MONTICELLO NUCLEAR GENERATING PLANT		3004	
TITLE:	SRI REVIEW AND APPROVAL FORM	Revision 12	06/02/92
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SRI No. 93-006 Rev. No. ϕ Add. No.

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TITLE: BASIS FOR CONTINUED OPERATION WITH CRACK IN CORE SPRAY HEADER

1. Approval to Prepare SRI: [signature] Date 3/8/93
   Gen Supt Engr or Designee
2. Prepared By: [signature] Date 3/8/93
3. Reviewed By: Bruce Parsons Date 3/9/93
4. Reviewed By DCA: Ken Kuhns Date 3/9/93
5. Operations Committee Review: Meeting No. 1886 Date 3-9-93
6. USAR or Subsequent Submittal Affected: Yes ____ No X (If Yes, SAC Review Required)

   SAC Review Completed: N/A Date 3-9-93
7. License Amendment or
   Unreviewed Safety Question: Yes ____ No X (If Yes, NRC Authorization is required; complete a., b., c., below).

   Ref: 50:59(c)
   a. SAC Review Completed: ____ Date ____
   b. Amendment Request Transmitted: ____ Date ____
   c. Authorization Letter Received: ____ Date ____
8. SRI Approval
   a. Gen Supt Engr or Designee [signature] Date 3/9/93
   b. Design Change Administrator Ken Kuhns Date 3/9/93
9. SRI Completed
   a. Responsible Individual ____ Date ____
   b. Design Change Administrator ____ Date ____

FOR ADMINISTRATIVE USE ONLY	Resp Supv: GS E	Assoc Ref: AWI-05.01.09	SR: N	Freq: 0 yrs
	Doc Type: 3020	ARMS: 3004	Admin initials: S/U (f)	Date: 7/23/92

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MONTICELLO NUCLEAR GENERATING PLANT		3004	
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INSTRUCTIONS (Ref: 4 AWI-05.01.09)

1. Obtain approval from Gen Supt Engr or designee to prepare an SRI.
2. Obtain an SRI number from the Design Change Administrator.
3. Prepare descriptions for items checked "Yes" on Page 2 of Form 3004 and attach to Form 3004 for OC review. If the SRI is prepared for an FOI Assessment, then the "Completion Items" section of this form is not required. FOI required completion items ***SHALL*** be addressed in the FOI Assessment.
4. The SRI description and safety evaluation ***SHALL*** be prepared and attached to Form 3004 for OC review. Content PER 4 AWI-05.01.09, USE MPBB SAFETY-EVAL-FORMAT in the [QUALITY PERFORMANCE] folder.
5. Include the reason why an SRI is required in the safety evaluation.
6. Prepare a Safety Evaluation Summary by completing Form 3473 and processing per 4 AWI-02.07.01 (USAR CONTROL).
7. A completion summary describing completion, resolution or performance of items covered by the SRI ***SHALL*** be attached to Form 3004 along with all other required documents when submitted to the record file.

Completion Items

	Required Yes	Required No	Completed Date
1. Specification Change (4 AWI-02.06.01)		X	
2. Technical Specification Change (4 AWI-02.07.01)		X	
3. Drawing Change (4 AWI-02.04.01)		X	
4. Operations Manual Change (4 AWI-02.02.06/02.02.02)	X		
5. New Procedure or Procedure Changes		X	
6. Technical Manual Change (4 AWI-02.05.0X Series)		X	
7. Training	X		
8. USAR Change (4 AWI-02.07.01)		X	
9. Control of Temporary Installations (4 AWI-04.04.03)		X	
10. Safety Evaluation Summary to NSS, complete Form 3473	X		

Summary of Operations Committee Review (Include any follow-up actions from OC meeting):

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INTRODUCTION:

The in-vessel visual inspection performed during the 1993 refueling outage identified a small crack indication on the 'B' core spray header where the piping and the tee box meet as shown in figure 1. The function of the junction box and the attached distribution pipes is to direct core spray flow from the vessel inlet nozzle through the annulus area to the spargers inside the shroud. The integrity of this safety related piping system must be maintained to ensure that core spray flow can be delivered to the core to achieve its design function. The purpose for this SRI is to evaluate the acceptability of operating with this crack through one fuel cycle with no operational changes or restrictions.

