pumps, including the TDAFWP (e.g., Caution statements state that if Condensate Storage Tank (CST) level decreases to less than 10,000 gallons, then alternate water sources for AFW pumps will be necessary). Local manual operation of the TDAFWP may be required during a Station Blackout (SBO) scenario. An abnormal operating procedure provides direction necessary to perform these actions. This procedure also contains a step notifying the operator to refer to other procedures for possible sources of makeup to the CST (as CST water level is depleted by pump operation).

PINGP also maintains a special document called an "Alternate Source Book" (ASB) that provides information to personnel during off-normal plant operations and during implementation of SAMGs (Decision Maker, Evaluators and Implementers) when developing strategies to mitigate a severe accident. The ASB provides information on resources for:

- Electrical Power Supply
- Water Makeup Supply
- Pneumatic (Air) Supply, and
- Fission Product Scrubbing Supply

In addition to the normal and emergency sources of water to the AFW pumps called for in the EOPs, the ASB identifies a number of alternate on-site and external water sources for providing water to the SGs (see response to RAI SAMA 8.c below). Also, the EDMG for manual operation of TDAFW pumps also contains procedural guidance similar to that suggested in this SAMA (see the response to RAI SAMA 8.a. above for a discussion of the potential for use of the EDMG procedures in response to other events).

Due to the guidance to the operations and emergency response staff already in place, implementation of this proposed SAMA would have no beneficial impact.

RAI SAMA 8.c

c. Procedure and equipment for using a portable pump to provide feedwater to the SGs with suction from either the external fire ring header or intake canal.

NSPM Response to RAI SAMA 8.c

The suggested action is the subject of an EDMG procedure for injecting water into the steam generators. Such an action would be considered by the operators and emergency response personnel following an event involving loss of heat sink (see the response to RAI SAMA 8.a above for a discussion of the potential for use of the EDMG procedures in response to non-extreme damage scenarios). A portable, diesel-powered pump and instructions for connecting the pump to supply water to the SGs from various sources (including the river) is in place, and emergency response personnel have been trained on the use of the equipment and on the procedures.

The ASB also identifies the portable diesel pump as a potential means of delivering water to the SGs, and refers the reader to the EDMGs for guidance in implementing this strategy. In addition, the ASB identifies other potential water sources, including 1) connection to the fire main using fire hoses and 2) drawing water from the external circulating water basins or Mississippi River using portable suction hoses and a fire pumper truck (supplied from the local fire station) delivering water to fire hydrant connections. These strategies would provide the water to the SGs via fire hoses connected to either the condensate system (condensate polisher strainer drains) or to the SG blowdown line drains.

Based on the procedures and equipment already available, NSPM considers this strategy to have been already implemented at PINGP.

RAI SAMA 8.d

d. Procedure for recovering emergency diesel generators D-1 and D-2 by supplying alternate cooling from well water or fire water through a spool piece on the inlet to the emergency diesel generator heat exchangers.

NSPM Response to RAI SAMA 8.d

Sections F.5.1.1 and F.5.1.2 of the ER describe the identification of candidate SAMAs through the review of PRA basic event importance measures. In general, events having a Risk Reduction Worth (RRW) importance measure of 1.02 or greater were considered for SAMA identification. Failure of the Cooling Water (CL) system supply to the Unit 1 Emergency Diesel Generators (EDGs) was modeled explicitly in the Rev. 2.2 SAMA PRA models (failure of active supply valves to open and remain open, common cause failure (CCF) to open, and failure of normally-open manual valves in the supply lines to remain open). The importance measures of all of these events in the Unit 1 and Unit 2 CDF cutsets show that this function is not providing a significant contribution to the overall PRA results (RRW measure is approximately 1.001 or less for all events, including the CCF event).

In addition, an EDMG procedure provides the procedural guidance recommended in this suggested SAMA (see the response to RAI SAMA 8.a above for a discussion of the potential for use of the EDMG procedures in response to non-extreme damage scenarios). The strategy is to provide a means to cool EDG D1 or D2 independent of the Cooling Water system. An external cooling supply is provided by removing the spool piece between the existing Cooling Water system supply control valve and the diesel heat exchangers.

