U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/99004(DRP)
Licensee:	Northern States Power Company
Facility:	Monticello Nuclear Generating Station
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	May 21 through July 1, 1999
Inspectors:	S. Burton, Senior Resident Inspector D. Wrona, Resident Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Monticello Nuclear Generating Station
NRC Inspection Report 50-263/99004(DRP)

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

<u>Operations</u>

- A reactor startup on May 27, 1999, was performed in accordance with approved procedures. Infrequent evolution briefings performed for the startup were thorough and comprehensive. Reactor thermal limits were properly monitored throughout the startup. (Section O1.2)

- During a reactor startup with one average power range monitor bypassed per trip system, the licensee had procedural requirements in place to ensure that the minimum number of average power range and associated intermediate power range nuclear instruments remained operable. (Section O1.3)

<u>Maintenance</u>

- Operability of safety-related service water pumps was properly assessed after sandblasting material was introduced into the vicinity of the equipment through the ventilation system during the preparation of the building exterior for painting. (Section M1.1)

- Surveillance tests and valve lineups associated with reactor startup were performed by qualified individuals using approved procedures. Deficiencies identified during the performance of activities were properly dispositioned and corrected. (Section M1.2)

- Controls on overtime utilization were adequately implemented by the licensee for the licensee staff that the inspectors assessed. The licensee's control of overtime met the Technical Specification administrative requirements governing overtime. (Section M8.2)

<u>Engineering</u>

- Operability determinations for equipment susceptible to failure during a high-energy line break were incomplete and did not include an assessment of the full spectrum of potential single failures as required by the safety analysis report. Rather than perform the complex analysis required to determine equipment operability for the additional single failures not previously analyzed, the licensee elected to correct the deficiency by reinforcing the degraded structure that caused the adverse condition. (Section E1.1)

- The engineering department had not fully evaluated the impact of configuring the emergency diesel generator with the droop set above zero in the standby mode. Procedural enhancements and an adjustment of one breaker thermal overload setpoint were performed to further ensure operability of safety-related equipment. An engineering evaluation was initiated to reconfigure the emergency diesel generator

controls to have droop set at zero when the emergency diesel generator was in the standby mode, precluding the continual evaluation of component operability as equipment degraded or was repaired.
(Section E1.2)

Plant Support

- Although personnel were responsible to monitor their own accumulated dose, the licensee did not lower electronic dosimeter alarm setpoints to reflect radiological conditions during plant shutdown, a poor practice. (Section R8.1)

Report Details

Summary of Plant Status

The unit began the inspection period shutdown due to an unplanned scram that occurred on May 8, 1999, during the prior inspection period. The reactor was restarted on May 27, the generator was connected to the electrical distribution grid on May 28, and 100 percent power was achieved on May 29, 1999. With the exception of a brief reduction to 50 percent power on June 18, 1999, to replace recirculation pump motor generator brushes, the unit operated at approximately 100 percent power for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors observed various aspects of plant operations, including compliance with Technical Specifications (TSs); conformance with plant procedures and the Updated Safety Analysis Report (USAR); shift manning; communications; management oversight; proper system configuration and configuration control; housekeeping; operator performance during routine plant operations; the conduct of surveillance tests; and plant power changes.

The conduct of operations was professional and safety-conscious. Evolutions such as surveillance tests and plant power changes were well controlled and deliberate, and were performed in accordance with procedures. Shift turnover briefings were comprehensive and were typically attended by the operations superintendent and representatives from the scheduling, security, instrument and control, and electrical and mechanical maintenance departments. Housekeeping was generally good and discrepancies were promptly corrected. Containment isolation valves and portions of the reactor core isolation cooling system were found to be properly aligned. Specific events and noteworthy observations are detailed below.

