CP S1

December 10, 1982

Docket No. 50-263

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

On October 19, 1982, the Commission issued a Confirmatory Action Letter which requested information on the cracks you found on the recirculation system piping at the Monticello Nuclear Generating Plant. Specifically, the Confirmatory Action Letter requested that you submit, to the Commission, the results of your inspection, your corrective actions, justification to return to power, and that you receive NRC concurrence before returning the unit to power.

In a November 22, 1982 letter, as supplemented December 3, 1982, you submitted the information requested by the Confirmatory Action Letter and also included in this letter, your response to IE Bulletin 82-03, Revision 1.

Our review included an evaluation of your submittals, review of your repair procedures and discussions with members of your staff. Based on this review, we conclude that you have satisfied the conditions of the Confirmatory Action Letter and are, hereby authorized to return Monticello to power, subject to the conditions of the enclosed license amendment.

The Commission has issued the enclosed Amendment No.   to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment (1) approves the repair to the recirculation system piping and authorizes resumption of power, subject to certain conditions, and (2) changes the Technical Specifications to revise the Limiting Conditions for Operation and Surveillance Requirements for the coolant leak detection system.

The Technical Specification change is in response to your July 6, 1981 application, as supplemented by subsequent discussions between the NRC and your staff. Other proposed changes, as requested in your July 6, 1981 application, are still under staff review and will be addressed in a future action.

OFFICE							
SURNAME							
DATE							

Mr. D. M. Musolf

December 10, 1982

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY
Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Amendment No. to DPR-22
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures
See next page

Distribution
Docket File
NRC PDR
Local PDR
ORB#2 Rdg.
D. Eisenhut
S. Norris
H. Nicolaras
OELD
SECY
L. J. Harmon 2
T. Barnhart 4
L. Schneider
D. Brinkman
ACRS 10
OPA Clare Miles
R. Diggs
NSIC
Gray
ASLAB
5 extra

F.R. Notice to Amendment

* Previous concurrence sheet concurred on by:

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD	DL:D	
SURNAME	S.Norris	H.Nicolaras*	D.Vassallo*	G.Lainas *	R.Bachmann*	D.Eisenhut	
DATE	12/10/82	12/9/82	12/9/82	12/9/82	12/9/82	12/10/82	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

December 10, 1982

Mr. D. M. Musolf

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. to DPR-22
2. Safety Evaluation
3. Notice of Issuance

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F.R. NOTICE
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AMENDMENT

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	S. Norris	H. Nicolaras	D. Vassallo	G. Lainas	R. Bachmann		
DATE	12/ /82	12/9/82	12/9/82	12/ /82	12/9/82		

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Mr. D. M. Musolf
Northern States Power Company

cc:

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Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
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Monticello Nuclear Generating Plant
Northern States Power Company
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Glen Ellyn, IL 60137

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated July 6, 1981 and supplemented by letters dated November 22, and December 3, 1982 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-22 is hereby amended by adding paragraph 2.C.7 to read as follows:

   7. Repairs to the Recirculation System Piping

      The repairs to the recirculation system piping are approved and the unit is hereby authorized to return to power operation, subject to the following condition:

Prior to the startup of Cycle 11, the licensee shall submit by August 1, 1983 for the Commission's review and approval, a program for inspection and/or modification of the recirculation system piping.

The license is further amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.2 is hereby amended to read as follows:

2. Technical Specifications

   The Technical Specifications contained in Appendices A and B as revised through Amendment No. 14 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 10, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Remove the following pages and insert identically numbered pages:

5
126
150

Y. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

AB. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in the reactor coolant system pressure boundary.

AC. Identified Leakage - Identified leakage shall be:

1) Reactor coolant leakage into drywell collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

2) Reactor coolant leakage into the drywell atmosphere from sources which are specifically located and known not to be Pressure Boundary Leakage or which do not significantly impair the methods used to detect reactor coolant leakage.

AD. Unidentified Leakage - Unidentified leakage shall be all reactor coolant leakage which is not Identified Leakage.

