EXHIBIT A

Monticello Nuclear Generating Plant

License Amendment Request Dated June 24, 1983

Proposed Changes to the Technical Specifications
Appendix A of Operating License DPR-22

Pursuant to 10 CFR Part 50, Section 50.59 and Section 50.90, the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications:

1. Overtime Limitations

Proposed Changes

Add restrictions on plant employee overtime as shown in Exhibit B, pages 233 and 233a.

Reason for Change

This change implements the requirements of NUREG-0737, Item I.A.1.3. The proposed wording conforms to the guidelines contained in Generic Letter 82-12 dated June 15, 1982 with the exception of deviations described in our letters dated November 16, 1982 and February 21, 1983. These deviations were approved by the Commission in a letter dated March 17, 1983 from Mr Domenic B Vassallo, Chief, Operating Reactors Branch #2, Division of Licensing, USNRC.

An additional minor change on page 233 is also requested at this time. "Plant training program" is changed to "training program" in the last line of Specification 6.1.E on page 233. As noted in our License Amendment Request dated September 24, 1982 the training program is now under the Manager Production Training in the corporate headquarters.

Significant Hazards Evaluation

The proposed overtime guidelines constitute an additional limitation, restriction, or control not presently included in the Technical Specifications.

The proposed change in wording of Specification 6.1.E is a purely administrative change to the Technical Specifications.

For these reasons operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

2. Reporting Safety/Relief Valve Failures and Challenges

Proposed Change

Add a requirement to report safety/relief valve failures and challenges as shown in Exhibit B, page 249a.

Reason for Change

This change implements the requirements of NUREG-0737, Item II.K.3.3.

Significant Hazards Evaluation

The proposed reporting requirement constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications.

For this reason, operation of the Monticello Nuclear Generating Plant in accordance with the proposed change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

3. RCIC Restart and RCIC Suction Transfer

Proposed Changes

Add limiting conditions for operation and surveillance requirements for RCIC system automatic suction transfer from the condensate storage tank to the suppression pool. Add limiting conditions for operation and surveillance requirements for automatic restart on subsequent low water level after the RCIC System has shutdown due to high reactor water level.

The automatic restart capability has not yet been completed. The work will be scheduled for the next refueling outage. Footnotes provide for delayed applicability of these requirements.

Refer to Exhibt B, pages i, vi, 46, 60b, 61, 71, 111, 111a, and 112. To avoid confusion, changes requested in our February 15, 1983 License Amendment Request are also shown on these pages.

The allowable deviation specified in Table 4.2.2 for condensate storage tank level of -6 inches is believed to be a reasonable number based on the sensitivity of the instrumentation installed and the volume of water remaining in the tank at the lowest permissible transfer (1' 6"). Calibration and testing of these instruments is not possible during operation. Testing at refueling intervals has been specified.

3. RCIC Restart and RCIC Suction Transfer (continued)

Reason for Change

The proposed changes implement NUREG-0737, Items II.K.3.13 and II.K.3.22. Plant modifications required by these NRC Action Plan items were described in detail in our letters dated December 31, 1981, September 8, 1982, and April 21, 1983 and in various BWR Owner' Group submittals referenced in our letters.

Significant Hazards Evaluation

The proposed limiting conditions for operation and surveillance requirements constitute additional limitations, restrictions, and controls not presently included in the Technical Specifications.

For this reason, operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

4. Isolation of RCIC Modification

Proposed Change

Add a time delay to the trip setting specified for the RCIC High Steam Flow isolation in Table 3.2.1 of the Technical Specifications. Refer to Exhibit B, page 50.

Reason for Change

This change implements the requirements of NUREG-0737, Item III.K.3.15. Modifications were described in our letters dated December 30, 1980 and December 22, 1981. NRC acceptance of our modification was documented in a letter dated January 6, 1983 from Mr Domenic B Vassallo, Chief, Operating Reactors Branch #2, Division of Licensing, USNRC.

No modifications to the HPCI System, which uses a venturi installed in the steam line, were necessary.

4. Isolation of RCIC Modification (continued)

Significant Hazards Evaluation

The proposed time delay for RCIC System isolation on high steam flow constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications.