O1.2 Reactor Startup On May 27, 1999

a. Inspection Scope (71707)

The inspectors observed portions of activities associated with restart of the reactor following the unplanned shutdown on May 8, 1999. Activities observed included pre-job briefings, drywell closure, reactor restart, reactor heat-up, synchronization of the main generator to the electrical distribution grid, and power ascension.

b. Observations and Findings

The inspectors observed portions of the operations committee review of startup readiness. Committee members discussed maintenance activities and equipment availability for startup. The inspectors reviewed closure documentation associated with condition reports generated as a result of the scram and found corrective actions to be reasonable and complete. Infrequent evolution briefings performed for the startup were thorough and comprehensive, and the shift managers involved with the briefings were clear about their expectations with respect to procedural use, safety, and operators being slow and deliberate during all activities. Reactor startup, heat-up, mode changes, and synchronization of the main generator to the electrical distribution grid were performed in accordance with procedures. During the startup, a process computer printout showed that a reactor fuel thermal limit was exceeded during the continuous withdrawal of one control rod. Operators, with recommendation from the reactor engineering staff, promptly inserted the rod in question. Reactor engineers determined that the thermal limit had not been exceeded and that the cause of the problem was due to the methodology used by the computer during the continuous rod withdrawal to compute the thermal limits. Subsequently, the control rod in question was withdrawn one step at a time to the same position to which it had been withdrawn earlier. Calculations performed throughout the step-withdrawal of the control rod demonstrated that thermal limits were acceptable. Reactor startup continued without further complication.

c. Conclusions

A reactor startup on May 27, 1999, was performed in accordance with approved procedures. Infrequent evolution briefings performed for the startup were thorough and comprehensive. Reactor thermal limits were properly monitored throughout the startup.

O1.3 Startup with Average Power Range Monitors (APRM) Bypassed

a. Inspection Scope (71707)

The inspectors observed that plant startups were conducted with one APRM channel in each reactor trip system continually bypassed. The inspectors reviewed TSs, standard TSs, the USAR, operating procedures, and interviewed engineers with respect to this practice.

b. Observations and Findings

The inspectors were concerned that an interrelating function between Intermediate Range Monitors (IRM) and APRMs, which was required to be operational when moving the mode switch from the startup position to the run position, could inadvertently be bypassed on more than one channel per trip system. When the mode switch was being moved from the startup position to the run position, and prior to the withdrawal of the IRMs, a downscale condition on an APRM in conjunction with an upscale condition on an associated IRM would cause a scram signal to be input from the associated IRM/APRM circuit. This relationship was commonly referred to as the IRM/APRM companion scram. Because the licensee operated with one APRM per trip system always bypassed, the inspectors were concerned that barriers might not exist to prevent an operator from bypassing an IRM associated with an APRM different from the one always bypassed in a trip system. Under these circumstances, two IRM/APRM companion scram circuits would be bypassed in a single trip system and the operable number of channels of nuclear instruments would be less than the minimum number of operable channels per trip system allowed by TSs. The inspectors interviewed operations department and engineering department personnel and found that an operator aid placard had been in place in the past to prevent this condition. Personnel could not readily identify what barrier currently existed to prevent this condition from inadvertently occurring.

A review of the TS indicated that the IRM/APRM companion scram was not specifically identified in TS table 3.1.1, "Reactor Protection System (SCRAM) Instrument Requirements," as a requirement for Monticello. Further review found that the companion scram was a requirement of the basis for TS Safety Limit 2.3.A, "Fuel Cladding Integrity Safety Limit." Additionally, a review of standard TSs indicated that the companion scram was required and listed in both the instrument table and the basis. Because the operability of the companion scram directly related to the basis for the fuel cladding integrity safety limit, the inspectors communicated their concern to the licensee. The inspectors noted that specific precautions to ensure that the proper number of nuclear instrument channels remained operable under all conditions did not exist. The licensee's reactor engineering group reviewed the configuration and noted that operating procedures require APRMs to be indicating greater than 5 percent of scale prior to operators transferring the mode switch from the startup position to the run position. Although this procedural requirement was not specifically identified as being established to ensure that instrument operability requirements were met, the inspectors concurred that requiring APRMs to be on scale prior to placing the mode switch in the run position effectively ensured that the minimum number of channels per trip system remained operable during startup configurations.

c. Conclusions

When performing a reactor startup with one average power range monitor bypassed per trip system, the licensee had procedural requirements in place to ensure that the minimum number of average power range and associated intermediate power range nuclear instruments remained operable.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Activities

a. Inspection Scope (62707)