Therefore, as the importance of these events was previously evaluated to fall below the SAMA candidate screening criterion, and since the procedures are already in place, NSPM considers this strategy to have already been implemented at PINGP.

RAI SAMA 8.e

e. As an alternative to SAMA 15 (Portable DC Power Source), reconfiguring the non-safety main feedwater loads to be powered from DC Bus B rather than the addition of a portable DC power source for 21 AFW pump breaker control as proposed for SAMA 15.

NSPM Response to RAI SAMA 8.e

ER Section F.6.6 showed that SAMA 15 had a small positive net value. However, changing the DC power supplies to the Unit 2 Main Feedwater system loads (instead of the associated motor-driven AFW pump) involves modifications to a larger set of components (pump breaker control power, feedwater regulating and bypass valves, etc.). In addition, the suggested SAMA would extend the DC power asymmetry between the units to the Main Feedwater system (in addition to the AFW system) and additional costs for procedure changes and training would be required. The modification would cost significantly more than the averted cost-risk estimate associated with SAMA 15 ($0 for Unit 1 and $19,324 for Unit 2) and would provide no additional risk benefit.

Therefore, the proposed SAMA would not be cost beneficial.

RAI SAMA 8.f

f. Modifying the charging pump(s) electrical connections to enable re-powering from alternate 480VAC power supply (e.g., opposite unit) using pre-staged cables.

NSPM Response to RAI SAMA 8.f

The important safety function supported by the charging pumps as modeled in the PRA is to provide water for Reactor Coolant Pump (RCP) seal injection. RCP seal injection, one of the two available means of RCP seal cooling, can be provided by 1 of 3 charging pumps. In the event that seal injection is lost, the seal cooling function is provided automatically by Reactor Coolant System (RCS) water flowing through the seals after having been cooled by passing through the RCP thermal barrier heat exchanger (TBHX), which is cooled by the Component Cooling (CC) system. The only support system shared by the charging pumps and the CC system pumps is AC power (both sets of pumps are supported by 4KV AC buses 15 and 16 on Unit 1, and 25 and 26 on Unit 2). Therefore, one means of losing RCP seal cooling is to lose safeguards AC power (station blackout).

However, unlike many other PWRs, this is not the dominant contributor to the seal LOCA core damage frequency at PINGP. Non-SBO induced RCP seal LOCA sequences contribute approximately 26% of the Unit 1 CDF [22% of the Unit 2 CDF], while SBO-induced RCP seal LOCA sequences contribute only approximately 9% [8%]. This is due to the ability to cross-tie the train-related 4kV buses between units, the availability of dedicated emergency diesel generators for each 4kV safeguards bus, and the differences between the EDG sets between the units (different cooling systems, different manufacturers, etc.). The dominant sequences involving loss of all RCP seal cooling involve loss of Cooling Water (CL), which fails the CC system and support for ECCS injection systems, and ultimately failure of the normal supply of water to the charging pumps from the Volume Control Tank (VCT), followed by failure of the transfer of the charging pump suction supply to the Refueling Water Storage Tank (RWST). SAMAs 2, 3, 9, 10, 12, 19a were developed and evaluated to address the CL, CC, and RWST to charging pump suction supplies for these important sequences.

Both the charging pumps and the CC pumps for each unit are already each powered from two independent trains of safeguards 4kV AC power, and each of those trains of AC power can already be transferred to the opposite unit train-related 4kV bus. The charging pumps are powered from safeguards 480V AC power; pumps 11, 12, and 13 [21, 22, and 23] are powered from 480V buses 121, 111, and 121 [221, 211 and 221] respectively via MCCs 1K2, 1K1, and 1K2 [2K2, 2K1, and 2K2], respectively. The CC pumps are powered from 4kV safeguards buses 15 [25] and 16 [26], respectively. Failures of individual 4kV buses, and failures of electrical equipment between the 4kV buses and the charging pumps (in which the 4kV buses remain available) are low probability events and do not result in loss of more than two charging pumps. In these cases, at least one charging pump and one CC train remains available to support both means of RCP seal cooling. For this reason, the basic event importance for all such equipment failures falls below the screening thresholds described in Sections F.5.1.1 and F.5.1.2 of the ER.