The proposed time delay of 5 ± 2 seconds conforms to the BWR Owner's Group recommendations. These recommendations have been approved by the NRC Staff. A three second time delay has been shown to be sufficient to prevent spurious isolation. Time delays of up to 7 seconds do not affect the steam line break analysis which assumes a 13-sec valve closure delay period before the valve begins to stroke closed. These conclusions are documented in General Electric Topical Report NEDE-24953.

For these reasons, operation of the Monticello Nuclear Generating Plant in accordance with the proposed change will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

5. Addition of Hydraulic Snubbers

Proposed Change

Add additional snubbers to Table 3.6.1 of the Technical Specifications. Refer to Exhibit B, pages 131, 132, and 132a.

Reason for Change

Technical Specification 3.6.H.3 requires submitting revisions to Table 3.6.1 with the next License Amendment Request following installation of new safety related snubbers. The additional snubbers were added as a result of modifications to the scram discharge volume during the last outage.

Additional changes requested in our February 15, 1983 License Amendment Request are also shown on Exhibit B, pages 131, 132, and 132a to avoid confusion.

5. Addition of Hydraulic Snubbers (continued)

Significant Hazards Evaluation

The proposed additions to the list of safety related snubbers constitute additional limitations, restrictions, and controls not presently included in the Technical Specifications.

For this reason, operation of the Monticello Nuclear Generating Plant in accordance with the proposed change will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

6. Additional Accident Monitoring Instrumentation

Proposed Changes

Add new limiting conditions for operation and surveillance requirements for high range effluent monitors, containment high range monitor, containment wide range pressure monitor, wide range containment water level monitors, and containment hydrogen and oxygen monitors. Refer to Exhibit B, pages 229 b through 229d. Renumber existing pages 229d through 229s.

Reason for Changes

These changes implement the requirements of NUREG-0737, Items II.F.1.1 through II.F.1.6. Guidance provided in a letter dated December 7, 1981 from Mr Thomas A Ippolito, Chief, Operating Reactors Branch #2, Division of Licensing, USNRC was used in the preparation of these changes.

Plant modifications to install additional accident monitoring instrumentation required by NUREG-0737, Items II.F.1.1 through II.F.1.6 were described in our letters dated December 30, 1980, December 31, 1981, and May 24, 1982.

Significant Hazards Evaluation

The proposed additions to the Technical Specifications related to new accident monitoring instrumentation constitute additional limitations, restrictions, and controls not presently included.

6. Additional Accident Monitoring Instrumentation (continued)

Significant Hazards Evaluation (continued)

For this reason, operation of the Monticello Nuclear Generating Plant in accordance with the proposed change will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EXHIBIT B

License Amendment Request dated June 24, 1983

Docket No. 50-263
License No. DPR-22

Exhibit B consists of revised pages for the Monticello Nuclear Generating Plant Technical Specifications as listed below:

i, vi, vii
46
50
60b
61
69
71
111
111a
112
131
132
132a
229b
229c
229d
229e - 229t (renumber of pages only)
233
233a
249a

Note: To avoid confusion, certain changes requested in our License Amendment Request dated February 15, 1983 are also shown on the Exhibit B pages.

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REV

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Emergency Core Cooling Subsystems Actuation When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2. C. Control Rod Block Actuation The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3. D. Other Instrumentation Whenever the reactor is in the RUN Mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.7 shall be met.	

TABLE 3.2.1 - Continued

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions
b. High Drywell Pressure (5)	≤ 2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	≥ 10'6" above the top of the active fuel	2	2	E
b. High Drywell Pressure	≤ 2 psig	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	≤150,000 lb/hr with ≤60 second time delay	2(4)	2	F
b. HPCI High Steam Flow	≤300,000 lb/hr	2(4)	2	F
c. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	≤45,000 lb/hr with 5 ± 2 sec time delay	2(4)	2	G
b. RCIC Steam Line Area	≤200°F	16(4)	16	G

Table 3.2.7

Other Instrumentation

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
A. Turbine/Feedwater Trip					
1. High Reactor Level	≤ 14'6" above top of active fuel	2	2	2	A
B. RCIC Initiation					
1. Low-Low Reactor Level	≥6'6" & ≤6'10" above top of active fuel	1	2	2	B
C. HPCI/RCIC Turbine Shutdown					
a. High Reactor Level	≤ 14'6" above top of active fuel	1	2	2	A
D. HPCI/RCIC Turbine Suction Transfer					
e. Condensate Storage Tank Low Level	≥2' 0" above tank bottom	1	2	2	C

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:
   a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition (Turbine/Feedwater Trip only), or
   b. Place the plant under the specified required condition using normal operating procedures.

