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AUTH. NAME AUTHOR AFFILIATION
MARCHI, M.L. Wisconsin Public Service Corp.
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SUBJECT: Forwards addl info re review of relief request RR2-1 to allow continued plant operation w/two pin hole leaks in 3/4 inch ASME Code Class 2 chemical injection weldment per 960814 & 15 verbal requests.

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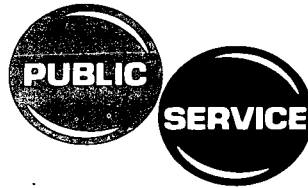
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WISCONSIN PUBLIC SERVICE CORPORATION

10 CFR 50.55a

August 15, 1996

U.S. Nuclear Regulatory Commission
 Document Control Desk
 Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
 Operating License DPR-43
 Kewaunee Nuclear Power Plant
Relief Request No. RR2-1 Request for Additional Information

Reference: 1) Letter from ML Marchi (WPS) to Document Control Desk (NRC)
 dated August 8, 1996
 2) Letter from ML Marchi (WPS) to Document Control Desk (NRC)
 dated August 12, 1996

This transmittal provides additional information in response to the staff's verbal request on August 14 and 15 regarding their review of Relief Request RR2-1 to Allow Continued Plant Operation with Two Pin Hole Leaks in a 3/4 inch ASME Code Class 2 Chemical Injection Weldment.

Should you or any member of the staff require additional information feel free to contact me or any member of my staff.

9608280098 960815
 PDR ADOCK 05000305
 P PDR

Sincerely,

C. A. Schwab for

M. L. Marchi
 Manager - Nuclear Business Group

CAS
 Attach. 270110
 cc - US NRC - Region III
 US NRC Senior Resident Inspector
 Mr. Lanny Smith, PSCW

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ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

August 15, 1996

Additional Information Relief Request RR2-1

Introduction

The following information is provided in response to the staff's request for additional information relative to the Relief Request RR2-1.

Flaw Characterization and Analysis

In order to strengthen its analysis of the flaw, WPS took the following actions following discussions with the staff on the afternoon of August 14, 1996:

- 1) An investigation of further NDE techniques to better characterize the flaw.
- 2) Additional review and assessment of the weld.
- 3) A review of the Fracture Mechanics analysis, and
- 4) An independent assessment of the information provided to the staff regarding RR2-1.

NDE Techniques

We have contacted vendors with regard to available techniques to further characterize the weld flaw. None were able to provide assurance that techniques in use today could provide conclusive volumetric examination results.

EPRI was an additional contact pursued. EPRI has an Eddy Current Technique that may be able to detect flaws 0.125" a maximum below the OD a surface of a Full Penetration Fillet Weld. However, the technique has only been qualified on Stainless Steel Socket Welds, Carbon Steel Socket Welds and Dissimilar Metal Socket Welds. No procedure regarding this technique is available at this time nor has this technique been demonstrated at high temperatures.

EPRI has also made a mockup for Ohio Edison of a similar Socket. They are checking to see if Ohio Edison qualified this technique and procedure and if their technique, if available, can be used at 316 degree F. for carbon steel.

EPRI provided some of their research papers on the work they have been doing for examinations of piping geometries similar to our configuration. Their conclusion is that this type of exam is feasible, but will require special techniques, and the examination methods need to be developed and demonstrated on mockups.

Although additional ultrasonic testing may be performed, it would not be able to provide credible evidence that no flaws beyond what we have already characterized exist. However, we will continue to assess the value of further UT.

Additional discussions with our NDE vendor reaffirmed our original conclusions. These discussions included possible pursuit of other examination methods, but none were identified.

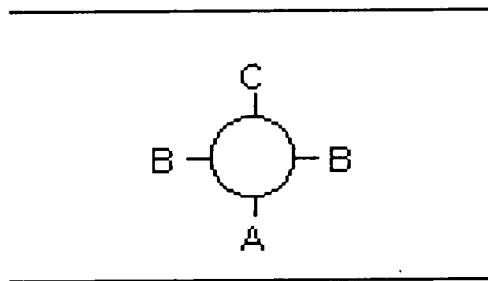
Additional Review and Assessment of the Weld

The weld fabrication and inspection records were examined to assist in identification of the likely cause of the leak.