Therefore, implementation of the suggested SAMA would not be cost beneficial at PINGP.

RAI SAMA 8.g

g. Installing a connection flange and valve on safety injection (SI) pump flow test return line to the refueling water storage tank to enable cross-connection of SI pumps to AFW piping via a temporary connection/hose.

NSPM Response to RAI SAMA 8.g

As described in the response to RAI SAMA 8.c above, a number of alternative means of providing an independent supply of water to the steam generators in the event that all other water sources are unavailable have already been implemented via the ASB, EDMG procedures, and the SAMGs. Connection of the Safety Injection (SI) pumps to divert RWST water to the SGs on the same unit experiencing a loss of heat sink via temporary connections may not effectively reduce the risk of core damage from this event. Supplying water to the SGs from the SI pumps on the opposite unit would involve a greater length of hose, and the hose required would have to able to withstand high pressures.

Given the alternate supplies and strategies already available to the operators, implementation of the suggested strategy would not be cost beneficial at PINGP.

RAI SAMA 8.h

h. Modifying the charging and volume control system to allow cross-tie of the charging pumps from opposite unit using temporary connections.

NSPM Response to RAI SAMA 8.h

The risk-significant function supported by the charging pumps is to provide RCP seal injection, preventing an RCP seal leak from occurring. The most probable situation in which all three charging pumps fail is a single unit station blackout (SBO) event. Due to the ability to crosstie the train-related AC buses between units at PINGP, the potential for a single unit SBO to occur is lower than at single unit plants and at multiple unit plants without this capability. Core damage sequences in which the charging pumps on the opposite unit may be available for cross-tie to the affected unit have a frequency of approximately 2.3E-6/rx-yr on both units, and are dominated by non-SBO-related RCP seal LOCAs. This is considered an upper bound frequency; in some of these sequences power may be lost to the opposite unit standby charging pumps, or other equipment or operator failures may prevent them from being used. These factors were not investigated fully for this response.

This RAI suggests that temporary connections could be used to make the necessary alternate flow path available to the other unit. However, the charging pumps are positive displacement pumps that develop the very high discharge pressures necessary for injection into the RCS. For personnel safety, it is assumed that this connection would involve a modification to install a hard-pipe line (meeting current charging pump discharge piping standards) between the Unit 1 and Unit 2 charging pumps. At least one manual valve on either end would be required for isolation from the normally-operating high-pressure charging system. Both of these valves would have to be opened to provide flow through the temporary connection pathway. The shortest path for the piping run would be to cross the Auxiliary Building 695' elevation floor in the overhead of the CC heat exchanger room between the units. Assuming this minimum-distance pathway is available and can be used for the modification, roughly 100'-125' of high pressure piping would need to be installed.

If a (potentially optimistic) 1E-1 probability of operator failure to perform this local recovery action prior to development of an RCP seal LOCA is applied, then the core damage risk savings associated with this SAMA is approximately 2E-6/rx-yr on each unit. However, most of these core damage sequences would not bypass containment (AFW is generally available in these sequences, such that induced SGTR is not a factor).

SAMA 3 (Provide Alternate Flow path from RWST to Charging Pump Suction) is comparable to this SAMA both in terms of CDF reduction and impact to dominant core damage sequences and release categories. From the ER Section F.7.2.3, the calculated averted cost-risk values for SAMA 3 were:

Unit	Base Averted Cost-Risk (ACR)	95th Percentile ACR
Unit 1	$74,956	$179,894
Unit 2	$76,654	$183,970
Total	$151,610	$363,864

SAMA 3 involved installing a bypass around the motor-operated valve that must open to supply charging pump suction flow from the RWST upon loss of VCT level. This line would include an air-operated valve, whereas the suggested SAMA investigated here would include two manual isolation valves. The additional, new piping installed under SAMA 3 would need to be far shorter in length than would this SAMA, and the piping design and installation requirements would be less as SAMA 3 involved installation on the suction side of the charging pumps. The SAMA 3 cost estimate was $250,000 per unit. If the proposed SAMA could be installed for this amount, then the modification would only be cost-beneficial at the 95th percentile ACR for both units combined. However, based on the considerations outlined above, the cost to implement this modification would be expected to exceed the SAMA 3 implementation costs. Therefore, implementation of the suggested SAMA is not considered to be cost beneficial for PINGP.