* Required conditions when minimum conditions for operation are not satisfied:

A. Reactor in Startup, Refuel, or Shutdown Mode.
B. Comply with Specification 3.5.F.2.
C. Align HPCI and RCIC suction to the suppression pool. Restore channels to operable status status within 30 days or place the plant in Required Condition A.

Table 4.2.1
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level (Note 7)	once/month	Once/3 months	Once/day
2. Drywell High Pressure (Note 7)	once/month	Once/3 months	None
3. Reactor Low Pressure (Pump Start)	Note 1	Once/3 months	None
4. Reactor Low Pressure (Valve Permissive)	Note 1	Once/3 months	None
5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Note 1	Once/3 months	None
7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
8. Condensate Storage Tank Level	Refueling Outage	Refueling Outage	None
9. Reactor High Water Level	Once/month	Once/3 months	None
ROD BLOCKS			
1. APRM Downscale	Notes (1,5)	Once/3 months	None
2. APRM Flow Variable	Notes (1,5)	Once/3 months	None
3. IRM Upscale	Notes (2,5)	Note 2	Note 2
4. IRM Downscale	Notes (2,5)	Note 2	Note 2
5. RBM Upscale	Notes (1,5)	Once/3 months	None
6. RBM Downscale	Notes (1,5)	Once/3 months	None
7. SRM Upscale	Notes (2,5)	Note 2	Note 2
8. SRM Detector not in Start-up Position	Note 2	Note 2	Note 2
9. Scram Discharge Volume-High Level	Once/3 months	Refueling outage	None
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	Note 1	Once/3 months	Once/day

Bases Continued:

increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

The feedwater/turbine trip is actuated by a high reactor water level signal. This trip mitigates the feedwater controller failure transient by tripping the feedwater pumps and the turbine.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1 through 3.2.7, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the amount shown in Table 4.2.2.

Table 4.2.2- Continued
Trip Function and Deviations

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive)	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale IRM Upscale	-2/125 of Scale +2/125 of Scale
	APRM Downscale APRM Upscale	-2/125 of Scale See Basis 2.3
	RBM Downscale RBM Upscale Scram Discharge Volume-High Level	-2/125 of Scale Same as APRM Upscale + 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure Low Reactor Water Level	+ 12 psi -3 Inches
Other Instrumentation	High Reactor Water Level Low-Low Reactor Water Level Low Condensate Storage Level	+ 6 inches - 3 inches - 6 inches

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

4.2 BASES

shutdown shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.

F. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in 3.5.F.2 below, the RCIC system shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel. To be considered operable, the RCIC system shall meet the following conditions:

   a. The RCIC shall be capable of delivering 400 gpm into the reactor vessel at 150 psig.

   b. The controls for automatic transfer of the RCIC pump suction from the condensate storage tank to the suppression chamber shall be operable.

   c. The controls for automatic restart on subsequent low reactor level after it has been terminated by a high reactor level signal shall be operable.*

*Effective beginning of cycle 11

F. Surveillance of Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

Item	Frequency
Pump operability	Once/month
Motor operated valve operability	Once/month
Flow rate test	After major pump maintenance and every three months
Simulated automatic actuation, transfer of suction to suppression pool, and automatic restart on subsequent low reactor water level*	Once/Operating Cycle

*Demonstration of automatic restart required beginning of cycle 11.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. From and after the date that the RCIC system is made or found to be inoperable for any reason, except automatic transfer of pump suction, reactor operation is permissible only during the succeeding 15 days unless such system is sooner made operable, provided that during such 15 days all active components of the HPCI system are operable. With the controls for automatic transfer of pump suction inoperable, operation for up to 30 days is permissible if the pump suction is aligned to the suppression pool. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor pressure reduced to 150 psig within 24 hours.	2. When it is determined that the RCIC system is inoperable, the HPCI system shall be demonstrated to be operable immediately and daily thereafter.

