ATTACHMENT A

1. Remove pages 3/4 2-14, 3/4 2-15, 3/4 2-16, 3/4 3-2, 3/4 3-3, 3/4 3-4, 3/4 3-7, 3/4 3-8, 3/4 3-15, 3/4 3-18, 3/4 3-19, 3/4 3-20, 3/4 3-21, 3/4 6-5, 3/4 6-10, 3/4 8-6, 6-2

2. Insert pages 3/4 3-2, 3/4 3-3, 3/4 3-4, 3/4 3-7, 3/4 3-8, 3/4 3-15, 3/4 3-18, 3/4 3-19, 3/4 3-20, 3/4 3-21, 3/4 6-5, 3/4 6-5a, 3/4 6-10, 3/4 8-6, 6-2

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TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | Manual Reactor Trip | 2 | 1 | 2 | 1, 2 and * | 12 |
| 2. | Power Range, Neutron Flux (1) | 4 | 2 | 3 | 1, 2 | 2 |
| 3. | Power Range, Neutron Flux High Positive Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 4. | Power Range, Neutron Flux, High Negative Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 5. | Intermediate Range, Neutron Flux (2) | 2 | 1 | 2 | 1, 2 and * | 3 |
| 6. | Source Range, Neutron Flux (3) | | | | | |
| | A. Startup | 2 | 1 | 2 | 2# and * | 4 |
| | B. Shutdown | 2 | 0 | 1 | 3, 4 and 5 | 5 |
| 7. | Overtemperature $\Delta T$ | | | | | |
| | Three Loop Operation | 3 | 2 | 2 | 1, 2 | 2 |
| | Two Loop Operation | 3 | 1** | 2 | 1, 2 | 9 |
| 8. | Overpower $\Delta T$ | | | | | |
| | Three Loop Operation | 3 | 2 | 2 | 1, 2 | 2 |
| | Two Loop Operation | 3 | 1** | 2 | 1, 2 | 9 |
| 9. | Pressurizer Pressure-Low (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |

# High Voltage to detector may be de-energized above P-6.

## TABLE 3.3-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 10. | Pressurizer Pressure--High | 3 | 2 | 2 | 1, 2 | 7 |
| 11. | Pressurizer Water Level--High (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |
| 12. | Loss of Flow - Single Loop (Above P-8) | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1 | 7 |
| 13. | Loss of Flow - Two Loops (Above P-7 and below P-8) | 3/loop | 2/loop in two operating loops | 2/loop each operating loop | 1 | 7 |
| 14. | Steam Generator Water[(4)] Level--Low-Low | 3/loop | 2/loop in any operating loops | 2/loop in each operating loop | 1, 2 | 7 |
| 15. | Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | 2/loop-level and 2/loop-flow mismatch | 1/loop-level coincident with 1/loop-flow mismatch in same loop | 1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch | 1, 2 | 7 |
| 16. | Undervoltage-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |
| 17. | Underfrequency-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |

TABLE 3.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 18. Turbine Trip(5) | | | | | |
| A. Auto Stop Oil Pressure | 3 | 2 | 2 | 1 | 7 |
| B. Turbine Stop Valve Closure | 4 | 4 | 4 | 1 | 8 |
| 19. Safety Injection Input(6) from ESF | 2 | 1 | 2 | 1, 2 | 1 |
| 20. Reactor Coolant Pump Breaker Position Trip (Above P-7) | 1/breaker | 2 | 1/breaker per operating loop | 1 | 11 |
| 21. Reactor Trip Breakers | 2 | 1 | 2 | 1, 2* | 1 |
| 22. Automatic Trip Logic | 2 | 1 | 2 | 1, 2* | 1 |

(1) Low setpoint block permitted by P-10

(2) Manual bypass permitted by P-10 or rack mounted bypass switches during testing

(3) Manual bypass permitted above P-6, blocked above P-10

(4) Automatically blocked by closing loop-stop valves

(5) Blocked below P-9

(6) Low Pressure Pressurizer Safety Injection block permitted by P-11
Low Steamline Pressure Safety Injection block permitted by P-11
Manual Block permitted after Safety Injection System Reset and P-4