RAI SAMA 8.i

i. Purchase or manufacture of a gagging device that could be used to close a stuck-open SG safety value on the ruptured steam generator prior to core damage in SGTR events

NSPM Response to RAI SAMA 8.i

Two recent license renewal applicants addressed this SAMA as part of their analysis (either on initial submittal or in response to an RAI). Beaver Valley found it to be cost beneficial at the upper bound of a sensitivity analysis, whereas Indian Point found it to be cost beneficial in the base case. Both plants used a $50,000 estimated implementation cost for this SAMA.

The Beaver Valley submittal stated that this SAMA involved procedure changes to require the operators to close the primary loop isolation valve associated with the ruptured SG, and then to gag the stuck open relief valve. This would reduce but not eliminate the radiation exposure to personnel received during the relief valve gagging operation. Like Indian Point, PINGP does not have RCS loop isolation valves. Therefore, in addition to steam and heat-related risk to personnel, the gagging operation is assumed to involve some additional amount of radiation exposure risk. The design and implementation of any gagging device would have to address issues related to personnel safety.

Based on the PRA Rev. 2.2 SAMA results, the CDF associated with SGTR events in which gagging a stuck open relief valve may be of value is about 2E-7/yr for Unit 1 and 1E-6/yr for Unit 2. These sequences involve failure of the operators to cooldown and depressurize the RCS prior to stopping the primary-to-secondary leakage and prior to SG overfill, followed by failure of a SG relief valve to remain closed. The Indian Point RAI response also assumed that implementation of this SAMA would effectively eliminate the risk of temperature-induced SGTR events. This produced a large positive net value for this SAMA. As Indian Point is located near New York City, it may be expected that the dose savings might be very large there, whereas it might not be expected to be so large at Prairie Island where the local population is far lower. However, from the PINGP ER, Sections F.7.2.1.3 and F.7.2.1.4, SAMA-17 and SAMA-19a were found to work on the same set of core damage sequences as may be expected from this SAMA. SAMA 17 and SAMA 19a showed positive benefit to the SGTR avoided costs for both units (U2 more than U1), although, overall, the numbers were negative based on the high costs of those modifications. Given the relatively lower implementation cost associated with this SAMA, this modification may be cost beneficial. This SAMA has been entered into the corrective action program for a more detailed examination of viability and implementation cost.

Enclosure 2

PINGP Calculation ENG-ME-148, Revision 1

10 Pages

Enclosure 2
PINGP Calculation ENG-ME-148, Revision 1

PINGP 1083A, REV. 0
Page 1 of 2 (Front)
Retention: Life

U950815010

CALCULATION VERIFICATION CHECKLIST

Calculation No.: ENG-ME-148 Revision No.: Ø1

Use of Computer for Calculation

- [X] Manual Calculation (no computer results)
- [ ] Computer
  - [ ] Verified Program (Reference ____ Provides Verification)
  - [ ] Unverified Program (Verification of Results Required)

Verification Item * (Refer to Site Engineering Manual, Administrative Standard 1.2.3) **Initials / Date**

1.0 Purpose — [initials] 18/4/95

- Clear objective and problem statement.
- Identification of affected structure, system, and/or component.
- Identification of the intended use of the calculation results.
- Identification of summary results.

2.0 Methodology — [initials] 18/4/95

- Discussion of the method/approach and major steps.
- Definition of any limitations of methodology.

3.0 Acceptance Criteria — [initials] 18/4/95

- Clear definition of the acceptance criteria.
- Exceptions clearly defined.

This "calculation" is a "position paper" format. Acceptance criteria is not directly applicable

4.0 Assumptions — [initials] 18/4/95

- Sufficient rationale to permit verification of assumption.
- Unverified assumptions identified a such.
- References provided for assumptions.