3.0 LIMITING CONDITIONS FOR OPERATION

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, based on sump monitoring, shall be limited to:

   a. 5 gpm Unidentified Leakage
   b. 2 gpm increase in Unidentified Leakage within any 4 hour period
   c. 20 gpm Identified Leakage
   d. no pressure boundary leakage

2. With reactor coolant system leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, identify the source of increased leakage within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

4. If any Pressure Boundary Leakage is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

5. At least one of the leakage measurement instruments associated with each sump shall be operable and the drywell particulate radioactivity monitoring system shall be operable or a sample of the containment atmosphere shall be taken [illegible] least eve[illegible] four hours. Otherorde[illegible] he reac- to

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out:

   a. Unidentified and Identified Leakage rates shall be recorded at least once every 4 hours using primary containment floor and equipment drain sump monitoring equipment.

   b. Primary containment atmospheric particulate radioactivity shall be recorded at least once every 4 hours.

   c. Drywell pressure and temperature shall be recorded at least once every 12 hours.

2. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

   a. Primary containment atmosphere particulate monitoring systems-performance of a sensor check at least once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.

   b. Primary containment sump leakage measurement system-performance of a sensor check at least once per 4 hours and a channel calibration test at least once per cycle.

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the **predicted** and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

2-5

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant system boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal **leakage** and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

Testing of all safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1080 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the mainsteam flow stoppage resulting from a postulated MSIV closure from a power of 1670 MW_t. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 71% of the

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO FACILITY OPERATING

LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 Introduction

During this current outage, a planned replacement of piping insulation was carried out, that permitted inspection of all welds in the Recirculation System at the Monticello Nuclear Generating Plant. The primary inspection was performed by ultrasonic (UT) methods, augmented by radiography to assist in evaluating suspect locations. This inspection resulted in the detection of cracks in 3 safe end to pipe welds and one pipe to elbow weld in the 12" Riser piping, and one crack in the 22" diameter Manifold End Cap to pipe weld. On the basis of these results, the licensee decided to reinforce the End Cap weld and one Riser pipe to safe end weld with a weld overlay similar to that performed earlier on Quad Cities 1 Reactor Water Cleanup System piping, even though the cracks were considered to be very shallow.

During touch-up grinding preparatory to the weld overlay, a leak was noted on the Riser pipe to safe end weld. The leak occurred at a different location than the indications identified by the ultrasonic examination. After the leak occurred, additional ultrasonic inspection was barely able to identify the leaking crack. After sealing the leak, the weld overlay was successfully accomplished.

Because the leak was not identified by ultrasonic testing (UT) and concern that the cracks were deeper than originally determined, the licensee decided to overlay all the Riser welds showing indications. During this overlay process, two more pipe to safe end welds developed small leaks, were sealed, and the weld overlay applied. The weld overlay was accomplished without incident on the Manifold End Cap weld.

After all repairs, a hydrostatic test at 110% of operating pressure was performed. This resulted in the detection of another very small leak in another Riser elbow to pipe weld. This was successfully sealed and overlay welded.

In all, 3 Riser safe end to pipe welds and two Riser elbow to pipe welds were found to be cracked, and were reinforced by a weld overlay. In addition, one cracked Manifold End Cap to pipe weld was reinforced by overlay welding.

In addition, this Safety Evaluation addresses an application dated July 6, 1981, in which the licensee proposed changes to the Technical Specifications to revise the requirements for the coolant leak dectection system.

2.0 Discussion

On October 19, 1982, the Commission issued a Confirmatory Action Letter which requested information on the cracks found on the recirculation system piping at Monticello. Specifically, the Confirmatory Action Letter requested that the licensee submit to the Commission, the results of the licensee's inspection, corrective actions, justification to return to power, and receive NRC concurrence before returning the unit to power. In a November 22, 1982 letter supplemented December 3, 1982, the licensee submitted additional information.