DESCRIPTION:

The crack is located on the left arm of the 'B' core spray distribution pipe adjacent to the junction box weld. It is concluded that the crack is most likely caused by intergrannular stress corrosion cracking (IGSCC) for the following reasons:

1. The pipe material is Type 304, stainless steel.

2. The crack is in the heat affected zone of the junction box weld.

3. There is the presence of welding residual stress and bending stress.

4. The coolant in this area is highly oxidizing and hydrogen water chemistry is not effective in this region.

5. The characteristics of the crack is similar to that typically demonstrated by IGSCC.

6. Other BWR facilities have experienced similar crack indications in the same area. (IEB 80-13).

The crack is located at the 3:00 position (viewing the pipe axially towards the tee box) about 1/4" from the weld. Initial estimates of crack length based on video camera surveys was about 3". A later measurement with a reference scale placed at the crack location indicated the circumferential crack length was actually closer to 2 1/4". In addition, an underwater UT examination of the crack area using NSP's approved method for IGSCC detection revealed that the crack was through wall, did not appear to have major ID origin and no ID or OD cracking was evident at the 12:00 and 6:00 positions.

EVALUATION

An evaluation of the significance of the core spray pipe crack is provided for the structural integrity of the pipe, the crack leakage estimate, the lost parts analysis, and the failure detection mechanism (See attachments 1 & 2).

1.) Structural Integrity

In order to determine the integrity of the core spray line with the crack, a crack arrest evaluation was performed. The stresses due to pipe restraint and the fabrication residual stresses were included in the evaluation. Because the applied normal loading is predominantly displacement controlled, the stresses relax as the crack grows and the flexibility of the pipe increases. The results of the analysis showed that when the crack reaches 180^0 of the circumference, the flexibility is increased sufficiently to relieve almost all of the displacement controlled stresses. Consequently, the crack growth is expected to be negligible or at virtual arrest prior to reaching 180^0.

Because of the predominant secondary stresses, the crack can be expected to arrest prior to reaching 180^0. An assessment was made to determine the critical flaw size of the core spray pipe by treating stresses associated with the design loadings as primary stresses and performing a net section collapse evaluation. The loadings considered in this evaluation are downcomer flow impingement loads, seismic, pressure, weight, and thermally induced loads. Cyclic stresses from recirc pump vane passing frequencies occur outside of the expected natural frequencies of the core spray header pipe and therefore need not be considered. The results of this evaluation confirm that a through wall crack of up to 240^0 around the circumference would not cause pipe failure. It is concluded that the structural integrity of the piping with a crack will be maintained for all conditions of operation for the next operating cycle.

Additional stresses to the core spray piping from a water hammer event are very unlikely due to the various water hammer prevention mechanisms employed. The piping between the core spray pump discharge check valve and the isolation valve is maintained full of water and pressurized to 30 psig or greater by the condensate service water system. Core Spray system operability is contingent on this keep fill system being in service when core spray is in standby. The internal core spray piping system is also kept full of water during normal operation and the non-condensibles are vented out through the high point vent hole in the junction box just inside the reactor pressure vessel. Even if a water hammer event were to occur, the resulting additional stress to the piping has been previously evaluated and deemed insignificant.

2.) Crack Leakage Estimate

There are no direct measurements of leakage from the crack during the operation of the core spray system except for gross failure detections. However, from previous analyses and tests performed for the cracks observed in other BWR's, it is possible to establish an upper bound leakage for the crack identified at the Monticello Plant.

A crack growth evaluation was performed and it was estimated that the current crack size is expected to grow less than 1.2 inches during the next 18 month cycle. The total crack length at the end of the next cycle is estimated to be close to 100^0 of pipe circumference.