In addition to minor maintenance activities observed during routine plant tours, the inspectors observed performance of maintenance activities conducted in accordance with Procedure 4151-1PM, Revision 4, "SBLC [Standby Liquid Control] Accumulator 11," and Procedure 4151-2PM, Revision 4, "SBLC Accumulator 12."

b. Observations and Findings

Work performed during maintenance activities was professional and thorough. All work was performed in accordance with approved procedures and the workers were knowledgeable of their assigned tasks. Appropriate radiological work permits were followed. The inspectors observed that maintenance supervisors and engineers were involved in the activities.

Additionally, the inspectors noted that during the sandblasting of an exterior wall in preparation for painting, some of the sandblasting material had been introduced through the ventilation system into the service water bays in the area of the safety-related emergency service water pumps. The licensee had also made a similar observation and evaluated the impact of the material on the operability of the safety-related pumps. The inspectors reviewed the operability determination and discussed it with the licensee. No deficiencies were noted.

c. Conclusion

Operability of safety-related service water pumps was properly assessed after sandblasting material was introduced into the vicinity of the equipment through the ventilation system during the preparation of the building exterior for painting.

M1.2 General Comments on Surveillance Test Activities

a. Inspection Scope (61726)

The inspectors observed or reviewed the performance of all or portions of the activities contained in the following surveillance test procedures:

- Procedure 2154-13, Revision 23, "RCIC System Prestart Valve Checklist," performed on May 23, 1999.
- Procedure 1371, Revision 4, "Drywell Prestart Inspection," performed on May 24, 1999.

b. Observation and Findings

In general, the inspectors found that the activities specified in the surveillance test procedures were performed in a professional and thorough manner and completed in accordance with the procedures. Personnel were knowledgeable and generally demonstrated effective three-part communications, self-checking, and peer-checking. When conducted, pre-job briefings were comprehensive. The inspectors frequently observed supervisors and system engineers monitoring job progress. When applicable, appropriate radiation control measures were in place.

The inspectors observed operators perform portions of a system lineup in accordance with the "Reactor Core Isolation Cooling System Prestart Valve Checklist." During the performance of the lineup, two operators were in the room at the same time, independently working on different portions of the checklist. The inspectors questioned operators on how they verified the position of valves with missing identification tags. Operators indicated that they would use the controlled piping and instrument diagrams to locate the valve. The inspectors observed a valve with a missing identification tag and noted that the operators had requested a label. No deficiencies were identified.

The inspectors noted no concerns when they observed system engineers perform the "Drywell Prestart Inspection." The engineers identified minor discrepancies and initiated work orders to have the issues corrected prior to startup. A small amount of miscellaneous debris was identified by the engineers and by the inspectors and removed from the drywell.

c. Conclusions

Surveillance tests and valve lineups associated with reactor startup were performed by qualified individuals using approved procedures. Deficiencies identified during the performance of activities were properly dispositioned and corrected.

M8 Miscellaneous Maintenance Issues (71707, 92902)

M8.1 (Closed) Licensee Event Report 50-263/98001: Containment Isolation Valve Leakage Greater than Allowed by Technical Specifications.

During refueling outage local leak rate testing, the licensee identified that the leakage associated with three containment isolation valves, one of which was the "D" inboard main steam isolation valve, caused the combined maximum flow path leakage to exceed the TS limit. The licensee also identified that the leakage associated with main steam isolation valve leakage was in excess of the TS limit.

This event was determined to be of low safety significance since the redundant valve for each of the three valves had minimal leakage. All three valves were repaired prior to the licensee returning the unit to power following the refueling outage.

In the licensee event report, the licensee stated that the condition could have existed during some of the previous operating period and assessed the impact of the inoperable valves on total allowable leakage. The licensee determined that no limits were exceeded due to the associated in-line valves having a lower than allowable leakage. Because the valves were determined to be inoperable at the time of discovery, and the plant had actually operated with leakage less than allowed by TS, the inspectors concluded that no violation of NRC requirements had occurred. This issue was entered into the licensee's corrective action program under CR 98000737, "D MSIV [main steam isolation valve] Exceeds Allowed Leakage in Technical Specifications"; CR 98000958, "FW-94-2 [feedwater isolation check valve] Failed LLRT [local leak rate test] 0137-08-2 on 3/26/98"; and CR 98001091, "Local Leak Rate Failure of AO-2378 [torus purge inboard containment isolation valve]."

M8.2 Use and Documentation of Overtime Deviations

a. Inspection Scope (71707)

The inspectors reviewed the following documents:

- Security ingress and egress times for portions of April and May 1999 for four individuals;
- Forms 3361, "Authorization to Exceed Overtime Work Restrictions," completed between April and May 1999;
- 4 AWI-08.10.01, Revision 3, "Overtime Restrictions and Fitness for Duty Requirements"; and
- Technical Specification 6.1.F, "Administrative Controls for the Use and Authorization of Overtime."

b. Observations and Findings

The inspectors reviewed overtime use during April and May 1999, which encompassed two forced outages. The inspectors sampled individuals who had major roles associated with equipment concerns identified during the outages. The inspectors used the above records to aid in determining potential deviations from overtime guidelines. The inspectors utilized main gate ingress and egress times as a record of time on site and based the conclusions on this assumption. The inspectors also recognized that the ingress and egress times did not necessarily reflect actual work hours.

The inspectors performed a spot check of the ingress and egress times for four individuals and noted that working hours were generally limited to the hours specified in TS 6.1.F and when the hours deviated from those specified in TS 6.1.F, the licensee appropriately documented and approved the deviation.

c. Conclusions

Controls on overtime used were adequately implemented by the licensee for the licensee staff that the inspectors assessed. The licensee's control of overtime met the Technical Specification administrative requirements governing overtime.

III. Engineering

E1 Conduct of Engineering

E1.1 High-Energy Line Break Forces on Masonry Walls

a. Inspection Scope (37551)

The inspectors reviewed the licensee's actions associated with Inspection Followup Item (IFI) 50-263/97018-06(DRP) as documented in CR 97003052, "IEB 80-11 Program Inadequately Documented the Analysis and Effects of HELB Forces on Masonry Walls. Need Further Review."

b. Observations and Findings

Inspection Followup Item 50-263/97018-06(DRP) identified that formal documentation of the effects of a high-energy line break on 78 safety-related masonry walls may not exist. The walls which were evaluated in response to NRC Bulletin 80-11, "Masonry Wall Design," did not include an evaluation of the effects of area pressurization due to a high-energy line break.

The licensee subsequently determined the effects of high-energy line breaks on masonry walls and identified that two block walls, which were part of the stairwell enclosure leading to the "B" residual heat removal room, were susceptible to failure. The licensee documented the resolution of this issue in CR 97003052. The licensee assumed that the failure of the block walls would render the "B" residual heat removal train and the "B" core spray train inoperable and provided the analysis to show how the requirements specified in the USAR were still complied with.

The set of conditions that were required to be satisfied to adequately cope with a high-energy line break were specified in the USAR. Among these conditions was the ability to cope with a high-energy line break given that a single active failure could also occur. The inspectors reviewed CR 97003052 and noted that the licensee did not document all applicable single active failures. The inspectors questioned which single active failures were considered and subsequently provided examples of failures that had not been reviewed which could impact the equipment operability determination. Due to the complexity of analyzing the newly postulated single failures mechanisms, the licensee elected to correct the blockwall deficiency by reinforcing the walls susceptible to failure during a high-energy line break. The inspectors determined that the impact on safety of not completing the full analyses was minimal because HELB guidelines allow for operator actions and the licensee would have been able to detect and correct adverse conditions using operator actions. This negated the need to assess single active failures along with failure of the division "B" equipment previously mentioned above.

c. Conclusions

Operability determinations for equipment susceptible to failure during a high-energy line break were incomplete and did not include an assessment of the full spectrum of potential single failures as required by the safety analysis report. Rather than perform the complex analysis required to determine equipment operability for the additional single failures not previously analyzed, the licensee elected to correct the deficiency by reinforcing the degraded structure that caused the adverse condition.