2.1 Description of Cracks

Table 1, from the licensee's submittal of November 22, 1982, describes the details of the results of the inspections prior to the detection of the leaking elbow to pipe weld in Riser G during the hydrostatic test. Note that all except two were determined to be very short axial (90° to the weld) cracks, because they are very short in comparison to the wall thickness, they are reported by the licensee as "radial". Short axial cracks have been noted previously, and leaks emanating from them were noted and reported at Quad Cities 1 in 1980. They probably occur in locations with high residual welding stresses in the circumferential direction. They are typically short, because the sensitized heat affected zone extends less than 1/2" on either side of the weld, and intergranular stress corrosion cracking (IGSCC) requires sensitization to be present. As noted at Quad Cities 1, however, such cracks can propagate into and through the weld, if it has high carbon and low ferrite.

Axial cracks are of much less concern from a safety standpoint than circumferential cracks that can grow through the wall and around the circumference of the pipe, for two reasons. First, the stress on the axial crack is almost all caused by pressure, and typically the pressure stress is low compared to the total stress acting on a circumferential crack, where bending stresses can be significant.

Second, because IGSCC is confined to sensitized material, they cannot grow to significant lengths. This point will be covered more fully later under Fracture Analysis.

Axial or radial cracks, if short, are very difficult to detect and size by UT, because they form under the crown of the weld, and it is usually difficult to direct the sound beam at the proper angle. They often can only be detected at very limited transducer locations. This appears to have been the case on the Monticello riser welds.

It should also be mentioned that because they can only be short in relation to the wall thickness, and the stresses tending to open them are low, even when they are through wall, they will cause very litte actual leakage, perhaps not enough to be detected with normal procedures.

In summary, although axial or radial IGSCC cracks are hard to find by UT, they will cause only small leaks and will not grow long enough to initiate a pipe burst unless the piping itself is completely sensitized.

2.2 Description of the Overlay Reinforcement

The weld overlay on the Riser piping welds consisted of a complete circumferential reinforcement nominally .54" thick. The nominal thickness of the piping is .75". The axial length of the overlay is 6", tapering at a nominal 18° angle. The overlay on the Manifold End Cap to pipe weld is similar, with a thickness of .5" on a wall thickness of 1.1", and is 5 inches in minimum length.

The effect of the overlay is to provide a reinforcement of IGSCC resistant material. The welding process also induces beneficial compressive residual stresses in the underlying cracked pipe, in both the hoop and axial directions.

2.3 Code Stress Analysis

The repaired piping was evaluated according to Section III, and was found to meet all requirements including seismic and fatigure requirements. This was done by conservatively taking no credit for the entire circumferential volume where the cracks were detected. A doughnut shaped groove was assumed to be removed in a manner to remove all of the cracked area. Although the geometrical configuration is not typical of Code design, the stress analysis was performed using the Code rules. The fatigue analysis used the standard set of transient conditions, and included a strength reduction factor of 5 in the calculation. The calculations show that the repair of joints in the Riser piping and the Manifold End Cap met all Code requirements for at least 5 years.

3.0 Evaluation

3.1 Effect of the Overlay Repair on the Recirculation System

The weld overlay shrinkage induces beneficial residual compressive stresses in the cracked pipe, but also causes other effects. These have been evaluated by NuTech for the licensee, and the results of their evaluation are summarized here. The overlay repair to the Riser safe end to pipe weld causes both an axial and radial shrinkage.

One effect caused by the 5/16" radial shrinkage is to compress the area of the safe end where the internal secondary thermal sleeve is attached. The only deleterious effect anticipated is that the ring nut holding the plate spring and secondary thermal sleeve in place will be compressed, making removal by unthreading difficult if not impossible. When the safe ends are replaced the parts can be cut apart.

Another minor effect is caused by the axial shrinkage produced by the weld overlay in the horizontal runs of the riser piping. This includes the three safe end to pipe welds, and the elbow to horizontal pipe weld. In each case, a bending stress of approximately 7 ksi is induced in the Sweepolet to pipe weld. As this is a displacement controlled stress, similar to a thermal stress, it represents a small additional secondary stress, and is acceptable.

A more significant effect is caused in riser D, where the vertical pipe to elbow weld was overlayed. Because the horizontal run to the safe end is fairly short, the bending stress induced at the pipe to safe end weld is large enough to require careful consideration. Although the analysis performed treating this displacement controlled stress as an additional thermal stress showed that the limits of Section III of the Code are not exceeded, the propensity for IGSCC at the safe end to pipe may be increased.