The original installation utilized standard proven power house fabrication techniques. Installation of a sockolet to 16" OD FW piping was performed using E6010 electrode with the SMAW process for the root pass. E7018 electrode was used for the filler passes to a completed thickness of approximately 5/8".

A review of the vendors weld procedure confirmed that the weld in question was performed essentially as follows:

1. Gap the sockolet as required by the Welding Procedure Specifications, align the fitting and tack weld at a minimum of 2 places.
2. The tack welds are cleaned of slag and the ends visually inspected by the welder for soundness.
3. The root pass is usually started at point "A" with progression to points "B", both sides of the joint.
4. The welder then moves to the opposite side of the 16" pipe and continues the pass from points "B" to point "C".



5. This process sequence is continued until completion of the weld.
6. The final assembly was post weld heat treated.

As can be noted, all starts and stops (root, intermediate, and final weld passes) are typically in the same area of the weldment. This is due to the configuration of the socket relative to the main run. The included joint angle at point "B" is minimal with the curvature of the main run causing the bevel angle to be tighter at point "B".

At the completion of the weld the Liquid Penetrant Examination process was used. Our records for this weld show that there were no indications on the surface of this weld. The welding procedure required no root pass or intermediate pass non-destructive examinations for this weld joint.

Cluster porosity was identified by visual examination of the leak. The pin hole indications appear in the area of the starting or stopping point of the final weld pass at a location consistent with the weld procedure described above. All previous weld passes would have started and/or stopped at the A, B and C locations noted above which could give rise to porosity or slag inclusion at these locations. Insufficient cleaning would have resulted in covering slag or porosity in the subsequent weld passes. Porosity may be small enough to go unnoticed by the welder subsequent to the next pass of weld. A review of the fabrication traveler leaves the number of weld passes indeterminate. The Weld Procedure Specification allowed as few as two (2) passes and as many as ten (10) passes dependent upon the pipe wall thickness.

Although porosity can occur anywhere in the weld in any of the weld passes, it is most likely to occur and result from improper initiation or termination of the welding arc.

The postulated cause of the leak is an alignment of cluster porosity in the final pass and subsurface weld flaws at the start/stop point. The subsurface weld flaws are postulated to be slag inclusions based on the above discussion of fabrication techniques and flaw characterization.

Assuming this is the case, we may postulate fracture of the inclusion over time as a result of heating and cooling of the line. Because of the inherent toughness of the weld and base material at our operating temperatures, the fracture will likely be limited to the inclusion. This is a key point and is made in the Westinghouse Report "Evaluation of Circumferential Flaw Limit load conditions: Kewaunee Feedwater Line", previously submitted (Reference 2). The material fails in a limit mode, that is, it yields, and does not fracture in a brittle fashion. Additional information to support this has been provided by Westinghouse based on Section XI Appendix J(1).

Calculated stress levels on the weldment with the pin hole shows stress levels to be very low. The calculated stress considering loading from WPSC piping analysis show the stress as approximately 1 ksi. This is very small in comparison to allowable stresses.

This is further supported by the fact that the leak has been closely monitored since its discovery on August 6, 1996, and has exhibited the expected stable behavior.

This additional information on the construction on the weldment, the material toughness, and the low stress levels on this weld, further supports our judgement that the leak is not due to a fracture, but rather weld porosity and is stable.

Review of Fracture Mechanics Analysis

In order to further address the NRC's questions regarding the assumptions of the fracture mechanics analysis, WPSC and Westinghouse reviewed the analysis which was provided to the staff as Attachment 2 to reference 2.

The results of this review are:

The analysis assumes a conservative circumferential orientation of the flaw. The flaw dimensions were based on the information derived from the radiograph. While these assumed dimensions bound the flaw size interpreted from the NDE methods, it is recognized that the NDE techniques have limitations which could result in the existence of an undetected flaw of different dimensions or orientation. However, the likelihood of such a flaw is considered to be very low given the information presented above.

Westinghouse also confirmed that the assumptions regarding stresses on the weldment used in the analysis conservatively bounded the calculated stress levels provided by WPSC today, August 15, 1996.

Given the low likelihood of an undetected flaw, the conservative assumptions regarding flaw orientation, and the bounding assumptions used regarding stresses on the weldment, WPS has concluded that the analysis provides a conservative assessment of the structural adequacy of the weldment. Westinghouse will be performing further analysis using the stress levels provided by WPSC, in place of the bounding values used previously.