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 10 - Not applicable.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

| DESIGNATION | CONDITION AND SETPOINT | FUNCTION |
|---|---|---|
| P-6 | 1/2 Neutron flux (intermediate range) above setpoint | Allows manual block of source range reactor trip |
| | 2/2 Neutron flux (intermediate range) below setpoint | Defeats the block of source range reactor trip |
| P-7 | 2/4 Neutron flux (power range) below setpoint (from P-10) or 1/2 Turbine impulse chamber pressure below setpoint (from P-13) | Blocks reactor trip on: Low flow or reactor coolant pump breakers open in more than two loops, undervoltage, underfrequency, pressurizer low pressure, and pressurizer high level |
| P-8 | 2/4 Neutron flux (power range) below setpoint | Blocks reactor trip on low flow |
| P-9 | 2/4 Neutron flux (power range) below setpoint | Blocks reactor trip on turbine trip |
| P-10 | 3/4 Neutron flux (power range) above setpoint | Allows manual block of power range (low setpoint) reactor trip |

TABLE 3.3-1 (continued)

| DESIGNATION | CONDITION AND SETPOINT | FUNCTION |
|---|---|---|
| | | Allows manual block of intermediate range reactor trip and intermediate range rod stops (C-1) |
| | | Blocks source range reactor trip (back-up for P-6) |
| | 3/4 Neutron flux (power range) below setpoint | Defeats the block of power range (low setpoint) reactor trip |
| | | Defeats the block of intermediate range reactor trip and intermediate range rod stops (C-1) |
| | | Input to P-7 |
| P-13 | 1/2 Turbine impulse chamber pressure above setpoint | Input to P-7 |
| P-14 | 2/3 Steam generator water level above setpoint on any steam generator | Closes all feedwater control valves |
| | | Trips all main feedwater pumps which closes the pump discharge valves |
| | | Actuates turbine trip |

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. SAFETY INJECTION AND FEEDWATER ISOLATION | | | | | |
| a. Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic[(1)] | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| c. Containment Pressure-High | 3 | 2 | 2 | 1, 2, 3 | 14 |
| d. Pressurizer Pressure - Low | 3 | 2 | 2 | 1, 2, 3# | 14 |
| e. Low Steamline Pressure | | | | | |
| Three Loops operating | 3/loop | 2/loop any loop | 2/loop any loop | 1, 2, 3# | 14 |
| Two loops operating | 3/loop | 2/loop any operating loop | 2/any operating loop | 1, 2, 3# | 15 |

[(1)] Manual block permitted after Safety Injection System Reset and P-4

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | | |
| a. Manual | 2/steam line | 1/steam line | 2/operating steam line | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| c. Containment Pressure--Intermediate-High-High | 3 | 2 | 3 | 1, 2, 3 | 14 |
| d. Low Steamline Pressure Three Loops (Loop Stop Operating Valves Open) | 3/loop | 2/loop Any loop | 2/loop Any loop | 1, 2, 3# | 14 |
| Two Loops Operating | 3/loop | 2/loop any operating loop | 2/any operating loop | 1, 2, 3# | 15 |
| e. High Steam Pressure Rate | 3/loop | 2/loop any loop | 2/operating loop | 3##, 4 | 18 |

## TABLE 3.3-3 (Continued)
## ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 5. TURBINE TRIP & FEEDWATER ISOLATION | | | | | |
| a. Steam Generator(2) Water Level--High-High | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1, 2, 3 | 14 |
| 6. LOSS OF POWER | | | | | |
| a. 4.16kv Bus 1. Loss of Voltage (trip feeder) | 1/4.16kv Bus | 1/4.16kv Bus | 1/4kv Bus | 1, 2, 3, 4 | 33 |
| 2. Loss of Voltage (start diesel) | 1/4.16kv Bus | 1/4.16kv Bus | 1/4kv Bus | 1, 2, 3, 4 | 33 |
| b. Grid Degraded Voltage (4.16kv Bus) | 2/4.16kv Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |
| c. Grid Degraded Voltage (480v Bus) | 2/480v Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |

(2) Administrative block permitted for two-loop operation

Table 3.3-3 (Continued)

TABLE NOTATION

# Trip function may be bypassed in this MODE below P-11.

## Trip function automatically bypassed above P-11, and is bypassed below P-11 when Safety Injection on low steam pressure is not manually bypassed.

### The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within six hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to two hours for surveillance testing per Specification 4.3.2.1.1., provided the other channel is operable.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT STANDBY within the following six hours.

b. Above P-11 and P-12, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the bypass condition; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT SHUTDOWN within the following 12 hours.