We have concluded that this does not represent a serious safety concern, for the following reasons:

1. The safe end to pipe welds at Monticello do not appear to be particularly subject to circumferential cracking from IGSCC, which would be the type caused by high bending stress. All safe end to pipe welds were inspected, and indications of circumferentially oriented IGSCC was only found on one joint, (E Riser) where it appeared to be associated with short axial cracks, and was very short (1.06").

2. The bending stress induced by the weld overlay is displacement controlled (self equilibrating loads) and would tend to be relieved by initiation of cracking.

3. If cracking did occur from this bending stress, it would tend to be asymmetrical, thus propagating through the wall in a local area. Thus it would be expected to leak, and thereby be detected long before it could propagate circumferentially to an extent that would jeopardize the overall integrity of the pipe.

Therefore, we conclude that the possible increase in propensity for IGSCC in D Riser Safe End to pipe weld does not constitute a significant safety concern even if cracking should develop.

Because the Manifold End Cap overlay is at the end of a piping run, the shrinkage induced has no effect on other parts of the system.

3.2 Fracture Analysis

3.2.1 Background

NuTech performed two types of fracture analyses to show that the safety margins against failure are at least equivalent to the margins inherent in the ASME Code.

One analysis method used is based on a new proposed flaw evaluation methodology for Section XI of the Code. This includes IWB 3640, "Acceptance Criteria for Flaws in Austenitic Stainless Steel Piping," and the associated Appendix C, "Evaluation of Flaws in Austenitic Stainless Steel Piping." Although these new sections have not yet been approved through the Main Committee, they have been approved through the first two levels, and full approval is expected at the next Code meeting. The NRC will review these modifications to the Code, for concurrence.

The basis for this criterion is the well known and accepted limit load for plastic collapse method of analysis. Specific development of this method for the evaluation of flaws in stainless steel piping has been done under EPRI contracts, and has been described in several reports, including References 1 and 2. For Code use, this calculational method has been used to develop simple tables, from which acceptable flaw sizes and shapes as a function of applied stresses can be read directly. These are Tables IWB 3642-1 and -2 for axial cracks, and Table 3641-1 and -2 for circumferential cracks. There are separate tables for Normal Conditions and Emergency and Faulted Conditions, with different safety margins. The tables provide a safety margin of between 2.5 and 3 for Normal Conditions, and about 1.5 for Emergency and Faulted Conditions. These are consistent with the overall basis of the Code. A comparison of the criterion with results of actual burst tests on stainless steel piping will be made later in this review, when the repair to the Manifold End Cap is discussed.

Note that the presence of more than one crack does not change the calculations. Multiple axial cracks do not interact, and are treated separately. Because safety evaluations of flaws must include considerations of future growth, proposed Appendix C also includes rules for calculating growth by fatigue and stress corrosion. The methodology for evaluating fatigue propagation appears acceptable, but we still have some reservations about the crack growth rates for IGSCC given in the Code. This is of no concern for the repaired cracks at Monticello, (as will be described later) but it could affect our evaluation of other cases.

3.2.2 Riser Flaw Repair Evaluation

NuTech performed an Appendix C evaluation in accordance with the proposed Appendix C of the Code of the most limiting flaw and repair on the riser piping. The short axial (or radial) flaws were approximated by using an assumed one inch long thru-wall flaw in the pipe, and using minimum pipe and overlay thicknesses. Table IWB 3642-1 was used to determine the allowable depth into the combined pipe-overlay wall. This gives a value of .75 a/t, or .89". The total thickness is a minimum of 1.19", comprised of a minimum wall thickness (allowing for counterbore) of .687", plus a minimum overlay of .50". Thus, the Code would permit crack growth by fatigue and stress corrosion of (.89 - .687") or .203". NuTech calculated the crack growth due to fatigue to be only .005" during the next 5 years of operation. NuTech also calculated growth by IGSCC in an axial direction. (IGSCC is not expected to occur in the type 308L high ferrite weld overlay) and concluded that the maximum expected growth would add only .009" to the length of the existing crack. Both of these values are insignificant.