3.0 LIMITING CONDITIONS FOR OPERATION

G. Minimum Core and Containment Cooling System Availability

1. During any period when one of the standby diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the low pressure core cooling and containment cooling subsystems connected to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

4.0 SURVEILLANCE REQUIREMENTS

G. Surveillance of Core and Containment Cooling System

1. When it is determined that one of the standby diesel generators is inoperable, all low pressure core cooling and containment cooling service water systems connected to the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
PS1-H2	MAIN STEAM	DRYWELL	953	071	I
PS1-H3	MAIN STEAM	DRYWELL	950	148	I
PS2-H2	MAIN STEAM	DRYWELL	950	120	I
PS3-H2	MAIN STEAM	DRYWELL	950	240	I
PS4-H3	MAIN STEAM	DRYWELL	950	212	I
RV24-H3	SAFETY-RELIEF	DRYWELL	950	110	I
RV24-H4	SAFETY-RELIEF	DRYWELL	935	100	I
RV24-H4A	SAFETY-RELIEF	DRYWELL	935	100	I
RV24-H5	SAFETY-RELIEF	DRYWELL	935	110	I
RV24-NS2	SAFETY-RELIEF	DRYWELL	934	081	I
RV24-NS3	SAFETY-RELIEF	DRYWELL	962	090	I
RV24-N1	SAFETY-RELIEF	DRYWELL	953	090	I
RV24A-H4A	SAFETY-RELIEF	DRYWELL	947	048	I
RV24A-H7	SAFETY-RELIEF	DRYWELL	953	088	I
RV24A-H8	SAFETY-RELIEF	DRYWELL	939	032	I
RV24A-NS1	SAFETY-RELIEF	DRYWELL	952	050	I
RV24A-NS2	SAFETY-RELIEF	DRYWELL	952	055	I
RV24A-N1	SAFETY-RELIEF	DRYWELL	956	086	I
RV25-H1	SAFETY-RELIEF	DRYWELL	953	180	I
RV25-H1A	SAFETY-RELIEF	DRYWELL	953	180	I
RV25-H2	SAFETY-RELIEF	DRYWELL	948	190	I
RV25-H2A	SAFETY-RELIEF	DRYWELL	948	190	I
RV25-H3	SAFETY-RELIEF	DRYWELL	934	180	I
RV25-NS1	SAFETY-RELIEF	DRYWELL	952	160	I
RV25-NS2	SAFETY-RELIEF	DRYWELL	952	195	I
RV25-N2	SAFETY-RELIEF	DRYWELL	956	159	I
RV25A-H2	SAFETY-RELIEF	DRYWELL	945	120	I
RV25A-H2A	SAFETY-RELIEF	DRYWELL	945	120	I
RV25A-H7	SAFETY-RELIEF	DRYWELL	953	135	I
RV25A-NS1	SAFETY-RELIEF	DRYWELL	934	110	I
RV25A-NS2	SAFETY-RELIEF	DRYWELL	934	102	I
RV25A-NS3	SAFETY-RELIEF	DRYWELL	952	122	I
RV26-H1	SAFETY-RELIEF	DRYWELL	953	200	I
RV26-H1A	SAFETY-RELIEF	DRYWELL	953	200	I
RV26-H2	SAFETY-RELIEF	DRYWELL	947	200	I
RV26-H2A	SAFETY-RELIEF	DRYWELL	947	200	I
RV26-H3A	SAFETY-RELIEF	DRYWELL	935	200	I
RV26-N1	SAFETY-RELIEF	DRYWELL	956	200	I
RV26A-H2	SAFETY-RELIEF	DRYWELL	940	250	I
RV26A-H2A	SAFETY-RELIEF	DRYWELL	935	250	I
RV26A-NS1	SAFETY-RELIEF	DRYWELL	934	240	I
RV26A-NS2	SAFETY-RELIEF	DRYWELL	934	230	I
RV26A-N1	SAFETY-RELIEF	DRYWELL	950	250	I
RV26A-N2	SAFETY-RELIEF	DRYWELL	951	250	I
RV27-H1	SAFETY-RELIEF	DRYWELL	950	320	I
RV27-H1A	SAFETY-RELIEF	DRYWELL	950	230	I
RV27-H5	SAFETY-RELIEF	DRYWELL	945	270	I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
RV27-H6	SAFETY-RELIEF	DRYWELL	945	270	I
RV27-NS1	SAFETY-RELIEF	DRYWELL	934	250	I
RV27-NS2	SAFETY-RELIEF	DRYWELL	934	280	I
RV27-N1	SAFETY-RELIEF	DRYWELL	956	270	I
RV27A-H2A	SAFETY-RELIEF	DRYWELL	953	290	I
RV27A-H3	SAFETY-RELIEF	DRYWELL	953	290	I
RV27A-H9	SAFETY-RELIEF	DRYWELL	938	290	I
