DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat losses and additions.

1.5.7 Staggered Test Basis

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \times \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door of the personnel hatch and the emergency hatch is closed and sealed during refueling or during personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required.

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and three channels each of the following are operable: reactor coolant temperature, reactor coolant pressure, pressure-temperature, flux-imbalance flow, power-number of pumps, and high reactor building pressure. The engineered safety features actuation system must have three analog channels and two digital channels functioning correctly prior to a startup. Additional operability requirements are provided by Technical Specifications 3.1.12 and 3.4 for equipment which is not part of the RPS or ESFAS.

Operation at rated power is permitted as long as the systems have at least the minimum number of operable channels given in Column C (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

There are four reactor protective channels. A fifth channel that is isolated from the reactor protective system is provided as a part of the reactor control system. Normal trip logic is two out of four. The minimum number or operable channels required is three. While a bypassed channel is considered inoperable, a channel placed in the tripped condition is considered operable. Thus, only one channel may be placed in bypass at any one time in order to maintain the minimum number of required channels. This results in a trip logic of two out of three. It should be noted that an effective trip logic of one out of two can be achieved by placing one channel in bypass and one channel in the tripped condition.

The four reactor protective channels are provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protective system bypass switch key permitted in the control room. That key will be under the administrative control of the Shift Supervisor. Spare keys will be maintained in a locked storage accessible only to the Station Manager.

Each reactor protective channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the Shift Supervisor. The use of a key operated shutdown bypass switch for on-line testing or maintenance during reactor power operation has no significance when used in conjunction with a key operated channel bypass switch since the channel trip relay is locked in the untripped state. The use of a key operated shutdown bypass switch alone during power operation will cause the channel to trip. When the shutdown bypass switch is operated for on-line testing or maintenance during reactor power operation, reactor power and RCS pressure limits as specified in Table 2.3-1 are not applicable.

The source range and intermediate range nuclear instrumentation overlap by one decade of neutron flux. This decade overlap will be achieved at 10^{-10} amps on the intermediate range instrument.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Each voltage source and its associated breakers and SCR control relays comprise a trip system. Thus, the two trip systems and their associated trip devices form a 1-out-of-2 logic used twice which is referred to as a 1-out-of-2x2 logic.

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) (Deleted)

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown within 24 hours.

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

(g) (Deleted)

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCE

(1) FSAR, Section 7.2.3.4.

(2) BAW-10167A, "Justification for Increasing the Reactor Trip System On-line Test Interval."

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breakers, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Intermediate Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	45 Days STB	RF	
8. High Reactor Coolant Pressure	ES	45 Days STB	RF	
9. Low Reactor Coolant Pressure	ES	45 Days STB	RF	
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	RF	
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	45 Days STB	RF	
13. High Reactor Building Pressure	DA	45 Days STB	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	45 Days STB	RF	
b) Discharge Pressure Switches	NA	45 Days STB	RF	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	
54. Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II.F.1.3

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
55. Containment Pressure Monitor (PT-230,231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80, -81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	RF	RF	
59. Reactor Vessel Head Level	NA	RF	RF	
60. Core Exit Thermocouples	MO	NA	RF	
61. Subcooling Monitors	MO	RF	RF	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not applicable
RF - Refueling Outage
STB - STAGGERED TEST BASIS

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL JUSTIFICATION AND
NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

TECHNICAL JUSTIFICATION
NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Background:

Reference 1 provided the NRC Safety Evaluation Report (SER) for B&WOG Topical Report BAW 10167 and Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval." Reference 2 provided the SER for Supplement 2 to BAW 10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval - Additional Information on Allowed Outage Times." The accepted version of the topical report (BAW-10167A) was submitted to the NRC July 17, 1992 (J. H. Taylor, BWNT to NRC Document Control Desk). The BAW-10167A justifies extension of the RPS test interval to 45 days on a staggered test basis and provides for the indefinite bypass of one RPS channel. The proposed revision to TS provided in Attachment 1:

- Extends the frequency for RPS instrument channel tests in Table 4.1-1 from monthly to 45 days on a staggered test basis;
- Removes the limitation provided in Table 3.5.1-1 on placing one RPS channel in bypass and one channel in trip;
- Adds the definition of "staggered test basis" to Section 1.5; and
- Revises the associated Bases.