The calculations for the allowable depth of the crack are overly conservative in this case, because the Code arbitrarily cuts off the allowable depths given in the tables for axial cracks to .75 a/t. Extrapolation of the values in the table would show that thru-wall cracks would be acceptable at the stress levels existing at these joints, if it weren't for the leakage problem.

We have checked NuTech's calculations and agree with its conclusions regarding the acceptability of the repaired riser welds according to Appendix C. We also conclude that, because of the truncation of the tables at .75 a/t, even more margin than required by the Code actually exists in these repaired welds. For example, at the design pressure, the Code would permit a crack this deep to be almost 6" long. Further, because the Type 308L overlay is not subject to IGSCC, essentially no growth of the existing cracks is to be expected, making the repaired welds less likely to cause future problems than the unrepaired welds in the system.

Some of the cracks in the riser welds were missed by the ultrasonic inspection, and were only discovered by leakage. We have no assurance that other riser welds do not also have short axial cracks essentially thru-wall. We have performed calculations in accordance with Appendix C to evaluate the safety margin that would be expected, should such undiscovered cracks be present. Because the tables only include crack depths up to .75 a/t, graphic extrapolation was used to estimate the length of a thru-wall flaw that would be acceptable from a safety standpoint. This approach yielded a value of about 1.8 inches for a thru-wall axial crack in an unrepaired riser pipe joint. The maximum expected length of an axial IGSCC crack would not be more than about 1.70 inches. This value assumes that a crack could grow completely across the weld at the OD, and 1/2 inch into base metal on both sides of the weld. As the sensitized zone probably does not extend even 1/2 inch from the weld, this is a very conservative estimate.

We conclude, therefore, that the maximum length of a possible thru-wall axial crack would be acceptable using the limit load analysis, with a safety margin on pressure of at least 2.5.

3.2.3 Manifold End Cap Flaw Repair Evaluation

NuTech performed similar calculations to show the acceptability of the End Cap weld repair. In this case, it used the pipe thickness without overlay for its evaluation, even though the overlay was actually done. These calculations also showed complete Code acceptability using the proposed Appendix C and IWB 3640.

These calculations are dependent on the accuracy of the sizing of the shallow axial crack that was reported, and could be invalidated by either incorrect sizing or rapid IGSCC growth of the detected crack.

We therefore performed additional calculations using other assumptions. The first calculation assumed that the detected crack was completely through the .95" pipe wall, and that the overlay was the specified minimum of .50". This approach then, assumes a crack of .63 a/t in a wall 1.45" thick. Using the table for Normal Conditions gives an acceptable flaw length of over 8", at the stress levels caused by design pressure (1248 psi vs. 1000 psi for operating). We consider it very improbable that a crack of such length could form considering the compressive residual stresses induced in the pipe by the weld overlay.

This particular geometry, 23" OD and 1.45" wall, is very similar to that of stainless steel pipes used by Battelle for burst tests. We compared the burst test results with the acceptable values in IWB 3642-1. We used the conditions imposed on the specimen used for test Number 25 described in Reference 3. In this test, a section of 24" diameter pipe with a 1.5" wall was used. A flaw was machined in the pipe in the axial direction that was 6" long and .9 inch deep. This almost duplicates the geometry used in the above calculations for the End Cap weld repair. The burst test did not result in an actual burst, but was terminated because of excessive leakage and the inability of the test specimen to hold pressure. The maximum pressure attained was 4050 psi. With this crack geometry, the proposed Appendix C would permit a maximum stress ratio (stress/S_m, where S_m is the Code specified allowable stress intensity) of .75; whereas failure occurred at a stress ratio of 1.92, demonstrating a safety margin on pressure of at least 2.56 against burst.

We also performed Appendix C calculations in accordance with the proposed Appendix C of the Code for the hypothetical case of an undetected thru-wall axial crack in another end cap. Under operating pressure, and again extrapolating the Code tables to thru-wall geometry, the results show that a thru-wall crack 3.2" long would meet the Code criterion. The maximum length expected by IGSCC would be less than 2". We conclude, therefore, that even thru-wall undetected cracks in other end caps are not likely to grow to a size that would decrease the Code intended margin.