RV27A-NS1	SAFETY-RELIEF	DRYWELL	952	282	I
RV27A-NS2	SAFETY-RELIEF	DRYWELL	952	279	I
RV27A-NS3	SAFETY-RELIEF	DRYWELL	952	282	I
RV27A-N1	SAFETY-RELIEF	DRYWELL	956	270	I
R26-NS1	SAFETY-RELIEF	DRYWELL	952	200	I
SS-1	MAIN STEAM	DRYWELL	953	279	I
SS-1AR	RECIRCULATION	DRYWELL	922	315	I
SS-1BR	RECIRCULATION	DRYWELL	922	135	I
SS-11	FEEDWATER	DRYWELL	952	302	I
SS-12	FEEDWATER	DRYWELL	952	058	I
SS-13	FEEDWATER	DRYWELL	952	258	I
SS-14	FEEDWATER	DRYWELL	952	096	I
SS-17A	RHR	DRYWELL	964	072	I
SS-17B	RHR	DRYWELL	964	072	I
SS-18A	RHR	DRYWELL	964	288	I
SS-18B	RHR	DRYWELL	964	288	I
SS-19	RHR	DRYWELL	964	341	I
SS-2	MAIN STEAM	DRYWELL	953	081	I
SS-2AR	RECIRCULATION	DRYWELL	927	302	I
SS-2BR	RECIRCULATION	DRYWELL	927	122	I
SS-20	RHR	DRYWELL	964	019	I
SS-3	MAIN STEAM	DRYWELL	950	212	I
SS-3AR	RECIRCULATION	DRYWELL	927	328	I
SS-3BR	RECIRCULATION	DRYWELL	927	148	I
SS-4	MAIN STEAM	DRYWELL	950	148	I
SS-4AR(A)	RECIRCULATION	DRYWELL	934	302	I
SS-4AR(B)	RECIRCULATION	DRYWELL	934	323	I
SS-4BR(A)	RECIRCULATION	DRYWELL	934	120	I
SS-4BR(B)	RECIRCULATION	DRYWELL	934	149	I
SS-40	HPCI	MAIN STEAM CHASE			I
SS-5AR	RECIRCULATION	DRYWELL	941	315	I
SS-5BR	RECIRCULATION	DRYWELL	941	135	I
SS-6AR	RECIRCULATION	DRYWELL	953	261	I
SS-6BR	RECIRCULATION	DRYWELL	953	099	I
SS-7	MAIN STEAM	DRYWELL	953	240	I
SS-7AR	RECIRCULATION	DRYWELL	953	323	I
SS-7BR	RECIRCULATION	DRYWELL	953	032	I
SS-8	MAIN STEAM	DRYWELL	953	120	I
SS-8AR	RECIRCULATION	DRYWELL	927	270	I
SS-8BR	RECIRCULATION	DRYWELL	927	090	I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
CRD-H210	CRD	E SCRAM DISCH VOL	947		A
CRD-H211	CRD	E SCRAM DISCH VOL	947		A
CRD-H213	CRD	E SCRAM DISCH VOL	947		A
CRD-H227	CRD	W SCRAM DISCH VOL	945		A
CRD-H232	CRD	W SCRAM DISCH VOL	945		A
CRD-H244	CRD	W SCRAM DISCH VOL	947		A
CRD-H245	CRD	W SCRAM DISCH VOL	947		A
CRD-H247	CRD	W SCRAM DISCH VOL	947		A
SS-121	PCAC	TORUS ROOM	928	252	A
SS-184	PCAC	TORUS ROOM	928	261	A
SS-185	CGCS	TORUS ROOM	928	260	A
SS-187	CGCS	TORUS ROOM	928	264	A
SS-191	PCAC	TORUS ROOM	928	234	A
SS-21	RHR	TORUS FL LV - S WALL			A
SS-211	CGCS	TORUS ROOM	931	072	A
SS-22	RHR	TORUS FL LV - S WALL			A
SS-23	RHR	B RHR ROOM FL LV			A
SS-24	RHR	A RHR ROOM FL LV			A
SS-25	RHR	TORUS CATWK-SE WALL			A
SS-26	CORE SPRAY	B RHR ROOM FL LVL			A
SS-27	CORE SPRAY	B RHR ROOM FL LVL			A
SS-28A	CORE SPRAY	A RHR ROOM FL LVL			A
SS-28B	CORE SPRAY	A RHR ROOM FL LVL			A
SS-29	RHR	OVER N2 ANALYZER	954		A
SS-30	RHR	OVER N2 ANALYZER	954		A
SS-31	RHR	TORUS CATWK			A
SS-32A	RHR	A RHR ROOM - BY HX	916		A
SS-32B	RHR	A RHR ROOM - BY HX	916		A
SS-33	RHR	ABOVE TORUS			A
SS-34	RHR	ABOVE TORUS			A
SS-35	HPCI	HPCI ROOM - N WALL	912		A
SS-36A	HPCI	HPCI ROOM - FL LVL			A
SS-36B	HPCI	HPCI ROOM - FL LVL			A
SS-37	HPCI	HPCI ROOM - W WALL	905		A
SS-38A	RCIC	RCIC ROOM - W WALL	906		A
SS-38B	RCIC	RCIC ROOM - W WALL	906		A
SS-41	CORE SPRAY	ABOVE TORUS CATWK	927		A
SS-42	HPCI	ABOVE TORUS RING HDR	906		A