Independent Assessment

Recognizing the importance of this evaluation, and in an effort to demonstrate that WPSC's evaluation is appropriate to assure the safe operation of KNPP, WPSC contracted an independent expert to review the information provided to the staff. Teledyne Brown Engineering performed this review and concluded the following (see Attachment 2):

The fracture mechanics analysis-evaluation performed by WEC (Westinghouse Electric Corporation) is judged to be conservative and to bound the worst case conditions.

Little or no flaw extension is expected prior to the September outage.

A logical rationale has been applied to assess the cause of the leakage and the likely source of weld degradation.

TBE also agrees that the restraint device will provide assurance that the socket connection will not fail catastrophically.

Summary of Flaw Characterization and Analysis

In summary, WPSC has provided additional information and assessments with respect to characterization of the flaw. This information, combined with that previously submitted, provides further assurance that KNPP can be safely operated with the pin-hole leaks described in RR2-1 until its September outage. The daily inspections of the leak since it was first found on Tuesday, August 6, 1996, have shown that the leak is stable, thus providing further supporting data for this conclusion.

Additional Followup Information

Notwithstanding the above, WPSC has taken additional steps to ensure safe operation by installation of a restraining device on the chemical injection line and by performing an evaluation of the consequences of complete failure of the weldment. Additional information regarding these steps is provided below.

Restraining Device

The restraining device is installed to limit the leakage that would result in the unlikely event of a complete failure of the weldment. The following provides the additional information requested by the staff on August 15, 1996, regarding this device.

The design considerations for the restraint are contained in Chapters 10 and Appendix B of the USAR. Chapter 10 contains the discussion of High Energy Line Breaks outside of Containment. The piping systems discussed in Chapter 10 include the Main Feedwater system piping outside of containment.

The relevant portion of Chapter 10 of the USAR for pipe restraints states the restraint loads require consideration of the resulting load from the pipe rupture. The design considered the weldment failure load to be 3,313 lbf. The other restraint loading combinations are those specified for type 1 structures in Appendix B of the USAR. Chapter 10 of the USAR identifies restraints as type 1 structures. The Appendix B loading combinations for class 1 structures are Normal Operating Conditions, Normal and Operational Basis Earthquake Conditions, and Normal and Design Basis Earthquake Conditions. These loading combinations require the appropriate combinations of dead, live, environmental, seismic and Design Basis Accident loads. The Design Basis Accident Loads are defined as those resulting from the double ended rupture of the RCS cold leg. There is no mention of Rupture loads being combined in any of these loading combinations. If the temporary restraint was considered a pipe support, the USAR requires consideration of Normal and Pipe Rupture loads combined, and Normal and Design Basis Earthquake and Pipe rupture loads combined. In the development of the design, the temporary restraint was not considered a pipe support.

The design calculations were reviewed in light of the concern that the restraint loading should consider Earthquake loads in combination with the weldment failure load. Therefore, the seismic loads were investigated.

From the stress report, KEW-MS-7 Part 44, the reactions from the 3/4" Chemical Injection pipe on the sockolet are listed in the table below. These values are the maximum absolute values from the results of all the loading combinations.

F_x	11 lbf	M_x	29 ft-lbf
F_y	24 lbf	M_y	29 ft-lbf
F_z	10 lbf	M_z	22 ft-lbf

From the Main Feedwater line stress report, FW-005-003 APX, the DBE accelerations at valve FW-12A are $g_{horizontal}=0.27$ and $g_{vertical}=0.004$. The horizontal acceleration value is the square root sum of the squares of the horizontal accelerations listed in the stress report. The valve is the nearest point on the piping system with acceleration values listed in the stress report. The valve is 2'-5.25" away from the sockolet connection on the Main Feedwater pipe. Using these accelerations, the seismic load on the restraint due to acceleration of its weight is 16.2 lbf horizontal and 0.24 lbf vertical.

A detailed calculation has not been performed because the magnitude of the additional seismic forces were found to be small in comparison to the load carrying capacity of the components.

A further basis for determining the conservatism and adequacy of the restraint design is consideration of the USAR allowable stresses for the original loading combination and revised loading combinations. The USAR stress limits on restraint materials, during the transient conditions or rupture event is 0.9 times the yield stress, except for peak stresses of short duration in the initial stage of pressurization or loading which has a limit of 1.5 times 0.9 times the yield strength of the material. Using the lower value, this results in a stress limit of 32.4 ksi. The design analysis used a stress limit of 21.6 ksi and the calculated stresses in the design are lower than 21.6 ksi.

If considered a support, the Normal and Pipe rupture load combinations or the Normal and Design Basis Earthquake and Pipe rupture load combinations are considered Faulted Loading Conditions in the USAR. The stress limits for faulted loading conditions are defined as the minimum specified yield strength of the material or 1.8 times the allowable stress, whichever is higher.

In summary, the restraint design calculations did not combine seismic and rupture loads in the original calculations. However, this review confirmed that the restraint meets USAR requirements whether it is considered a restraint or a pipe support. As defined in chapter 10 and Appendix B of the USAR combining the rupture loads and seismic loads is not required for these type of restraints. If the restraint were considered a pipe support, and the seismic forces are combined with the weldment failure loads, the restraint is adequate. Use of the higher allowable stresses, per the USAR, not used in the original design or analysis, further demonstrates the adequacy of the design.

Consequences of Weldment Failure

As noted above, our analysis has shown that weld failure is extremely unlikely. Furthermore, the resultant leakage of such a failure will be limited by the restraining device noted herein. None the less WPSC has evaluated the operational and safety consequences if a complete separation of the 3/4" sockolet.

The Justification for Continued Operation (JCO) submitted on August 8, 1996 provided an evaluation of the potential consequences of the complete failure of the 3/4 inch chemical injection line. The following restates the evaluation and provides additional information where appropriate:

1) Effect on plant equipment

USAR Chapter 10 analysis of feedwater pipe breaks conservatively bounds the potential failure of a 3/4 inch branch connection to the main feedwater piping. Circumferential breaks in branch runs greater than 1" were considered with the break area equivalent to the pipe diameter.

The feedline break analysis reviewed all needed equipment to handle the transient and its location relative to the feedwater piping location. The needed equipment provides for reactor trip, capability to maintain hot shutdown after the break, and ultimately achieve cold shutdown. All needed equipment was shown to be protected from adverse environment, or qualified to operate in conditions which meet or exceed those from the break. An EQ walkdown was performed of the area by the "A" Feedwater Isolation valves (Compartment CI) on August 15, 1996. The following EQ equipment is located in this compartment:

- 33074 - FW to S/G A Isolation Valve
- 33075 - FW to S/G A Isolation Valve
- 33080 - FW to S/G A Bypass IA1 Valve
- 33081 - FW to S/G A Bypass IA2 Valve
- 32015 - FW to S/G A Isolation Valve
- 3102701 - FW Reg Valve Limit Switch
- 3102702 - FW Reg Valve Limit Switch
- 3115701 - FW Bypass Valve Limit Switch
- 3115702 - FW Bypass Valve Limit Switch

This equipment is EQ Type H1 and meets all requirements of the EQ Rule.

The feedwater lines are Class 1 from the steam generators back to their respective isolation valves. A failure of any Class I feedwater line or malfunction of a valve installed therein will not impair the reliability of the Auxiliary Feedwater System, render inoperative any engineered safety feature, initiate a loss of coolant condition, or cause failure of any other steam or feedwater line. The consequences on plant equipment are therefore shown not to present any unanalyzed challenges to plant safety.

2) Effects on reactor safety

As supporting information to the qualitative assessment provided in the JCO, the engineering staff simulated the expected system behavior resulting from a complete failure of the weld in question. This analysis was performed with a plant specific RETRAN computer model. The feedwater control system was assumed to be in manual to estimate the minimum time to automatic reactor trip. The system responded as expected. The transient progressed slowly with a decrease in feedwater flow to the steam generator with the faulted feedline resulting in a low steam generator level trip eleven minutes after pipe break initiation.

A similar transient was examined using the KNPP plant specific simulator to assess the indications available to the operators to promptly diagnose the event. Three scenarios were run to bound the plant response to parameters not readily quantified: exact additional feedflow capability remaining, and progression time of the failure.

Case 1 - The simulator was programmed for a ramped in failure of the feedline over a period of 5 minutes and the feedwater control system allowed to respond with an increase of 5% (with the actual plant derated to 98% power this is estimated to bound the maximum response of the control system)

Case 2 - The simulator was programmed for a ramped failure of the feedline over a five minute period and the feedwater control system allowed to respond with an increase of 2% (minimum estimated response)

Case 3 - The simulator was programmed for an instantaneous failure of the feedline with the control system allowed to respond with an increase of 2%.

In each case annunciation related to the event occurred within three minutes of break initiation. The alarms received included Steam Generator Tilts, S/G Bypass CV Level Deviation, Reactor Thermal Power High, S/G Program Level Deviation, and S/G Level Low. These alarms actuated in various combinations and sequences depending on the scenario but in each case adequate information was available for operator diagnosis. The steam exclusion temperature indicators showed elevated temperatures in the area of the break but did not reach actuation levels in the simulator. Based on the indications observed, and the reinforcement provided during licensed operator requal training, the operators are expected to diagnose the event and take conservative actions to trip the unit under these conditions. This report will be provided to the crews to heighten their awareness of expected plant response.

Upon a reactor trip, feedwater is isolated and the event is terminated. Auxiliary feedwater will supply water to the Steam Generators and the plant will respond as it would to any normal reactor trip since main feedwater is no longer required.

In summary, the plant response to the postulated event was confirmed by both the engineering computer model and the dynamic simulator scenarios. The simulated plant response was as expected and demonstrates that ample indication is available for operators to promptly diagnose the event and adequate time is available for the operators to take actions to manually trip the unit. The failure of a 3/4" branchline from the main feedwater header will not result in a significant challenge to reactor safety.

3) Effects on USAR Safety Analysis

The USAR Chapter 14 safety analysis addresses the loss of normal feedwater in section 14.1.10. The unit is designed to withstand a complete loss of main feedwater from full power conditions under a number of conservative assumptions. The complete loss of normal feedwater does not result in any adverse conditions in the core. The analysis shows that with auxiliary feedwater flow being supplied to only one steam generator, the steam generator tube sheet remains covered and therefore heat removal capability is maintained. This assures that the residual heat is removed from the reactor core and that RCS water inventory will not be lost through the pressurizer safeties or relief valves. Since, as discussed in paragraph 1 above, the complete failure on this branch line does not alter assumptions used in the accident analysis, the conclusions of the USAR bounding analysis remain valid.

ATTACHMENT 2

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

August 15, 1996

Letter from Teledyne Brown Engineering to WPSC

August 15, 1996
96-58

Mr. Charles A. Schrock
Wisconsin Public Service Corporation
Plant Manager
Kewaunee Nuclear Power Plant
North 490, Highway 42
Kewaunee, WI 54216-9410

Subject: Two Pin Hole Leaks in a 3/4" ASME Code Class 2 Chemical Injection Weldment

References: 1. WPSC Letter to U.S. NRC, No. NRC-96-076, dated August 8, 1996.
2. WPSC Letter to U.S. NRC, No. NRC-96-080, dated August 12, 1996.

Dear Mr. Schrock:

Teledyne Brown Engineering - Engineering Services (TBE) has reviewed the two referenced Wisconsin Public Service Corporation (WPSC) letters with attachments. These documents were submitted to the NRC to provide justification to continue short term operation of the Kewaunee Nuclear Power Plant with the two pin hole leaks found in the weldment of the chemical injection sockolet to the 1A main feedwater line.

Based on telephone conversations with WPSC and the review of the referenced letters, TBE concurs with the conclusions drawn by WPSC and its consultant, Westinghouse Electric Corporation - Energy Systems (WEC) regarding the following:

The fracture mechanics analysis-evaluation performed by WEC is judged to be conservative and to bound the worst case conditions.

Little or no flaw extension is expected prior to the September outage.

A logical rationale has been applied to assess the cause of the leakage and the likely source of weld degradation.

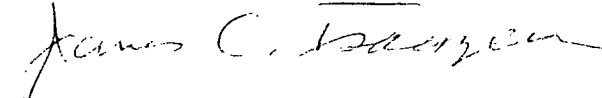
TBE also agrees that the restraint device will provide assurance that the socket connection will not fail catastrophically.

Charles A. Schrock
Wisconsin Public Service Corporation
96-58
August 15, 1996
Page 2

If you require further assistance, do not hesitate to contact TBE.

Very truly yours,

**TELEDYNE BROWN ENGINEERING
Engineering Services**



James C. Tsacoyeanes, P.E.
Consulting Engineer

JCT/cac

cc: D. F. Landers, TBE
R. A. Enos, TBE