3.3 Tearing Modulus Fracture Evaluation

NuTech also performed fracture mechanics calculations using the more sophisticated Tearing Modulus approach. This type of elastic-plastic fracture mechanics was initially developed on an NRC contract, and has been widely accepted and used during the past 5 years. It is recognized that the limit load approach is conservative, and that much larger margins are actually present in many cases.

Tearing Modulus calculations were performed for both the repaired Riser welds and the Manifold End Cap weld. As expected, the calculations show that very large margins against failure are present. Although the material properties in the actual pipes and overlays may be somewhat lower than those used for the calculations, it is apparent that margins well over those intended by the Code are shown to be present by this approach.

3.4 Conclusion of the Fracture Analysis Review

The safety margins provided by the overlay repair to the cracked Riser and Manifold End Cap welds are shown by the proposed Code rules cited above to be acceptable. Crack propagation to the extent of leakage is considered very unlikely.

Staff calculations using the same Code rules also show acceptable safety margins for postulated undetected and unrepaired thru-wall cracks in Riser and End Cap welds, although small amounts of leakage would occur.

Tearing Modulus analyses of cracked welds show that even larger safety margins exist than are inherent in the Code approach.

3.5 Augmented Leak Detection

By letter dated July 6, 1981, Northern States Power responded to Generic Letter 81-04, "Implementation of NUREG-0313, Rev 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Rev. 1 recommends that leak detection should be augmented for plants that have piping susceptable to IGSCC. In this letter, the licensee requested a change in the Technical Specifications to revise the Limiting Conditions for Operation and Surveillance Requirements in the leak detection system. The improved leak detection proposed by the licensee consists of the following:

a. In addition to the existing Technical Specification limit of 5 gpm Unidentified leakage, the licensee proposed to revise the Technical Specifications to add a condition, that in the event of an increase in unidentified leakage of two gallons/minute or more within any 4-hour period, or 20 gallons/minute total leakage (averaged over a 24-hour period), the reactor will be placed in a cold shutdown condition within 24 hours for inspection.

b. Drywell leakage will be measured and recorded every four hours.

c. At least one of the leakage measurement instruments associated with each sump will be operable.

d. The drywell atmospheric particulate radioactivity monitoring system will be operable or a sample shall be taken and analyzed every four hours.

We conclude that implementation of these measures provide the augmentation recommended in NUREG 0313, Rev. 1, and will provide additional assurance that possible cracks in pipes will be detected before growing to a size that will compromise the safety of the plant. Therefore, we find that the proposed changes to the Technical Specifications are acceptable.

3.6 Summary and Safety Conclusions

We have reviewed Northern States Power's submittals regarding actions taken during this refueling outage on the Recirculation Piping System in the Monticello Plant. This includes location and descriptions of the defect found, description of repairs performed, stress and fracture analyses of the present configuration of the system, and plans for augmented leak detection.

We conclude that the Monticello plant can safely return to power and operate in its present configuration at least until the next refueling outage.

Nevertheless, we still have concerns regarding the long term growth of small IGSCC cracks that may be present but not detected during this inspection. Further, we feel that the effect of the additional stress imposed on the Safe End to pipe weld in Riser D by the overlay may increase the probability of initiation of IGSCC at this location.

For these reasons, we require that plans for inspection and/or modification of the Recirculation Piping System during the next refueling outage be submitted for our review and comment before the start of the outage.

4.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 10, 1982

Principal Contributors: Warren Hazelton
Helen Nicolaras

References

Reference 1. EPRI NP-2472-SY "The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping", July, 1982.

Reference 2. EPRI NP-2705-SR "Structural Mechanics Program: Progress in 1981, October, 1982.

Reference 3. BMI-1866 "Investigation of the Initiation and Extent of Ductile Pipe Rupture," July, 1969.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised the license and the Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment (1) approves the repair to the recirculation system piping and authorizes resumption of power, subject to certain conditions, and (2) changes the Technical Specifications to revise the Limiting Conditions for Operation and Surveillance Requirements for the coolant leak detection system.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated July 6, 1981, as supplemented November 22 and December 3, 1982 (2) Amendment No. 14 to License No. DPR-22, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 10th day of December, 1982

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing