REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)
DISTRIBUTION FOR INCOMING MATERIAL 50-263

REC: NRC

ORG: MAYER L O
N STATES PWR

DOCDATE: 01/18/78
DATE RCVD: 01/24/78

DOCTYPE: LETTER NOTARIZED: YES

COPIES RECEIVED
LTR 3 ENCL 40

SUBJECT:
REQUEST FOR CHANGE IN TECH SPECS IN RESPONSE TO NRC LTR OF 09/15/77 CONCERNING AUGMENTED INSPEC OF PIPING SUSCEPTIBLE TO STRESS CORROSION CRACKING.

PLANT NAME: MONTICELLO

REVIEWER INITIAL: XBT
DISTRIBUTOR INITIAL:

***************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS *****************

SURVEILLANCE TESTING OF RELIEF VALVES & STRESS CORROSION CRACKING FOR
(DISTRIBUTION CODE A013)

FOR ACTION: BRANCH CHIEF DAVIS**W/7 ENCL

INTERNAL:
REG FILE**W/ENCL
I&E**W/2 ENCL
HANAUER**W/ENCL
SHAO**W/ENCL
BUTLER**W/ENCL
J. COLLINS**W/ENCL
NRC PDR**W/ENCL
OELD**W/ENCL
EISENHUT**W/ENCL
BAER**W/ENCL
GRIMES**W/ENCL
J. WETMORE**W/ENCL

EXTERNAL:
LPDR'S
MINNEAPOLIS, MN**W/ENCL
TIC**W/ENCL
NSIC**W/ENCL
ACRS CAT B**W/16 ENCL

DISTRIBUTION: LTR 24 ENCL 24
SIZE: 1P+22P

CONTROL NBR: 780260289

************************************ THE END ************************************

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

January 18, 1978

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Augmented Inspection of Piping Susceptible
to Stress Corrosion Cracking
License Amendment Request dated January 18, 1978

Attached are three originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A, of the Provisional Operating License for the Monticello Nuclear Generating Plant. The proposed changes are in response to an NRC letter dated September 15, 1977 from Mr D K Davis, Acting Chief, Operating Reactors Branch #2, Division of Operating Reactors. This letter requested us to review all Monticello piping for material which is susceptible to stress corrosion cracking and submit a program of augmented inspection if necessary. Our letter of December 14, 1977 informed the Commission that this work would be completed by January 31, 1978.

The attached License Amendment Request completes all of the action requested in Mr Davis's letter of September 15, 1977. We believe this submittal provides a final resolution of the issue of pipe cracking due to stress corrosion cracking.

L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/DMM/deh

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman
MECCA
Attn: R J Hatling
S J Gadler

Attachments

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

(License Amendment Request Dated January 18, 1978)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production & System Operation

On this 18th day of January, 1978, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

LICENSE AMENDMENT REQUEST
DATED JANUARY 18, 1978

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

Pursuant to 10CFR50.59, the holders of Provisional Operating License DPR-22 hereby propose the following changes to the Appendix A Technical Specifications:

PROPOSED CHANGES

A. Section 1, Definitions

Add the following new definitions to Section 1:

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 and 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.

AE. Non-Conforming Lines - Pipe and fitting material, including weld metal, which has not been shown to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition. Type 304 stainless steel is non-conforming unless:

1) All piping and welds are in the solution annealed condition, or
2) The component is protected from exposure to reactor coolant by cast or weld overlay austenitic stainless steel with 5% minimum ferrite or other materials having high resistance to oxygen-assisted stress corrosion.

AF. Service Sensitive Lines - defined as those that have experienced stress corrosion cracking in boiling water reactor service or are particularly susceptible to such cracking because of high stress or because they contain relatively stagnant, intermittent, or low flow coolant.

B. Specification 3.6.D/4.6.D, Coolant Leakage

1. Revise Specification 3.6.D to read:

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

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.

2. Revise Specification 4.6.D to read:

   D. Coolant Leakage

      1. Unidentified and Identified Leakage rates shall be computed at least once every 12 hours using primary containment floor and equipment drain sump monitoring equipment.