E1.2 Diesel Generator Droop Set Above Zero During Normal Operations

a. Inspection Scope (37551)

The inspectors noted that Procedure 0187-2, Revision 29, "12 Emergency Diesel Generator/12 Emergency Service Water Pump System Tests," did not leave the diesel droop set at zero upon completion of the test. The inspectors noted that operation of the emergency diesel generator (EDG) in this configuration during a loss of offsite power would result in diesel speed varying as loads on the diesel were changed. The inspectors additionally reviewed the Updated Safety Analysis Report, TSs, and the EDG technical manual to assess requirements associated with operation in this configuration.

b. Observations and Findings

Droop is a method of creating stability in a diesel generator speed governor when the generator is operating in parallel with other units. A droop setting above zero ensures that the generator does not attempt to transfer loads between other generators operating in parallel. A droop setting of zero allows the generator to accept and reject load without an associated change in speed. Droop settings above zero when the generator is operating on an isolated bus, as would be expected during a loss of offsite power, would result in the generator's speed, and associated current frequency, increasing with a reduction in load and decreasing with an increase in load.

Operating on an isolated bus with a varying alternating current frequency would result in motor speeds changing and indirectly affecting all non-resistive loads on the associated bus. The inspectors were concerned that operation in this configuration had the potential to cause pumps to operate in less than design flow or overcurrent conditions, depending upon operation of the EDG at a low or high operating frequency, respectively. Additionally, valve motor torque may be too low or too high to permit proper valve operation. Changes in EDG frequency also had the potential to impact the operability of solid-state equipment, such as battery chargers, that rely on a specific frequency band to remain operable.

The inspectors reviewed the EDG governor technical manual and found that it recommended that droop be set at zero when the EDG was operating on an isolated bus. The inspectors noted that although operators were trained to maintain EDG frequency at 60 cycles per second when the diesel was operating on an isolated bus, procedural precautions or guidance was insufficient to ensure that frequency changes did not affect equipment operability. Additionally, the inspectors were concerned about frequency affecting safety-related equipment operability. The licensee agreed with the inspectors and initiated condition report CR 99001403 to evaluate operability of affected components prior to restarting the reactor on May 27, 1999.

The licensee evaluated safety-related loads that were potentially affected by operation in this configuration and determined that all loads would function as designed; however, one load, the #11 emergency service water pump, was determined to have minimal margin to the breaker thermal trip setting. A work order was written and the overload device was reset prior to startup. Changes were also made to operating procedures to ensure that EDG frequency was maintained at 60 cycles per second when the EDGs were operated with droop set above zero. An engineering evaluation was initiated to reconfigure the EDG governor controls to have droop set at zero when the EDG was in the standby mode. Operation of the EDG with droop set at zero while in standby would preclude the licensee from having to continually evaluate component operability as equipment degraded or was repaired.

c. <u>Conclusions</u>

The engineering department had not fully evaluated the impact of configuring the emergency diesel generator with the droop set above zero in the standby mode. Procedural enhancements and an adjustment to one breaker thermal overload setpoint were performed to further ensure operability of safety-related equipment. An engineering evaluation was initiated to have droop set at zero when the EDG was in the standby mode, precluding the continual evaluation of component operability as equipment degraded or was repaired.

E8.1 <u>(Closed) Inspection Followup Item 50-263/97018-06(DRP)</u>: Effects of high-energy line break forces on masonry walls not documented. (See Section E1.1 of this report.)

E8.2 <u>Summary of Checklist Review Results for Y2K Readiness (TI 2515/141)</u>

The inspectors conducted an abbreviated review of Y2K activities and documentation using Temporary Instruction (TI) 2515/141, "Review of Year 2000 (Y2K) Readiness of Computer Systems at Nuclear Power Plants." The review addressed aspects of Y2K management planning, documentation, implementation planning, initial assessment, detailed assessment, remediation activities, Y2K testing and validation, notification activities, and contingency planning. The inspectors used NEI/NUSMG [Nuclear Energy Institute/Nuclear Utilities Software Management Group] 97-07, "Nuclear Utility

Year 2000 Readiness," and NEI/NUSMG 98-07, "Nuclear Utility Year 2000 Readiness Contingency Planning," as the primary references for this review.