5.0 Design Inputs — [initials] 18/4/95

- All applicable design inputs identified.
  - CODES, (ASME, CFR, STATE, etc)
  - STANDARDS (IEEE, ANSI, ANS, ASTM, etc)
  - USAR
  - Design Criteria
  - Input Data
  - Regulatory Guides/Requirements (NRC, EPA, STATE, etc.)
  - Design Bases Documents
- Appropriate verification of walkdown information.

As appropriate

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Page 2 of 2 (Back)

CALCULATION VERIFICATION CHECKLIST

(Continued)

Verification Item

6.0 Calculations

- Correct formulas/methods selected to support the problem statement and objective.
- Formula variables clearly labeled (including engineering units) and consistent with source references.
- Review of computer program data input/output.
- Reference provided, as appropriate, for sketches.
- Sufficient bases/rational provided to permit verification of engineering judgment.

This "calculation" is a "position paper" format. These verification items do not necessarily apply.

[signature] / 8/4/95

7.0 Conclusions

- Clear statement of the calculation results and consistency with the problem statement and objective.
- Acceptability of the results clearly defined.
- Recommendations for unacceptable results, provided, if applicable.
- Clear definition of limitations or requirements imposed by the calculation necessary to maintain the validity of the results.

[signature] / 8/4/95

8.0 References

- All pages of the attachments labeled with appropriate information (attachment no., project calculation no., revision no., number of sheets (Sht of _))

[signature] / 8/4/95

10.0 Administrative

- Calculation prepared neat and legibly with sufficient contrast to allow satisfactory copies to be produced.
- Calculation number/revision and sheet number provided on each page.
- Revision block and revision bars completed for revised calcs.
- All attachments provided are included in page numbering.
- Calculation's name and subject appropriately identified.
- Calculation properly logged in the Site Analysis Index.
- Analysis of Record form completed (PINGP 1075).

[signature] / 8/4/95

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NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

CALCULATION COVER SHEET

Calculation Number: ENG-ME-148

Calculation Rev. No.: -01- **Addenda No.:** —

Calculation Title: Cooling Water Header pipe Failure Causing Flooding in the Auxiliary Feedwater Pump/instrument Aircompressor room.

Safety Related?: Y

Calculation Verification Method (Check One):

_____ Design Review _____ Alternate Calculation _____ Qualification Testing

Scope of Revision: Additional Analysis on operability of Main Feed System.

Note: for clarity this evaluation is not in the calculation format recommended in the NSP Engineering Manual. This cover sheet is used to document reviews and approvals, and to ensure proper entry in Analysis of record data base.

Documentation of Reviews and Approvals:

Originated By:	SIGNED FAX FROM Arsethur, Attached	**Date:**	8/2/95
Checked By:	[signature] Robert L. Cole	**Date:**	8/4/95
Verified By:	[signature]	**Date:**	8/7/95
Approved By:	G Hore	**Date:**	8/8/95

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FORM 1083

CALCULATION COVER SHEET INSTRUCTIONS

1. **Calculation Number** – A unique calculation number is required for each NSP generated calculation in accordance with NPSP–E–7.1. Contact NPD General Superintendent Engineering for number assignment.
2. **Calculation Revision and Addenda** – Enter as appropriate. If this is not a calculation addenda input "NA" in this field. (Addenda is a supplement to an existing calculation where as a revision is a change to an existing calculation)
3. **Calculation Title** – Enter Calculation Title
4. **Safety Related** – Indicate Yes or No (See 5 and 7 below)
5. **Calculation Verification Method** – Identify verification method used if required. Refer to ANSI N45.2.11–1974 or the Site Engineering Manual for additional guidance. This field is required for all safety–related calculations.
6. **Scope of Revision** – Enter as Appropriate. If original calculation, state "Original" in this field.
7. **Documentation of Reviews and Approvals** – Sign and date fields as needed. Refer to the Site Engineering Manual for additional guidance.

NOTES:

A. All calculations require the signature of the Originator.

B. All calculations require checking (confirmation of numerical accuracy etc.) and the signature of the checker.

C. Safety–related calculations require verification (confirmation of the appropriateness of calculation method, use of design inputs, reasonableness of results, etc.). The originators supervisor <u>may not</u> be the verifier.

D. All calculations require the signature of the Approver.

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Enclosure 2
PINGP Calculation ENG-ME-148, Revision 1

PINGP 1063, Rev. 1
Front Page 1 of 1
Retention: Life

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

CALCULATION COVER SHEET

Calculation Number: ENG-ME-148

Calculation Rev. No.: -01- Addenda No.: —

Calculation Title: Cooling Water Header pipe Failure Causing Flooding In The Auxiliary Feedwater Pump/instrument Aircompressor room.

Safety Related?: Y

Calculation Verification Method (Check One):

___ Design Review ___ Alternate Calculation ___ Qualification Testing

Scope of Revision: Additional Analysis on operability of main Feed System.

Note: for clarity this evaluation is not in the calculation format recommended in the NSP Engineering Manual. This cover sheet is used to document reviews and approvals, and to ensure proper entry in Analysis of record data base.

Documentation of Reviews and Approvals:

Originated By:	R.V. Sethu	Date:	8/2/95
Checked By:		Date:	
Verified By:		Date:	
Approved By:		Date:	

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July 17, 1995

POSITION PAPER
ON
COOLING WATER HEADER PIPE FAILURE CAUSING FLOODING
IN
THE AUXILIARY FEEDWATER PUMP/INSTRUMENT AIR COMPRESSOR ROOM

1. BACKGROUND

The Prairie Island Individual Plant Examination (IPE) Report [Ref. 1] in Section 3.3.8 discusses the Internal Flooding Evaluation as part of the IPE process required by NRC Generic Letter 88-20 . A Probabilistic Risk Assessment (PRA) on internal flooding was performed to determine potential vulnerabilities due to flooding from sources such as "tank overfilling, hose and pipe ruptures, and pump seal leaks." The report concludes that the total core damage frequency (CDF) from internal flooding events for Prairie Island is 1.04 E-5/year. This constitutes 21 percent of the total CDF of 5.0 E-5 calculated in the study from all events.

The single most significant flood that accounted for almost all of the CDF from internal flooding events was a flood in the Auxiliary Feedwater Pump/Instrument Air Compressor Rooms due to a break of one of the two 24" Cooling Water (CL) loop header lines that traverse these rooms (Damage Class FEH-TB1 of Reference 1). The report postulates that the resulting flood will cause loss of all auxiliary feedwater pumps, loss of all instrument air, and loss of MFW pump lube oil cooling. Reactor trip is successful along with reactor coolant pump (RCP) seal cooling. Secondary cooling fails due to failure of auxiliary feedwater (AFW) and main feedwater (MFW). Short term reactor coolant system (RCS) inventory fails due to loss of pressurizer power operated relief valves (PORVs) which fail closed on loss of instrument air. No other internal flood scenario had a significant impact on core damage.

The initial action item resulting from this PRA study was to quantify the rate of rise in water level inside the AFW room, to determine the most feasible solution to mitigate the impact of such a postulated flood, and to recommend modifications to the procedures and/or the CL piping and AFW rooms, if necessary. However, prior to proceeding with any major modifications to the plant as a result of this study, it was decided to perform a deterministic examination of the postulation of such a catastrophic pipe failure in a "moderate energy" line with guidance from current NRC regulatory positions and the PI licensing basis.

Review of this sequence of events has resulted in further input on the possibilities of loss of main feedwater, condensate, and heater drain pumps due to loss of lube oil cooling. The break in the auxiliary feedwater pump room is down stream of the tap for the turbine building which supplies the condensate, heater drain, and feedwater pumps lube oil cooling. Since there is enough pressure to force water out of the break (35psi per hydraulic analysis) this same pressure plus the friction head will be available to provide flow to the turbine building. Therefore cooling to these pumps would be maintained even though it would be reduced. The turbine buildings cooling water supply also has isolation motor valves MV-32031,3 which can be closed to isolate the turbine building from the break and has a cross connect that can be used to supply cooling water to both turbine buildings from either remaining cooling water header

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should a break occur in the auxiliary feedwater pump room. Since there are capabilities to cool the main feed system pumps both during the break and after isolation it is not credible to assume that these pumps are lost due to the assumed break in the auxiliary feedwater pump area.