      2. Primary Containment atmospheric particulate radioactivity shall be monitored at least once every 12 hours.

      3. Drywell pressure and temperature shall be monitored at least once every 12 hours.

3. Revise the 3.6/4.6 Bases to include the above changes. Refer to Exhibit B Page 134.

C. <u>Revisions to Inservice Inspection Program</u>

   1. Revise pages 189S and 189T of Exhibit B of Monticello License Amendment Request dated August 30, 1977 to include new specifications 4.13.A.2 and 4.13.A.3 as follows:

      2. For Non-Conforming Lines which are not Service Sensitive, inspections required by 4.13.A.1 during the first 10-year inspection interval shall be completed by the end of the 1978 refueling outage. If these examinations reveal no incidence of stress corrosion cracking, the examination schedule may revert to that specified in 4.13.A.1.

      3. For Non-Conforming Lines which are Service Sensitive:

         a. The welds and adjoining areas of bypass piping of the discharge valves in the main recirculation loops, and of the austenitic stainless steel reactor core spray piping up to and including the second isolation valve, shall be examined at each reactor refueling outage or at other scheduled or unscheduled plant cold shutdowns. Successive examinations need not be closer than six months apart. In the event three successive examinations find the piping free of unacceptable indications, the examination may be extended to each 36 month interval, plus or minus 12 months, and may be limited to one bypass pipe run and one reactor core spray pipe run.

b. If Service Sensitive Lines other than those listed in 4.13.B.3.a above are identified, the welds and adjoining areas of this piping shall be subjected to examination at each reactor refueling outage or at other scheduled or unscheduled plant cold shutdowns on a sampling basis. Successive examinations need not be closer than six months apart. If unacceptable flaw indications are detected in any branch run, the remaining branch runs with similar functions and configurations shall be examined. In the event three successive examinations find the piping free of unacceptable indications, the examination schedule may revert to that specified in 4.13.A.1 with the exception that all examinations normally completed over a ten-year interval shall be completed each 80-month period.

2. Add new pages 189U and 189V to Exhibit B of Monticello License Amendment Request dated August 30, 1977 to revise the Bases to include the above changes. Refer to pages 189U and 189V in Exhibit B, attached.

REASON FOR CHANGES

These changes are being proposed at the request of the NRC Staff. They are consistent with the recommendations contained in NUREG-0313, July, 1977, "Technical Report on the Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

All of the reactor coolant system pressure boundary piping and fitting material, including weld material, has been reviewed and compared to the material selection and processing guidelines contained in NUREG-0313. Materials that do not meet these guidelines are identified in Table 1 (Non-Conforming Lines which are not Service Sensitive) and Table 2 (Non-Conforming Lines which are Service Sensitive). Tables 1 and 2 contain the following information:

a. System identification
b. Weld identification
c. History of completed examinations
d. Examination summary and future examination schedule

All Non-Conforming Lines at Monticello consist of piping constructed of Type 304 stainless steel. Field welds in these lines were not solution annealed. Service Sensitive Lines at Monticello consist of:

a. Core spray lines
b. Recirculation bypass lines (welds between 304 and 304L stainless steel)

To date, no other lines can be statistically demonstrated to be Service Sensitive.

The proposed changes conform to the requirements of NUREG-0313 with one exception. Type 316 stainless steel is not considered to be Non-Conforming in the mill annealed plus field welded condition. General Electric Company records show that no incidents of stress corrosion cracking have occurred in any Type 316 stainless steel pressure boundary line in which the piping system is in the metallurgical condition of "mill annealed + welded." 1145 weld joints of Type 316 stainless steel have performed satisfactorily in BWR service with no cracking incidents. 305 of these Type 316 stainless steel weld joints have seen duty in Service Sensitive Lines. Laboratory tests by General Electric have confirmed these observations.

Type 316 stainless steel is used at Monticello in the Residual Heat Removal System piping. These lines are listed in Table 3 which also contains a summary of examinations performed on them. No augmented inspection of these lines is planned.

SAFETY EVALUATION

The proposed changes to the technical specifications consist entirely of additions to the limiting conditions for operation and surveillance requirements now in effect. The proposed measures to detect the occurrence of stress corrosion cracking are positive actions consistent with existing technology.

Implementation of these proposed changes will significantly reduce the possibility that stress corrosion cracking will go undetected.

NON-CONFORMING LINES WHICH ARE NOT SERVICE SENSITIVE

SYSTEM IDENTIFICATION	WELD IDENTIFICATION	REPORT NO. OF COMPLETED EXAMINATIONS					EXAMINATION REQUIREMENTS
		1973	1974	1975 SPRING	1975 FALL	1977	
RECIRCULATION SYSTEM							Shop welds on Recirculation System were solution annealed and, therefore, are not considered NON-CONFORMING WELDS.
RECIRCULATION "A" (REW13A-28")	RCAF-2 RCAJ-3 RCAJ-5 RCAJ-7 RCAJ-15 RCAJ-17 RCAJ-21 RCAJ-23 RCAJ-24 RCAJ-28	SEE 79,73	FOOTNOTE (1)				A total of three (3) welds will be examined by the end of the next outage on Recirculation A and B. If these examinations reveal no incidence of stress corrosion cracking, the examination schedule will revert to normal requirements.
RECIRCULATION "B" (REW13B-28")	RCBF-2 RCBJ-3 RCBJ-5 RCBJ-6 RCBJ-7 RCBJ-13 RCBJ-15 RCBJ-19 RCBJ-21 RCBJ-22 RCBJ-26 RCBJ-28	SEE	FOOTNOTE (1) 68,72				(See Above)
RECIRCULATION MANIFOLD "A" (REW32-22")	RMAJ-8 RMAJ-15 RMAJ-16	45					One (1) weld will be examined by the end of the next outage on either Recirculation Manifold A or B. If this examination reveals no incidence of stress corrosion cracking, the examination schedule will revert to normal requirements.
RECIRCULATION MANIFOLD "B" (REW32-22")	RMBJ-8 RMBJ-15 RMBJ-16						

NON-CONFORMING LINES WHICH ARE NOT SERVICE SENSITIVE

SYSTEM IDENTIFICATION	WELD IDENTIFICATION	REPORT NO. OF COMPLETED EXAMINATIONS 1973	1974	1975 SPRING	1975 FALL	1977	EXAMINATION REQUIREMENTS
RECIRCULATION RISERS							Requirements are based on all Recirculation Risers.
"A" (REW32-12")	RRAF-2		116				Examinations performed during the first 80 months on Exam Category B-J, did not reveal incidence of stress corrosion cracking. Therefore, the examination schedule will revert to normal requirements for this Exam Category.
	RRAJ-3		117				
	RRAJ-5				54		
	RRAJ-7				55		
"B" (REW22-12")	RRBF-2						
	RRBJ-3						
	RRBJ-5						
	RRBJ-7						
"C" (REW21-12")	RRCF-2				106		Welds RRBF-2, RRFF-2, RRHF-2 and RRKF-2 (Exam Category B-F) will be examined by the end of the next outage. If these examinations reveal no incidence of stress corrosion cracking, the examination schedule will revert to normal requirements.
	RRCJ-3				107		
	RRCJ-5			43			
	RRCJ-7			40			
"D" (REW20-12")	RRDF-2	77					
	RRDJ-3	84					
	RRDJ-5			57			
	RRDJ-7			56			
"E" (REW19-12")	RREF-2					204	
	RREJ-3					203	
	RREJ-5					201	
	RREJ-7					162	
"F" (REW14-12")	RRFF-2						
	RRFJ-3						
	RRFJ-5						
	RRFJ-7						
"G" (REW15-12")	RRGF-2				112		
	RRGJ-3				108		
	RRGJ-5			44			
	RRGJ-7			45			
"H" (REW16-12")	RRHF-2						
	RRHJ-3						
	RRHJ-5						
	RRHJ-7						
"J" (REW17-12")	RRJF-2	82					
	RRJJ-3	83					
	RRJJ-5				51		
	RRJJ-7				52		
"K" (REW18-12")	RRKF-2						
	RRKJ-3						
	RRKJ-5						
	RRKJ-7		114				

(1) Component is protected from exposure to reactor coolant by austenitic stainless steel weld metal overlay, with 5% minimum ferrite.

NON-CONFORMING LINES WHICH ARE NOT SERVICE SENSITIVE

SYSTEM IDENTIFICATION	WELD IDENTIFICATION	REPORT NO. OF COMPLETED EXAMINATIONS					EXAMINATION REQUIREMENTS
		1973	1974	1975 SPRING	1975 FALL	1977	
HIGH PRESSURE COOLANT INJECTION (PS18-8")	PSAF-2B PSAF-2C		15 16				Examinations performed during the first 80 months did not reveal incidence of stress corrosion cracking. Therefore, the examination schedule will revert to normal requirements.
REACTOR WATER CLEANUP (REW3-4")	CWAJ-1 CWAJ-2A CWAF-2			58 59 57		59 56 36, 61	Examinations performed during the first 80 months did not reveal incidence of stress corrosion cracking. Therefore, the examination schedule will revert to normal requirements.
CONTROL ROD DRIVE RETURN (NOZZLE N9)	CAP TO SAFE-END					224	Control Rod Drive Return line capped during 1977 outage with 316L Stainless Steel.

NON-CONFORMING LINES WHICH ARE SERVICE SENSITIVE

SYSTEM IDENTIFICATION	WELD IDENTIFICATION	REPORT NO. OF COMPLETED EXAMINATIONS 1973	1974	1975 SPRING	1975 FALL	1977	EXAMINATION REQUIREMENTS
CORE SPRAY "A" (TW7-8"EF)	CSAJ-2A			73	11	129	Three or more successive examinations of these welds and adjoining pipe metal were performed and were found to be free of unacceptable indications. Subsequently, further examinations will be extended to each 36 month interval, plus or minus 12 months, and will be limited to one core spray line, A or B.
	CSAJ-3			72	19	138	
	CSAJ-4			69	20	124	
	CSAJ-5			64	21	164	
	CSAJ-8			66	22	123	
	CSAJ-9			75	23	125	
	CSAF-10	95		70	24	131	
	CSAF-14	93		74	25	132	
	CSAJ-16	98		68	26	165	
	CSAJ-17	99		71	27	135	
	CSAF-18			76	28	134	
CORE SPRAY "B" (TW11-8"EF)	CSBJ-2A			67	1	140	(See Above)
	CSBJ-3			65	3	141	
	CSBJ-4			77	4	121	
	CSBJ-5			83	17	166	
	CSBJ-8			84	5	122	
	CSBF-9			81	6	144	
	CSBF-12			80	7	145	
	CSBJ-13			82	8	146	
	CSBJ-14			78	9	147	
	CSBF-16			79	10	167	
RECIRCULATION BY-PASS "A" (REW24-4")	RBAJ-2					2	The Recirculation By-Pass Lines (A and B) were replaced with 304L Stainless Steel (<.035% Carbon) during the Fall 1975 refueling outage. The two welds identified (between 304L and the original 304) on each line, will be examined during each of the next two outages. If these examinations reveal no incidence of stress corrosion cracking, further examinations will be extended to each 36 month interval, plus or minus 12 months, and will be limited to one By-Pass line, A or B.
	RBAJ-16					15	
RECIRCULATION BY-PASS "B" (REW25-4")	RBBJ-2					18	
	RBBJ-19					32	

TYPE 316 STAINLESS STEEL LINES

SYSTEM IDENTIFICATION	WELD IDENTIFICATION	REPORT NO. OF COMPLETED EXAMINATIONS					EXAMINATION REQUIREMENTS
		1973	1974	1975 SPRING	1975 FALL	1977	
RESIDUAL HEAT REMOVAL (REW10-18"ED)	RHAJ-1					168	No incidence of stress corrosion cracking. No augmented inspection program planned
	RHAJ-2					169	
	RHAJ-3					170	
	RHAF-4	63				171	
RESIDUAL HEAT REMOVAL (TW20-16"ED)	RHBJ-1					173	No incidence of stress corrosion cracking. No augmented inspection program planned.
	RHBJ-3					174	
	RHBF-4					187	
	RHBF-20						
	RHBJ-21	58					
	RHBJ-22	59					
	RHBF-24						
RESIDUAL HEAT REMOVAL (TW30-16"ED)	RHCJ-1		87			182	No incidence of stress corrosion cracking. No augmented inspection program planned.
	RHCJ-3		86			183	
	RHCF-4					188	
	RHCF-20	87	96				
	RHCJ-21		112				
	RHCJ-22		113				
	RHCF-23	88	97				

EXHIBIT B

LICENSE AMENDMENT REQUEST
DATED JANUARY 18, 1978

This exhibit consists of the following pages revised to incorporate the proposed Technical Specification changes:

5
5A
118A
134

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 and 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.

AE. Non-Conforming Lines - Pipe and fitting material, including weld metal, which has not been shown to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition. **Type 304 stainless steel is non-conforming unless:**

1) All piping and welds are in the solution annealed condition, or

2) The component is protected from exposure to the reactor coolant by cast or weld overlay austenitic stainless steel with 5% minimum ferrite or other materials having high resistance to oxygen-assisted stress corrosion.

AF. Service Sensitive Lines - defined as those that have experienced stress corrosion cracking in boiling water reactor service or are particularly susceptible to such cracking because of high stress or because they contain relatively stagnant, intermittent, or low flow coolant.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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

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.

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

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

1. Unidentified and Identified Leakage rates shall be computed at least once every 12 hours using primary containment floor and equipment drain sump monitoring equipment.

2. Primary containment atmospheric particulate radioactivity shall be monitored at least once every 12 hours.

3. Drywell pressure and temperature shall be monitored at least once every 12 hours.

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 and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

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 leakge 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. Analysés 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 full power steam generation rate.

Replace the following pages in Exhibit B of the Monticello License Amendment Request Dated August 30, 1977 with the attached revised pages:

Remove License Amendment Request dated August 30, 1977 Page Number:	Insert Attached B Page Number:
189S	189S
189T	189T
-	189U
-	189V

For convenience in reviewing this material, page 189R from Exhibit B of License Amendment Request dated August 30, 1977 is also attached. No changes to page 189R are proposed.

3.0 LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To assure the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. To be considered operable, Quality Group A, B, and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components, respectively, except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

4.13 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to the periodic inspection and testing of components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To verify the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. Inservice inspection of Quality Group A, B, and C components shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. For Non-Conforming Lines which are not Service Sensitive, inspections required by 4.13.A.1 during the first 10-year inspection interval shall be completed by the end of the 1978 refueling outage. If these examinations reveal no incidence of stress corrosion cracking, the examination schedule may revert to that specified in 4.13.A.1.

3. For Non-Conforming Lines which are Service Sensitive:

   a. The welds and adjoining areas of bypass piping of the discharge valves in the main recirculation loops, and of the austenitic stainless steel reactor core spray piping up to and including the second isolation valve, shall be examined at each reactor refueling outage or at other scheduled or unscheduled plant cold shutdowns. Successive examinations need not be closer than six months apart. In the event three successive examinations find the piping free of unacceptable indications, the examination may be extended to each 36 month interval, plus or minus 12 months, coinciding with a refueling outage, and may be limited to one bypass pipe run and one reactor core spray pipe run.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Inservice Testing of Pumps and Valves

1. To be considered operable, Quality Group A, B, and C pumps and valves shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for operability of ASME Code Class 1, 2, and 3 pumps and valves, respectively, except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

b. If Service Sensitive Lines other than those listed in 4.13.B.3.a above are identified, the welds and adjoining areas of this piping shall be subjected to examination at each reactor refueling outage or at other scheduled or unscheduled plant cold shutdowns on a sampling basis. Successive examinations need not be closer than six months apart. If unacceptable flaw indications are detected in any branch run, the remaining branch runs with similar functions and configurations shall be examined. In the event three successive examinations find the piping free of unacceptable indications, the examination schedule may revert to that specified in 4.13.A.1 with the exception that all examinations normally completed over a ten-year interval shall be completed each 80-month period.

B. Inservice Testing of Pumps and Valves

1. Inservice testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Bases 3.13 and 4.13:

The inservice inspection and testing program conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection and testing of components classified into NRC Quality Groups A, B, and C will conform to the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code.

Using Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as a guide, all Monticello components have been classified into Quality Groups. This classification serves as the basis for determining which ASME Code Class inspection and testing requirements are applicable to a given component. 10 CFR 50, Section 50.55a(g) requires components which are part of the reactor coolant pressure boundary and their supports to meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 1. Other safety-related components must meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 2 or 3.

The inservice inspection program must be updated at 40 month intervals. The program for testing pumps and valves for operational readiness must be updated every 20 months. A description of the updated programs should be submitted to the NRC for review at least 90 days before the start of each period. A suggested format for this description is contained in Appendix A to reference (1).

The inservice inspection and testing program must, to the extent practical, comply with the requirements in editions and addenda to the ASME Code that are "in effect" no more than six months before the start of the period covered by the updated program. The term "in effect" means both having been published by the ASME, and having been referenced in paragraph (b) of 10 CFR 50, Section 50.55a. If a code required inspection or test is impractical, requests for deviations are submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i). The information specified in Appendix B to reference (1) should be submitted for each deviation requested. Deviation requests should, if possible, be submitted to the NRC for review at least 90 days before the start of each period. Deviations identified during an inspection period may be grouped and requested at the end of each calendar quarter. It is expected that a small number of deviations will be identified during the inspection period, particularily the first period when new inspection and testing techniques will be utilized. A requested deviation request may be considered acceptable to the Commission until a formal disapproval has been received.

Small, hairline cracks in austenitic stainless steel piping in BWR facilities has been observed on several occasions. Data indicates that Type 304 austenitic stainless steel piping in the reactor coolant pressure boundary of the boiling water reactor is susceptible to stress corrosion cracking. Such cracking is caused by a combination of significant amounts of oxygen in the coolant, high stresses, and some sensitization of metal adjacent to welds. Cracks have occurred in the heat affected zones adjacent to welds, but are not expected to occur outside these areas, provided the pipe material is properly annealed. Pipe runs containing stagnant or low velocity fluids have been observed to be more susceptible to stress corrosion cracking than pipes containing a continuously flowing fluid during plant operation. Historically, these cracks have been identified either by volumetric examination, by leak detection systems, or by visual inspection. Because of the inherent high material toughness of austenitic stainless steel piping, stress corrosion cracking is unlikely to cause a rapidly propagating failure resulting in a loss of coolant accident.

Although the probability that stress corrosion cracks will propagate far enough to create a significant safety hazard is slight, the presence of such cracks is undesirable. The following steps have been taken to minimize this problem:

1. Where practical, pipe runs constructed of material susceptable to stress corrosion cracking and which contained stagnant or low velocity fluid have been replaced with materials not susceptable to cracking or they have been eliminated.

2. The reactor coolant leakage detection technical specifications have been amended to enhance the ability to detect unidentified leakage that may include through-wall cracks.

3. The program of inservice inspection has been augmented to increase the probability of crack detection in lines susceptable to stress corrosion cracking.

This program conforms to the Commission's guidelines for plants with operating licenses (reference 2).

References:

1. Letter from D. L. Ziemann, Chief, Operating Reactors Branch #2, USNRC, to L. O. Mayer, NSP, dated November 24, 1976.

2. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July, 1977.