We conservatively estimated the maximum leakage expected through the crack, the configuration for a 180^0 throughwall crack. This configuration was considered to be the upper bound based on the crack arrest results. A crack width of 0.01 inch was conservatively assumed based on the results of Linear Elastic Fracture Mechanics (LEFM) methods which showed the crack opening to be <0.01 inch under the applied loads. Using the aforementioned assumptions, the leakage was determined to be 24 gpm.

3.) Best Estimate LOCA Analysis

Even though the structural integrity has been analyzed to be adequate, a best estimate LOCA analysis has been performed to determine the effects of a complete severance of the core spray piping. This analysis showed that if a complete severance of the B core spray system piping in the location of the present crack were to occur, and zero flow from the B core spray system was assumed to reach the internal shroud core spray sparger, no significant fuel damage will occur even with the worst case limiting single failure postulated, ie., LPCI loop injection valve failure.

4.) Lost Parts Analysis

Based on the structural analysis, the core spray pipe will not break and consequently, will not result in loose pieces in the reactor. However, an evaluation of the possible consequences of a potential loose piece was performed.

Two different types of loose pieces are postulated:

1.) A section of core spray pipe

2.) A small piece of the core spray pipe.

Since the core spray pipe with the crack is in the annular region of the reactor pressure vessel, this evaluation assumes that any potential large loose section generated from the core spray pipe will sink into the downcomer region and remain in the idle flow area on the shroud shelf.

For a loose small piece to reach and potentially block the inlet of a fuel assembly, it would have to be carried into the lower plenum. To accomplish this, it would have to be carried by the recirculation flow through the jet pump nozzle into the lower plenum, then make a 180^0 turn and be carried upward to the fuel assembly inlet orifices. However, the flow blockage would be much less than that required to initiate critical boiling transition in the bundle. Multiple pieces migrating to the same bundle may result in critical flow blockage, but the probability for such an occurrence is extremely low. It is also possible that such a piece could cause fretting on individual fuel pins which could result in failure of the fuel rod. In either case, these failure mechanisms are slow to develop and are easily detected through offgas monitoring so that early mitigation is easily achievable.

5.) Failure Detection Mechanism

Plant instrumentation is provided to initiate an annunciator in the event of a break in the core spray piping internal to the RPV and outside the shroud which includes the area of the subject crack. This alarm circuit has specific surveillance requirements in the technical specifications to ensure operability. The instrumentation is connected to measure the differential pressure between the area inside the shroud above the core plate and the core spray line inside the vessel. If the core spray line is intact, these two areas will be at the same pressure. If the line is not intact, the instrumentation will sense the differential pressure across the core shroud and a control room alarm will be initiated. Plant procedures are in place to provide direction to operations personnel in the event this alarm is received. These procedures direct the operators to begin an orderly plant shutdown. Since the operating time with a potentially broken core spray piping header is limited, the potential for damage to other reactor internals is minimized. In addition, operations will be given augmented training on this subject and how it could relate to the alarm condition in the control room prior to startup from this refueling outage. These procedures ensure orderly plant shutdown

Conclusion

I. Can the proposed activity increase the consequences of an accident previously evaluated in the USAR or in a pending USAR submittal?

As noted, any flow lost through the crack will not affect core spray performance with respect to Licensing Basis LOCA Analysis. Therefore, the radiological consequences of an accident described in chapter 14 of the USAR are not changed.

II. Can the proposed activity increase the probability of occurrence of an accident previously evaluated in the USAR?

The existence of the core spray header crack does not increase the probability of an accident previously evaluated in the USAR because the core spray header crack is located inside the reactor pressure vessel and cannot cause a loss of reactor vessel coolant.

III. Can the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR or in a pending USAR submittal?

The analysis verifies the structural integrity of the core spray header is maintained so that it is able to fulfill its function during a LOCA. Therefore, this core spray pipe crack does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR (Reference 1).