The results of this review will be combined with the results of other reviews in a summary report to be issued by July 31, 1999.

IV. Plant Support

R8 Miscellaneous Radiation Protection Issues

R8.1 Electronic Dosimeter Dose and Dose Rate Setpoints

a. Inspection Scope (71750)

During routine tours while the reactor was shutdown, the inspectors assessed radiological area entry requirements and radiation work permit (RWP) compliance for various plant locations. Radiation work permit 42, Revision 17, "911 Turbine - Condenser Room General Area," and RWP 44, Revision 14, "911 Turbine - Steam Jet Air Ejector Room," were reviewed as part of this assessment.

b. Observations and Findings

On May 23, 1999, the inspectors observed that the instructions specified in RWPs associated with general entry into the condenser room and the steam jet air ejector room while the unit was shutdown required electronic dosimeter settings that were the same as if the unit were in operation. Specifically, RWPs 42 and 44 specified electronic dosimeter dose rate and accumulated dose setpoints of 2,000 millirem per hour (mrem/hr) and 200 mrem; and 3,000 mrem/hr and 200 mrem, respectively. The highest radiation dose rates in the rooms, as posted, was about 10 mrem/hr. The inspectors were concerned that the electronic dosimeter alarm setpoints were not sufficient to alert personnel to a change in radiological conditions.

The radiation protection staff stated that personnel were trained to periodically monitor their accumulated dose on the electronic dosimeter and that the electronic dosimeter alarms were used as a backup method of alerting operators to unexpected conditions. Although personnel were responsible to monitor their own accumulated dose, the licensee did not take advantage of the capabilities of the electronic dosimeter alarm setpoints to aid workers in maintaining awareness of unanticipated changes in radiological conditions. The licensee concurred that this was a poor practice and elected to establish alarm setpoints that better coincided with shutdown conditions.

c. Conclusions

Although personnel were responsible to monitor their own accumulated dose, the licensee did not lower electronic dosimeter alarm setpoints to reflect radiological conditions during plant shutdown, a poor practice.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During routine activities or tours, the inspectors monitored the licensee's security program to ensure that observed actions were being implemented according to the approved security plan. The inspectors observed that persons within the protected area displayed proper photo-identification badges and those individuals requiring escorts were properly escorted. The inspector also verified that vital areas were locked and alarmed. Additionally, the inspectors verified that observed personnel and packages entering the protected area were searched by appropriate equipment or by hand. The inspectors toured the protected area perimeter fence and found no deficiencies.

F2 Status of Fire Protection Facilities and Equipment

F2.1 General Comments (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection. Fire extinguishers and fire hoses were properly stored and inspected by licensee personnel. No notable degradation of equipment was observed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 1, 1999. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Day, Plant Manager
M. Hammer, Site Manager
K. Jepson, Superintendent, Chemistry & Environmental Protection
M. Lechner, Acting General Superintendent Operations
L. Nolan, General Superintendent Safety Assessment
E. Reilly, General Superintendent Maintenance
C. S[illegible]bonski, General Superintendent Engineering
A. Ward, Manager Quality Services
L. Wilkerson, Superintendent Security
J. Windschill, General Superintendent, Radiation Services

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
TI 2515/141	Y2K Readinesss Assessment

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed

50-263/98-001	LER	Containment isolation valve leakage greater than allowed by Technical Specifications
50-263/97018-06	IFI	Effects of high-energy line break forces on masonry walls not documented

Discussed

None

LIST OF ACRONYMS USED

AOV	Air-Operated Valve
APRM	Average Power Range Monitor
CFR	Code of Federal Regulations
CS	Core Spray
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
IFI	Inspection Followup Item
IP	Inspection Procedure
IRM	Intermediate Range Monitor
LER	Licensee Event Report
LLRT	Local Leak Rate Test
mrem/hr	millirem per hour
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSP	Northern States Power
PDR	Public Document Room
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RWP	Radiation Work Permit
TI	Temporary Instruction
TS	Technical Specification
USAR	Updated Safety Analysis Report