Table 3.14.1

Instrumentation for Accident Monitoring

Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions *
Reactor Vessel Fuel Zone Water Level	2	1	A, B
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2	1	A, C
Drywell Wide Range Pressure	2	1	A, B
Suppression Pool Wide Range Level	2	1	A, B
Drywell High Range Radiation	2	1	A, D
Drywell and Suppression Pool Hydrogen and Oxygen Monitor	2	1	A, B
Offgas Stack Wide Range Radiation	2	1	A, D
Reactor Bldg Vent Wide Range Radiation	2	1	A, D

* Required Conditions

A. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the <u>total</u> number of channels, either restore the inoperable channels to operable status within seven days, or prepare and submit a special report to the Commission pursuant to Technical Specification 6.7.B.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

Table 3.14.1 (continued)

Instrumentation for Accident Monitoring

* Required Conditions (continued)

B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours.

D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (A) above.

Table 4.14.1

Minimum Test and Calibration Frequency for
Accident Monitoring Instrumentation

Instrument Channel	Test (Note 1)	Calibration (Note 1)	Sensor Check (Note 1)
Reactor Vessel Fuel Zone Water Level Monitor	-	Once/Operating Cycle	Once/month (Note 3)
Safety/Relief Valve Position (Pressure Switches)	-	Once/Operating Cycle	Once/month (Note 2)
Safety/Relief Valve Position (Thermocouples)	-	Once/Operating Cycle	Once/month (Note 2)
Drywell Wide Range Pressure Monitors	-	Once/Operating Cycle	Once/month
Suppression Pool Wide Range Level Monitors	-	Once/Operating Cycle	Once/month
Drywell High Range Radiation Monitors	-	Once/Operating Cycle	Once/month
Drywell and Suppression Pool Hydrogen and Oxygen Monitors	-	Once/Operating Cycle	Once/month
Offgas Stack Wide Range Radiation Monitors	-	Once/Operating Cycle	Once/month
Reactor Bldg Wide Range Radiation Monitors	-	Once/Operating Cycle	Once/month

Notes:

(1) Functional tests, calibrations, and sensor checks are not required when the instruments are not required to be operable. If tests are missed, they shall be performed prior to returning the instruments to an operable status.

(2) Proper instrument response shall be verified during each safety/relief valve actuation.

(3) These instruments are off-scale high during normal plant operation.

Bases:

3.14/4.14 The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Learned Task Force Status Report and Short Term Recommendations".