Instrument Drift Data:

Page 12 of reference 1 requires that each licensee confirm that they have reviewed drift information including as found and as left values for each instrument channel involved and have determined that drift occurring in that channel over the period of the extended surveillance test interval will not cause the setpoint value to exceed the allowable values as calculated for that channel by their setpoint methodology. In addition, each licensee should maintain onsite records showing the actual setpoint calculations and supporting data that are available for planned future NRC staff audits. This data should consist of monthly information taken over an extended period of time (approximately 2 - 3 years).

Drift information including as found and as left values for each RPS instrument channel has been reviewed. This review has determined that drift occurring in the channel over the 180 day test interval will not cause the setpoint value to exceed the allowable values for that channel.

Discussion of Changes:

The following summarizes TS and Bases revisions in Attachment 1:

TS 1.5, page 1-4:
Currently, the Oconee Technical Specifications do not include a definition of "staggered test basis." The analyses provided in BAW-10167A are based

on performing the channel test of one of the four RPS channels every 45 days, rather than all four channels every 180 days (e.g., 45 days on a staggered test basis). Therefore, the definition of staggered test basis provided in Reference 3 is added as TS 1.5.7 consistent with revisions to Table 4.1-1 "Instrument Surveillance Requirements". This change is considered to be administrative in nature.

TS Table 3.5.1-1, page 3.5-5c

Currently NOTE (a) allows one RPS channel to be placed in bypass and one channel to be placed in trip, leaving an effective one out of two trip logic for a maximum of four hours. The analyses provided in BAW-10167 justify removal of this restriction (see Reference 3, LCO 3.3.1 Condition B).

3.5.1 Bases, page 3.5-2

The Bases for TS 3.5.1 have been revised consistent with the changes to Table 3.5.1-1 NOTE (a).

TS Table 4.1-1, pages 4.1-3, -4, -8, -8a

Currently, RPS instrument channel tests are required to be performed monthly. The analyses provided in BAW-10167A justify performing the RPS instrument channel tests on a 45 day staggered test basis.

4.1 Bases, page 4.1-2

The Bases for TS 4.1 have been updated to include a reference to BAW-10167A. This change is considered to be administrative in nature.

Evaluation:

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10CFR50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

Each accident analysis addressed within the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the change proposed within this amendment request. The probability of any Design Basis Accident (DBA) is not affected by this change, nor are the consequences of a DBA affected by this change since extension of the RPS on-line test interval and removal of limitations on placing one RPS channel in trip and one RPS channel in bypass based on an NRC approved Topical Report are not considered to be an initiator or contributor to any accident analysis addressed in the Oconee FSAR. Plant specific provisions of the associated NRC SER regarding drift data have been met. The probability of any DBA is not affected by this change, nor are the consequences of a DBA affected by this change since addition of the definition of "staggered test basis" is not considered to be an initiator or contributor to any accident analysis accident analysis addressed in the Oconee FSAR.

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

Operation of ONS in accordance with these Technical Specifications will not create any failure modes not bounded previously evaluated accidents. Consequently, this change will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

The Technical Specifications will continue to require the RPS trip setpoints remain within the assumptions of the accident analysis, thus preserving existing margins of safety. Therefore, there will be no significant reduction in any margin of safety.

Duke has concluded based on the above that there are no significant hazards considerations involved in this amendment request.

Environmental Impact Statement

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.

REFERENCES:

1. NRC Evaluation of BWOG Topical Report BAW-10167 and Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval." A. C. Thadani, NRC to C. W. Smythe, B&WOG; December 5, 1988

2. NRC Evaluation of BWOG Topical Report BAW-10167 Supplement 2, "Justification for Increasing the Reactor Trip System On-Line Test Interval - Additional Information on Allowed Outage Time." A. C. Thadani, NRC to B. P. Wunderly, B&WOG; July 8, 1992

3. NUREG 1430 "Standard Technical Specifications for Babcock and Wilcox Plants"

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 3

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat losses and additions.