2. PURPOSE OF THIS POSITION PAPER

The objective of this paper is to review the postulation of a double-ended guillotine pipe break in the cooling water header piping and its consequences. The break postulation is reviewed from a deterministic standpoint and is based on current Prairie Island licensing basis, plant material condition, and other factors.

This paper does not address other modifications suggested in the PRA study to lower the total CDF value, such as procedural changes to allow station air to be crosstied with the instrument air and others.

3. EFFECTS OF AND MITIGATING FACTORS FOR DOUBLE ENDED GUILLOTINE BREAK IN ONE CL LOOP HEADER

Under the present plant configuration, an instantaneous and complete double-ended guillotine break in the 24-inch cooling water loop header will result in a high flow of cooling water from both segments (normal and back flows) of the double-ended break. This large flow will result in the rise of water level of six inches in both AFW rooms within a few seconds. Six inch rise in water level inside the AFW rooms will result in the failure of electrical equipment in the rooms, loss of instrument air and the AFW pumps.

However, an instantaneous and complete double-ended break is not a design basis break, is outside the current licensing basis for the plant, and is a highly unlikely event. A more likely scenario could be a leak before a break in the header piping as an initiating event. The AFW rooms are periodically walked through by operators, engineering and maintenance staff, and security staff. Any leak in the cooling water header will be easily noticeable and operator action can be initiated within an hour of its occurrence. This provides assurance that any leak in the CL header loop piping will be detected and appropriate action taken before a major break in the line occurs.

Operations Manual H10.2 [Ref. 5] provides guidance for repairing code piping containing flaws that exceed code acceptance limits. This guidance is based on Generic Letter 90-05. The NRC staff reviews any leak before break analysis based on fracture mechanics technology utilizing the procedures given in Reference 6. If flaws are detected in the CL piping, fracture mechanics technology can be applied "to demonstrate that high energy fluid system piping is very unlikely to experience double-ended ruptures or their equivalent as longitudinal or

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diagonal splits."[Ref. 6]. Therefore, administrative, operational and analytical procedures are in place to provides sufficient assurance that flaws and through wall cracks, if detected, will be repaired or analyzed to avert any catastrophic failure of high or moderate energy piping systems such as the CL loop header piping.

4. OTHER WATER LINES INSIDE THE AFW ROOMS

A 12-inch condensate make-up water line crosses the two AFW rooms in the trench. The pressure in this line is small and is equal to the head of water in the condensate make-up storage tanks. The material of construction is austenitic steel where the "leak-before-break" criterion for austenitic steel is well established and accepted by the NRC. Because of its low pressure, a double ended break in this pipe is extremely unlikely and would not result in loss of lube oil cooling for the condensate, heater drain or the main feedwater supply system.

Smaller cooling water branch lines (6 and 4 inch branches from the CL loop header to the AFW pumps) in the AFW room were replaced during the Cooling Water Header Replacement Project (92Y170) in 1992. Here too, the leak before break criterion is applicable. Any leaks occurring in these pipes will be readily noticeable and can be isolated.

5. REVIEW OF THE MATERIAL CONDITION AND LICENSING BASIS OF THE CL SYSTEM

The Cooling Water System is a safety related system designed and constructed to Design Class I and QA Type I standards. These standards of design and construction are much more stringent than the standard quality utilized in industrial and fossil plant design and construction. Additionally, the CL header piping was completely replaced during the two unit outage in November 1992. The new piping is 33 percent thicker (½" compared to the original thickness of 3/8"). Also the internal surface of the new header pipe is coated with an epoxy coating to inhibit microbiologically induced corrosion (MIC). This additional margin above and beyond the design requirement should contribute to lowering the failure frequency of the CL header piping.

A review of the licensing basis for the PI plant [Ref. 4] reveals that the CL system is not a high energy line (defined as those having a service temperature above 200°F <u>and</u> a design pressure above 275 psig). The licensing basis at PI does not postulate any breaks or cracks in piping systems that do not fall into the definition of a high energy line. These low energy lines (as opposed to high energy lines) are analyzed and designed to the USAS B31.1-1967 piping code with no line breaks or cracks assumed.