TABLE 3.3-3 (Continued)

b. Above P-11 or P-12, demonstrate that the Minimum Channels OPERABLE requirement is met witin 1 hour; operation may continue with the inoperable channel bypassed and one channel may be bypassed for up to 2 hours for testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and cold shutdown within the following 30 hours.

ACTION 33 - With the number of OPERABLE Channels one less than the Total Number of Channels, the Emergency Diesel Generator associated with the 4kv Bus shall be declared inoperable and the ACTION Statements for Specifications 3.8.1.1 or 3.8.1.2, as appropriate shall apply.

ACTION 34 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until the performance of the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour.

ENGINEERED SAFETY FEATURES INTERLOCKS

| DESIGNATION | CONDITON AND SETPOINT | FUNCTION |
|---|---|---|
| P-4 | Reactor Trip | Actuates turbine trip. Closes main feedwater valves on Tavg below setpoint. Prevents opening of main feedwater valves which were closed by safety injection or high steam generator water level. Allows manual block of the automatic reactuation of safety injection. |
| | Reactor not tripped | Defeats the block of the automatic reactuation of safety injection. |
| P-11 | 2/3 Pressurizer pressure below setpoint | Allows manual block of safety injection actuation on low pressurizer pressure signal. Blocks automatic opening of the power relief valves. |
| | 2/3 Pressurizer pressure above setpoint | Defeats the manual block of safety injection actuation. |
| P-12 | 2/3 Tavg channels below setpoint | Blocks steam dump below setpoint. Allows manual bypass of the steam dump block to cooldown condenser dump valves. |

## 3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

### A.C. DISTRIBUTION - OPERATING

#### LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators with tie breakers open between redundant busses:

| | |
|---|---|
| 4160 | volt Emergency Bus #1AE and 480V Emergency Bus 8N |
| 4160 | volt Emergency Bus #1DF and 480V Emergency Bus 9P |
| 120 | volt A.C. Vital Bus #1 |
| 120 | volt A.C. Vital Bus #II |
| 120 | volt A.C. Vital Bus #III |
| 120 | volt A.C. Vital Bus #IV |

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

CONTAINMENT STRUCTURAL INTEGRITY

## LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABLITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above reguirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

## SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Liner Plate and Concrete The structural integrity of the containment liner plate and concrete shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by:

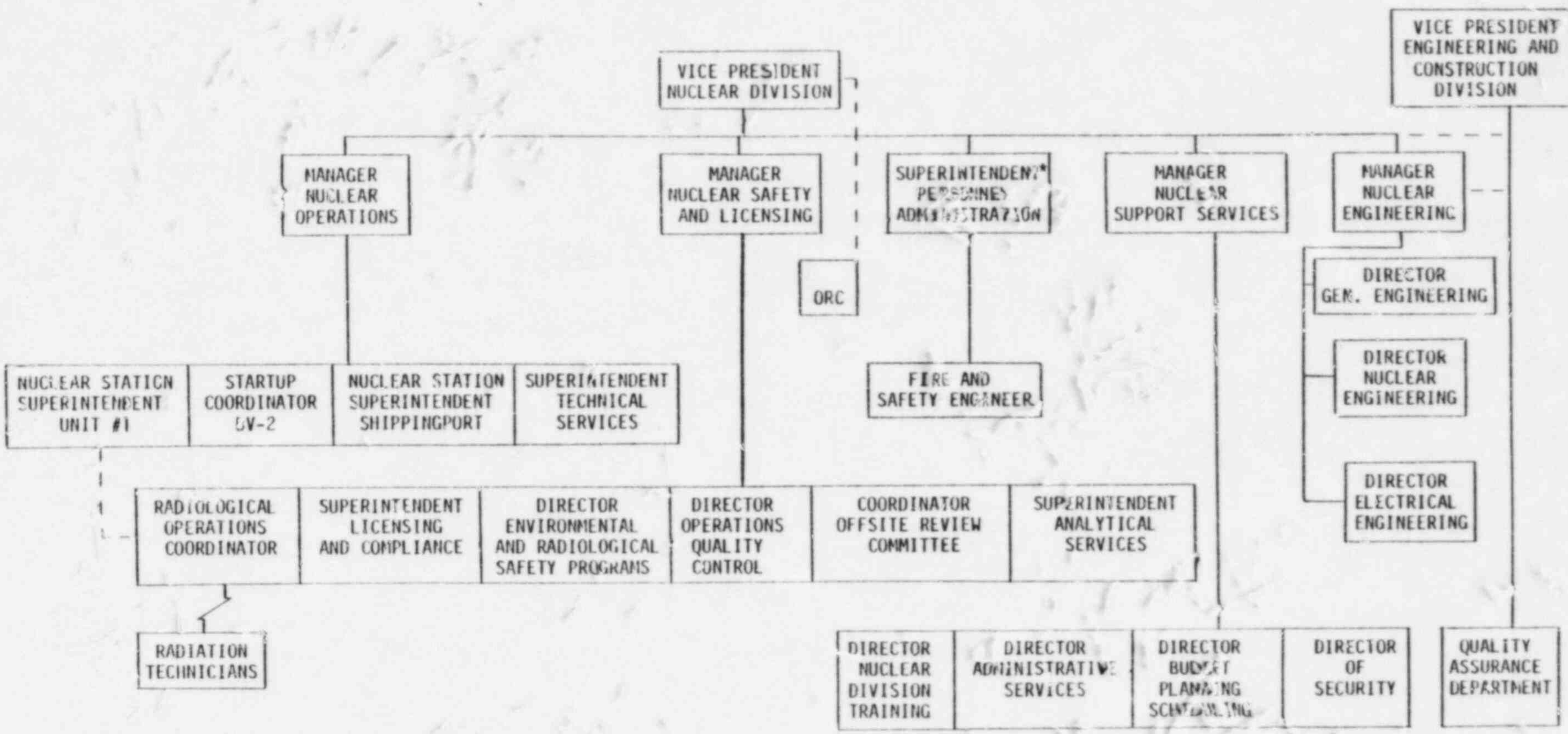
a. a visual inspection of the accessible surfaces and verifying no apparent changes in appearance or other abnormal degradation.