IV. Can the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR or in a pending USAR submittal?

The analysis verifies the structural integrity of the core spray header so that it is able to fulfill its safety function during a LOCA. The existence of the core spray header crack does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR because the crack is located on the core spray header inside the reactor pressure vessel and cannot cause a loss of reactor vessel inventory.

V. Can the proposed activity create the possibility of an accident of a different type than any evaluated previously in the USAR or pending USAR submittal?

The core spray header crack does not significantly change the form, fit or function of the core spray header. Because continued core spray header structural integrity is demonstrated, lost parts (loose pieces) are not expected. Therefore, the existence of the core spray header crack does not create the possibility of an accident of a different type than previously evaluated in the USAR.

VI. Can the proposed activity create the possibility for malfunction of equipment important to safety of a different type than any previously evaluated in the USAR or in a pending USAR submittal?

The core spray header crack does not significantly change the form, fit or function of the core spray header and is subject to the same conditions and scenarios as the existing core spray header. Because continued core spray header structural integrity is demonstrated, lost parts (loose pieces) are not expected. Therefore, the existence of the crack does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

VII. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specifications license amendment request submitted February 12, 1993, requires that the core spray pumps provide a minimum flow rate of 2800 gpm. This flow rate includes 100 gpm to account for leakage in addition to the flow rate of 2700 gpm assumed by the SAFER/GESTR-LOCA analysis. The summation of all sources of leakage, including the conservative estimate of core spray crack leakage, is less than the 100 gpm leakage allowance. Therefore, the margin of safety as required by the Technical Specification is not reduced. In addition, the pumps are actually tested for a minimum flow rate of 3020 gpm each, providing additional margin.