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However, current regulatory position for cooling water (service water) system would place the CL system as a moderate energy line. For this category, the NRC Standard Review Plan 3.6.1 and 3.6.2 (NUREG-0800) reviews the effects of a through-wall crack opening of ½ the pipe diameter in length and ½ the pipe wall thickness in width be assumed at locations that would result in the most severe environmental consequences. Per SRP 3.3.1, the fluid flow from the crack should be based on a circular opening of area equal to the rectangular through-wall crack opening described above. **It should be emphasized that this break criteria does not apply to the Prairie Island licensing basis but is presented here to quantify a break size that would be reasonable, from current licensing viewpoint, for studying the flooding vulnerability in the AFW rooms.**

The postulated crack opening in the cooling water line will result in a 2" diameter hole in the 24" Loop A or B line inside one of the two AFW rooms. Conservatively assuming that the pressure at the crack opening location to be equal to the operating pressure of 100 psig (this assumes no friction loss in the piping between the pumps and the crack location), the flow from a 2" diameter orifice is 673 gpm [Ref. 3]. Supporting calculations performed by NSP's Nuclear Analysis Department [Ref. 7] show that this flow rate can be readily handled by the floor drains, trench, and the gap under the doors leading to the AFW rooms with less than 3 inch rise in water level. The water level in the adjacent AFW room will not rise above the 6" level necessary to cause the equipment in the rooms to fail even with the fire door in its fully open position. Therefore, these postulated break scenarios will not cause the loss of the second AFW pump per unit and at least one instrument air compressor.

Also, it has been judged that these break scenarios will not cause the loss of lube oil cooling for the main feed pump or the condensate pump as postulated in the report and as re-stated in the Background portion of this paper. Preliminary evaluations regarding continuance of flow, during a postulated double ended break, through the 24" Turbine Building branch and ultimately the 3" supply line (3-2CL-9) to the feedwater pump and the heater drain pump oil coolers as well as the 1½" supply line (1½-2CL-17) to the condensate pump oil coolers have been performed. Based upon these evaluations it has been judged that there would be sufficient flow available, during a postulated break, to adequately supply the associated sets of pump oil coolers. The system engineers indicate from past experience that the systems oil coolers are capable of flow interruption for periods exceeding an hour due to the mass in the system oil coolers without taking the pumps out of service. Also once operator actions to isolate the break are initiated the cooling water flow to main water supply system lube oil coolers can be maintained.

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6. CONCLUSION

This position paper has presented deterministic arguments why the AFW rooms need not be modified as a result of the analysis from damage Class FEH-TB1 - Flood with Loss of Secondary Cooling and Bleed and Feed described in the PRA study [Ref. 1]. These reasons are based on:

- The loss of the normal steam generator cooling is not a credible assumption due to flooding of the auxiliary feedwater pump room.
- Instantaneous and complete double-ended break in the cooling water header loop piping is an extremely unlikely event and is outside the design and licensing basis for the system. The concept of "leak-before-break" in moderate to low energy lines will provide sufficient warning time for initiation of operator action to isolate the loop and other measures.
- A more reasonable scenario of a postulated crack in a moderate energy line, such as the cooling water header line, is supported by the current regulatory positions although the Prairie Island licensing basis does not require the postulation of a crack in the cooling water line. A crack in one of the two header loops will not cause the loss of the equipment in the adjacent AFW room.

Therefore, it is Northern States Power's position that no modifications need be made to the AFW rooms.

7. REFERENCES

1. Prairie Island Individual Plant Examination (IPE), NSPLMI-94001, Rev. 0, February 1994.

2. Pipe Failure Study Update, EPRI Report, TR-102266, April 1993.

3. Calculation of the Flow Through a 2" diameter Orifice, NSP Calc. No. ENG-ME-148, Rev. 0 (attached).

4 Prairie Island FSAR, Appendix I.

5. Prairie Island Operations Manual H10.2, Rev. 0

6. "NRC Leak-Before-Break (LBB.NRC) Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads," Topical Report NUREG/CR-4572, May 1986.

7. Detailed Analysis of Auxiliary Feedwater Pump Room Internal Flooding, NSP L&MI File V.SMN.94.006 prepared by D.R. Brown, verified by R. Best, 4/7/94.

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