b. a visual inspection of accessible containment liner test channels prior to each Type A containment leakage rate test. Any containment liner test channel which is found to be damaged to the extent that channel integrity is impaired or which is discovered with a vent plug removed, shall be removed and a protective coating shall be applied to the liner in that area.

c. a visual inspection of the dome area prior to each Type A containment leakage rate test to insure the integrity of the protective coating. If a loss of integrity of the protective coating is observed, any vent plug to a test channel which may be in the area where the protective coating has failed shall be seal welded and then the protective coating shall be repaired.

4.6.1.6.2 Reports An initial report of any abnormal degradation of the containment structure detected during the above required tests and inspections shall be made within 10 days after completion of the surveillance requirements of this specification, and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion. This report shall include a description of the condition of the liner plate and concrete, the inspection procedure, the tolerances on cracking and the corrective actions taken.

DUQUESNE LIGHT COMPANY
Nuclear Division




* Indicates Fire Protection Responsibility

Figure 6.2-1
Offsite Organization (Partial)

PROPOSED WORDING

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 $L_a$ at $P_a$ (38.3 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain the associated OPERABLE air lock door closed and either restore the associated inoperable air lock door to OPERABLE status within 24 hours or lock the associated OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the associated OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With a containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

## SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each containment entry, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage when the gap between the door seals is pressurized to greater than or equal to $P_a$ (38.3 psig) for at least 2 minutes, or by quantifying the total air lock leakage to insure the requirements of 3.6.1.3.b are met.

b. By conducting overall air lock leakage tests at not less than $P_a$ (38.3 psig), and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months, # and

2. Upon completion of maintenance which has been performed on the air lock that could affect the air lock sealing capability*

c. At least once per 18 months during shutdown by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.
*Exemption to Appendix J of 10 CFR 50.

ATTACHMENT B

Safety Evaluation

Proposed Change Request No. 86 amends the Beaver Valley Power Station, Unit No. 1 Technical Specifications, Appendix A concerning various administrative changes.

Description and Purpose of Change

1. Section 3.2.6 - Axial Power Distribution, is being deleted in accordance with the NRC Safety Evaluation enclosed with Technical Specification Amendment No. 9.

2. Tables 3.3-1 and 3.3-3 have been revised to identify conditions under which operating bypasses will block the Reactor Trip System Instrumentation and Engineered Safety Feature actuation channels. The permissive descriptions have also been revised to conform to the Updated Final Safety Analysis Report (UFSAR) format.

3. Section 3.8.2.1 - Onsite Power Distribution Systems, has been revised to require the 4160 volt emergency bus and applicable 480 volt emergency bus to be operable.