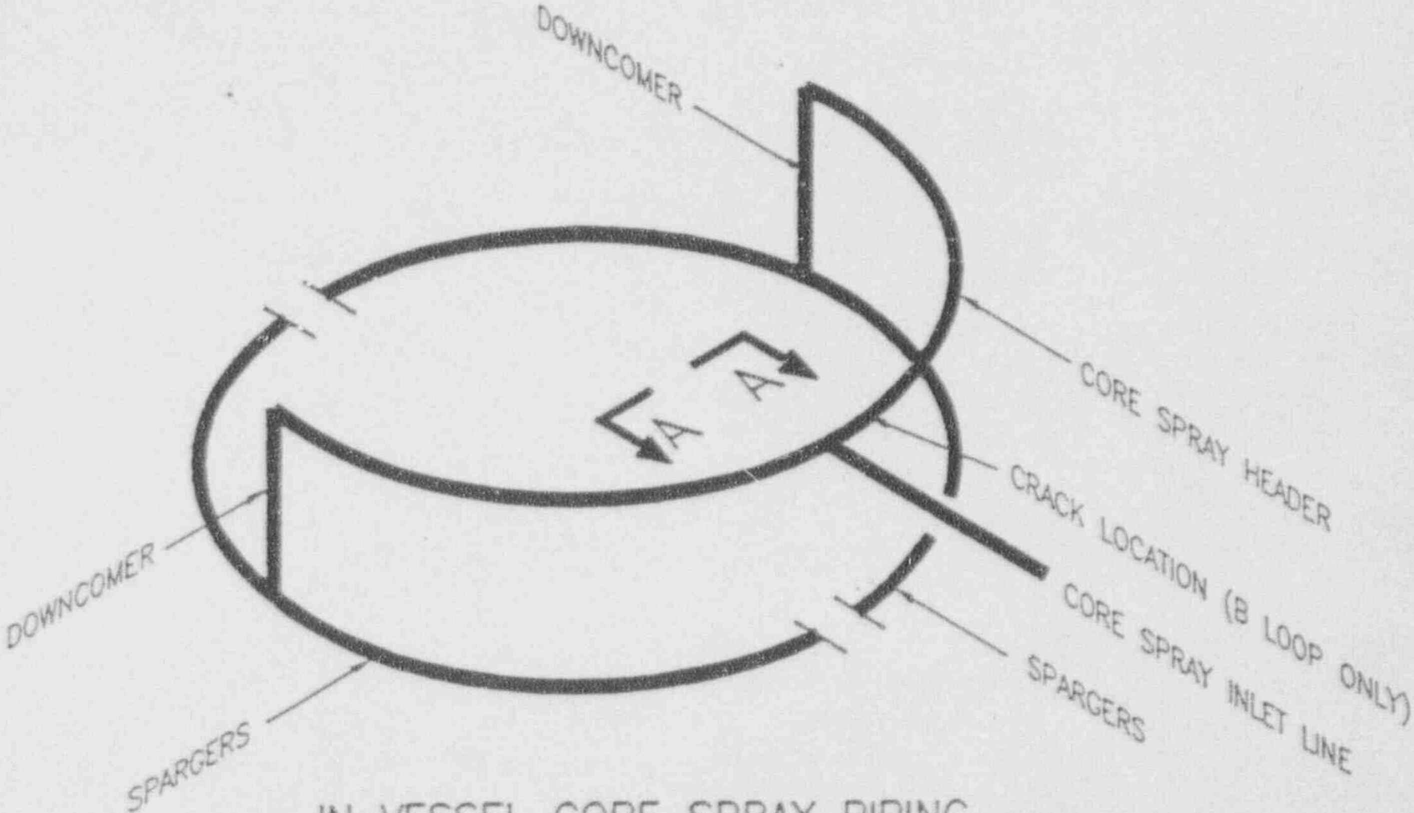
SUMMARY

Based on the aforementioned analysis, it is concluded that Monticello can safely operate in this condition during the next fuel cycle, and that no operational changes or restrictions are required during that period. A third party independent review has been conducted to substantiate that the technical approach and methodology utilized in this analysis are sound. In addition, previous BWR industry experience has demonstrated that operating multiple cycles results in crack growth well within the assumptions of this evaluation. Even in the unlikely event that the pipe severs, best estimate LOCA analysis demonstrates that adequate core cooling is assured assuming the most severe single failure. It is concluded that continued operation with the current core spray line crack for one additional cycle does not constitute an unreviewed safety question or a significant safety hazard.

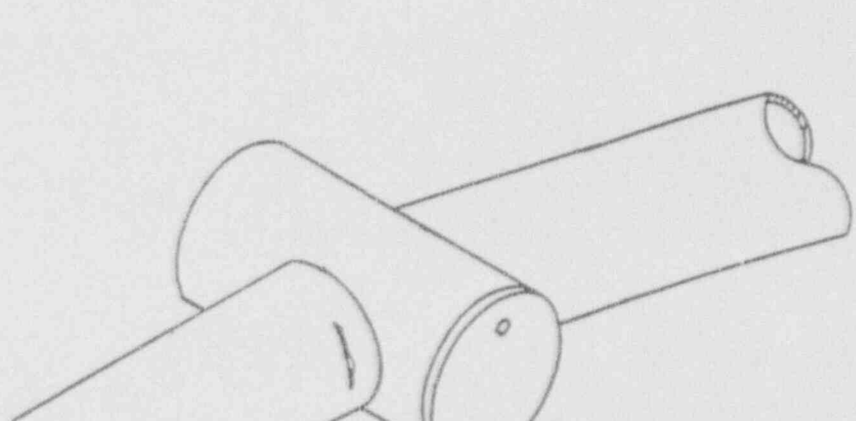
Attachments:

(1) GE-NE-637-0005-0393 Core Spray Crack Analysis for Monticello Nuclear Generating Plant, March 1993 General Electric Company

(2) AJG-93-015/PCR-93-032 Third Party Review of General Electric Evaluation Approach for Continued Operation of Cracked Core Spray Pipe at Monticello Nuclear Generating Plant by Structural Integrity Associates, INC.



IN VESSEL CORE SPRAY PIPING



SECTION A-A
DETAIL OF CRACK IN
CORE SPRAY PIPING

NOTE:
CRACK ON B LOOP
ONLY AT 90° (N-5A)

FIGURE #1
E# 930170
MONTI CAD M00585