1.5.7 Staggered Test Basis

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \quad x \quad \frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1$$

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door of the personnel hatch and the emergency hatch is closed and sealed during refueling or during personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required.

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and three channels each of the following are operable: reactor coolant temperature, reactor coolant pressure, pressure-temperature, flux-imbalance flow, power-number of pumps, and high reactor building pressure. The engineered safety features actuation system must have three analog channels and two digital channels functioning correctly prior to a startup. Additional operability requirements are provided by Technical Specifications 3.1.12 and 3.4 for equipment which is not part of the RPS or ESFAS.

Operation at rated power is permitted as long as the systems have at least the minimum number of operable channels given in Column C (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

There are four reactor protective channels. A fifth channel that is isolated from the reactor protective system is provided as a part of the reactor control system. Normal trip logic is two out of four. The minimum number or operable channels required is three. While a bypassed channel is considered inoperable, a channel placed in the tripped condition is considered operable. Thus, only one channel may be placed in bypass at any one time in order to maintain the minimum number of required channels. This results in a trip logic of two out of three. It should be noted that~~, for a limited period of time,~~ an effective trip logic of one out of two can be achieved by placing one channel in bypass and one channel in the tripped condition.

The four reactor protective channels are provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protective system bypass switch key permitted in the control room. That key will be under the administrative control of the Shift Supervisor. Spare keys will be maintained in a locked storage accessible only to the Station Manager.

Each reactor protective channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the Shift Supervisor. The use of a key operated shutdown bypass switch for on-line testing or maintenance during reactor power operation has no significance when used in conjunction with a key operated channel bypass switch since the channel trip relay is locked in the untripped state. The use of a key operated shutdown bypass switch alone during power operation will cause the channel to trip. When the shutdown bypass switch is operated for on-line testing or maintenance during reactor power operation, reactor power and RCS pressure limits as specified in Table 2.3-1 are not applicable.

The source range and intermediate range nuclear instrumentation overlap by one decade of neutron flux. This decade overlap will be achieved at 10^{-10} amps on the intermediate range instrument.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Each voltage source and its associated breakers and SCR control relays comprise a trip system. Thus, the two trip systems and their associated trip devices form a 1-out-of-2 logic used twice which is referred to as a 1-out-of-2x2 logic.

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic ~~for a maximum of four hours.~~

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) (Deleted)

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown within 24 hours.

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

(g) (Deleted)

(h) The RCP monitors provide input to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCE

(1) FSAR, Section 7.2.3.4.

(2) BAW-10167A, "Justification for Increasing the Reactor Trip System On-line Test Interval."

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breakers, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	~~MO~~ 45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Intermediate Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	~~MO~~ 45 Days STB	RF	
8. High Reactor Coolant Pressure	ES	~~MO~~ 45 Days STB	RF	
9. Low Reactor Coolant Pressure	ES	~~MO~~ 45 Days STB	RF	
10. Flux-Reactor Coolant Flow Comparator	ES	~~MO~~ 45 Days STB	RF	
11. Reactor Coolant Pressure Temperature Comparator	ES	~~MO~~ 45 Days STB	RF	

12/11/87
Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	~~MO~~ 45 Days STB	RF	
13. High Reactor Building Pressure	DA	~~MO~~ 45 Days STB	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	~~MO~~ 45 Days STB	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	~~MO~~ 45 Days STB	RF	
b) Discharge Pressure Switches	NA	~~MO~~ 45 Days STB	RF	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	
54. Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II.F.1.3

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	RF	RF	
59. Reactor Vessel Head Level	NA	RF	RF	
60. Core Exit Thermocouples	MO	NA	RF	
61. Subcooling Monitors	MO	RF	RF	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage
STB - Staggered Test Basis

Amendment No. 174 (Unit 1)
Amendment No. 174 (Unit 2)
Amendment No. 171 (Unit 3)
6/6/89