3.0 LIMITING CONDITIONS FOR OPERATION

3.15 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 granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

4.15 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 granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Bases 3.15 and 4.15

The inservice inspection program for the Monticello plant conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components classified into NRC Quality Groups A, B, and C conforms to the requirements of ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code. If a Code required inspection is impractical for the Monticello facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

A program of inservice testing of Quality Group A, B, and C pumps and valves is also in effect at the Monticello plant. Technical Specifications related to this program will be issued following NRC review and approval of the pump and valve testing program.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

4.16 RADIATION ENVIRONMENTAL MONITORING PROGRAM

Applicability

Applies to the periodic monitoring and recording of radioactive effluents found in the plant environs.

Objective

To provide for measurement of radiation levels and radioactivity in the site environs on a continuing basis.

Specification

A. Sample Collection & Analysis

1. The Radiation Environmental Monitoring Program given in Table 4.16.1 shall be conducted. Radioanalysis shall be conducted meeting the requirements of Table 4.16.2.

   A map and a table identifying the locations of the sampling points shall be provided in the Offsite Dose Calculation Manual (ODCM).

2. Whenever the Radiation Environmental Monitoring Program is not being conducted as specified in Table 4.16.1, the Annual Radiation Environmental Monitoring Report shall include a description of the reasons for not conducting the program as required and plans for preventing a recurrence.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunction of automatic sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.

4. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 4.16.3 when averaged over any calendar quarter, in lieu of any other report, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.7.C.2.a. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots > 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.8.A.2, 3.8.B.2, or 3.8.B.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiation Environmental Monitoring Report.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	5. Although deviations from the required sampling schedule are permitted under Item 3, above, whenever milk or leafy green vegetables can no longer be obtained from the designated sample locations required by Table 4.16.1, the Semi-annual Radioactive Effluent Release Report for this period shall explain why the samples can no longer be obtained and will identify the locations which will be added to and deleted from the monitoring program as soon as practicable. B. Land Use Census 1. A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft^2 producing fresh leafy vegetables, in each of the 16 meteorological sectors within a distance of five miles. The census shall also identify the locations of all milk animals and all 500 ft^2 or greater gardens producing broad leaf vegetation in each of the meteorological sectors within a distance of three miles. This census shall be conducted at least once per year between the dates of May 1 and October 31 by door to door survey, aerial survey, or by consulting local agricultural authorities associations.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

2. With a land use census identifying a location which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 4.16.A.1, the Semiannual Radioactive Effluent Release Report for this period shall identify the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

C. Interlaboratory Comparison Program

1. Analyses shall be performed on radioactive materials supplied as part of an NRC approved interlaboratory comparison program as described in the ODCM.

2. The results of analyses performed as a part of the above required program shall be included in the Annual Radiation Environmental Monitoring Report. When required analyses are not performed, corrective action shall be reported in the Annual Radiation Environmental Monitoring Report.

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
1. Airborne Radioiodine & Particulates	Samples from 5 locations: 3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q, 1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q, and 1 sample from a control location 8-20 miles distance and in the least prevalent wind direction	Continuous Sampler operation with sampler collection weekly.	Radioiodine analysis Weekly for I-131 Particulate: Gross beta activity on each filter weekly*. Analyses shall be performed more than 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample quarterly.
2. Direct Radiation	37 TLD stations established with duplicate dosimeters placed at the following locations:	Quarterly	Gamma Dose quarterly

* If gross beta activity in any indication sample exceeds 10 times the yearly average of the control sample, a gamma isotopic analysis is required.

** Sample locations are given on the figure and table in the ODCM.

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
2. Direct Radiation (Con't.)			
	1. Using the 16 meterological wind sectors as guidelines, an inner ring of stations in the general area of the site boundary is established and an outer ring of stations at 4 to 5 mile distance from the plant site is established. Because of inacessibility, two sectors in the inner and outer rings are not covered.		
	2. Seven dosimeters are established at special interest areas and a control station.		
3. Waterborne			
a. Surface	Upstream & downstream locations	Monthly composite of weekly samples (water & ice conditions permitting)	Gamma Isotopic analysis of each monthly composite Tritium analysis of quarterly composites of monthly composites

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 3 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
3. Waterbourne (con't.)			
b. Ground	Three samples from wells within 5 miles of the plant site and one sample from a well greater than 10 miles from the plant site.	Quarterly	Gamma Isotopic and tritium analyses of each sample
c. Drinking	One sample from the City of Minneapolis water supply	Monthly composite of weekly samples.	I-131 Analysis and Gross beta and Gamma isotopic analysis of each monthly composite Tritium analysis of quarterly composites of monthly composites
d. Sediment from Shoreline	One sample upstream of plant, one sample downstream of plant, and one sample from shoreline of recreational area	Semiannually	Gamma isotopic analysis of each sample