4. Section 3.6.1.1 - Containment Systems, has been revised to correct a typographical error.

5. The offsite organization, Figure 6.2-1, has been revised to update the title from Director to Superintendent, Personnel Administration and to incorporate the organization under the Manager, Nuclear Engineering.

6. Section 3.6.1.3 - Containment Air Locks, has been amended by revising the Action and Surveillance Requirements. The change to the Action statement consists of a provision to allow continued plant operation with one air lock door inoperable, provided that the operable air lock door is locked closed and verified to be locked closed at least once per 31 days. The locking of an air lock will be achieved by: disconnecting the power source of the D.C. motor driven pump unit which supplies the hydraulically operated latches on the personnel air lock doors, and by physically locking the emergency egress air lock doors (with lock and key to the outside door and a physical constraint to the inside door.) The surveillance requirements have been revised to require essentially what is stated in the "Periodic Retest Schedule" of 10 CFR 50 Appendix J, Section III.D.2. Also included is a change to require conducting overall air lock leakage tests when maintenance has been performed on that air lock.

## Basis

1. Is the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR increased? No

Reason: The deletion of Section 3.2.6-Axial Power Distribution is an administrative change and is consistent with the NRC Safety Evaluation enclosed with Technical Specification Amendment No. 9. This change is not a safety concern, since it was only applicable to Cycle 1 operation.

The alteration of Table 3.3-1, Reactor Trip System Instrumentation and Table 3.3-3, Engineered Safety Feature Actuation System Instrumentation is administrative in nature and does not physically change plant safety related systems, components or structures. This change is being made in order to identify conditions under which operating bypasses will block the reactor trip or engineered safety feature actuation channels and will not affect the function of any equipment or systems important to safety as addressed in the UFSAR Section 7.2, Reactor Trip System or Section 7.3, Engineered Safety Features System. The change to Section 3.8.2.1 - Onsite Power Distribution Systems is administrative and will not affect any equipment or system as discussed in UFSAR Section 8.4, Station Service Systems or Section 8.5, Emergency Power System. The change to Section 3.6.1.6 - Containment Structural Integrity, is being made to correct a typographical error. The present specification references a non-existent section 4.6.1.7. This change is administrative and will not affect any equipment or system discussed in UFSAR Section 5.2, Containment Structure.

Figure 6.2-1, Offsite Organization, has been revised to reflect reorganization changes. This change is not a safety concern and does not affect the UFSAR.

The proposed changes made to Section 3.6.1.3 - Containment Air Locks, are administrative changes and do not involve any physical changes to the air locks or other plant equipment, nor do they degrade the leak tightness of the air locks. The change to the Surveillance Requirements remains essentially the same as 10 CFR 50 Appendix J Section III.D.2, periodic retest schedule of Type B test requirements. For these reasons, the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR Sections 5.2.2.5.5, 5.2.4.8 and 5.6 is not increased.

2. Is the probability for an accident or malfunction of a different type than previously evaluated in the Final Safety Analysis Report created? No

   Reason: The proposed changes are administrative in nature and do not physically change the plant safety related systems, components or structures, therefore, the changes will not create the possibility for a new type of accident or malfunction of a different type than any previously evaluated in the UFSAR sections addressed above or the accident analysis of Section 14.

3. Is the margin of safety as defined in the basis for any Technical Specification reduced? No

   Reason: The Technical Specification BASES for the sections addressed above will not be affected by the proposed changes, as none of the systems or components will be physically changed or their function altered in any way. Therefore, the margin of safety inherent in the applicable bases will not be reduced.

4. Based on the above, is an unreviewed safety question involved? No

## Conclusion

The proposed changes are administrative in nature and do not involve physical change to plant safety related systems, components or structures, will not increase the likelihood of a malfunction of safety related equipment, increase the consequences of an accident previously analyzed, nor create the possibility of a malfunction different than previously evaluated in the UFSAR. Therefore, it is concluded that, since the change does not involve an unreviewed safety question in accordance with 10 CFR 50.59, the proposed changes do not constitute a significant hazards consideration. We have determined that this change will not authorize a significant change in the types or a significant increase in the amounts of effluents or in the authorized power level and will not result in any significant environmental impact. Therefore, pursuant to 10 CFR 50.5(d)(4), no environmental impact statement, negative declaration or environmental impact appraisal is required.

The OSC and ORC have reviewed the proposed changes, and based on the above safety evaluation, it is concluded there is reasonable assurance that the public health and safety will not be endangered by operations in the proposed manner.