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 4 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
4. Ingestion			
a. Milk	One sample from dairy farm having highest D/Q, one sample from each of three dairy farms calculated to have doses from I-131 >1 mrem/yr, and one sample from 10-20 miles	Monthly or biweekly if animals are on pasture	Gamma isotopic and I-131 analysis of each sample
b. Fish and Invertebrates	One sample of one game specie of fish located upstream and downstream of the plant site. One sample of Invertebrates upstream and downstream of the plant site.	Samples collected semi-annually	Gamma isotopic analysis on each sample (edible portion only on fish).

** Sample locations are given on the figure and table in the ODCM.

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products	One sample of corn from highest D/Q farm and one sample from 10-20 miles	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of potatoes from highest D/Q farm and one sample from 10-20 miles	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of broad leaf vegetation from highest D/Q garden and one sample from 10-20 miles	At time of harvest	I-131 analysis of edible portion of each sample

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.2
(Page 1 of 2)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,e]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4[b]	1 x 10-2				
^{3}H	2000(1000[b])					
^{54}Mn	15		130			
^{59}Fe	30		120			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr-Nb	15[c]					
^{131}I	1[b, d]	7×10^{-2}		1[d]	60	
$^{134,137}Cs$	15(10[b]), 18	1×10^{-2}	130	15	60	150
^{140}Ba-La	15[c]			15[c]		

TABLE NOTATION

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the a priori lower limit of detection as defined above (as picocurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute). Typical values of E, V, Y and Δt shall be used in the calculations.

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22 is the number of transformations per minute per picocurie

Y is the fraction radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting

b - LLD for drinking water.

c - Total for parent and daughter.

d - Applies to specific isotope analysis-not to gamma spectrum analyses.

e - Other peaks which are measurable and identifiable, together with the radionuclides in TAble 4.15.2, shall be identified and reported.

Table 4.16.3

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Vegetables (pCi/kg, wet)
H-3	2×10^{4} (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^{2} (b)				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^{2} (b)			3×10^{2} (b)	

a - For drinking water samples
b - Total for parent and daughter

3.16 and 4.16 BASES

A. Sample Collection & Analysis

The Radiation Environmental Monitoring Program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the plant operation. This program thereby supplements the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. After a specific program has been in effect for at least three years of operation, program changes may be initiated based on this experience.

The detection capabilities required by Table 4.15.2 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirement of 40CFR 141.

B. Land Use Census

This specification is provided to ensure that changes in the use of off site areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

C. Interlaboratory Comparison Program

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as a part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonable valid.

E. A training program for the fire brigade shall be maintained under the direction of a designated member of Northern States Power management. This program shall meet the requirements of Section 27 of the NFPA Code - 1976 with the exception of training scheduling. Fire brigade training shall be scheduled as set forth in the training program.

F. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:

1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

    a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

    b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exeed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, not more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.

    c. A break of at least eight hours including shift turnover time should be allowed between work periods.

    d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

e. Shift Technical Advisor (STA) and Shift Emergency Coordinator (SEC) on-site rest time periods shall not be considered as hours worked when determining the total work time for which the above limitations apply.

2. Any deviation from the above guidelines shall be authorized by the Plant Manager or designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. During plant emergencies the Emergency Director shall have this authority. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not allowed.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the general public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The radioactive effluent release reports shall include the following information for solid waste shipped offsite during the report period.

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate).

c. principal radionuclides (specify whether determined by measurement or estimate),

d. type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site of radioactive materials in gaseous and liquid effluents on a quarterly basis, changes to the ODCM, a description of changes to the PCP, a report of when milk or vegetable samples can not be obtained as required by Table 4.16.1, and changes in land use resulting in significant increases in calculated doses.

5. Annual Summaries of Meteorological Data

An annual summary of meteorological data shall be submitted for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability at the request of the Commission.

6. Report of Safety/Relief Valve Failures and Challenges. An annual report of safety/relief valve failures and challenges shall be submitted prior to March 1st of each year.