TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT#	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (RM-207)	1	*	≤15 mR/hr	10^{-1} - 10^{4} mR/hr	19
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	1	6	≤1.6×10^{3} cpm	10 - 10^{6} cpm	22
ii. Area (RM-RM-219 A & B)	2	1, 2, 3, & 4	≤30 R/hr	1 - 10^{7} R/hr	36
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection (RM 215B)	1	1, 2, 3, & 4	N/A	10 - 10^{6} cpm	20
ii. Particulate Activity RCS Leakage Detection (RM 215A)	1	1, 2, 3, & 4	N/A	10 - 10^{6} cpm	20
b. Fuel Storage Building Gross Activity (RMVS - 103 A & B)	1	**	≤4.0×10^{4} cpm	10 - 10^{6} cpm	21

* With fuel in the storage pool or building
* With Irradiated fuel in the storage pool
\# Above background

8405310202 840521
PDR ADOCK 05000334
P PDR

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (RM 207)	S	R	M	*
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	6
ii. Area (RM-RM-219 A & B)	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R	M	1, 2, 3 & 4
b. Fuel Storage Building Gross Activity (RMVS 103 A & B)	S	R	M	**

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be resumed provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with specification 4.11.1.1.1, and;

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that at least once per 8 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a Lower Limit of Detection (LLD) of at least 10^{-7} μCi/ml.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.5 A reactor coolant loop shall remain isolated until:

a. The isolated loop has been operating on a recirculation flow of $\geq$125 gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The reactor is subcritical by at least 1 percent $\Delta k/k$.

APPLICABILITY: ALL MODES.*

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.5.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.5.2 The reactor shall be determined to be subcritical by at least 1 percent $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

* With fuel in the vessel.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment sump discharge flow measurement system or narrow range level instrument, and

c. Containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the above required radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

1. The other two above required leakage detection systems are OPERABLE, and

2. Appropriate grab samples are obtained and analyzed at least once per 24 hours:

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

b. With the containment sump discharge flow measurement system and narrow range level instrument inoperable, restore at least one inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable in Modes 1, 2 and 3.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

SURVEILLANCE REQUIREMENTS (Continued)

b. Containment sump discharge flow measurement system-performance of CHANNEL CALIBRATION TEST at least once per 18 months.

c. Logging the narrow range level indication every 12 hours.

TABLE 4.4-3

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Capsule	Vessel Location	Lead Factor	Withdrawal Time (EFPY)
V	165°	1.37	1 EFPY (Removed)
U	65°	.89	3 EFPY
W	245°	.89	6 EFPY
Y	295°	.89	15 EFPY
X	285°	1.37	EOL
T	55°	.58	Standby
Z	305°	.58	Standby
S	45°	.43	Standby

PROPOSED WORDING

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H,reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

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3/4.7.5 ULTIMATE HEAT SINK - OHIO RIVER

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 654 Mean Sea Level, at the intake structure, and

b. An average water temperature of $\leq$86°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.1.2 The following spray and/or sprinkler systems shall be OPERABLE:

a. Containment (RHR Area)*

b. Containment (Cable Penetration Area)*

c. Auxiliary Feedwater Pump Area**

d. CCR Pump Area

e. Main Filter Bank

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a roving fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged such that the area is checked hourly when the system has to be operable. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 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.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated, or automatic in the flow path accessible during plant operation is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

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* With a containment area sprinkler system inoperable check this area during scheduled containment entries in modes 1-4 and once per shift in modes 5 and 6.

** Until such time as the backup auxiliary Feedwater Pump is operable, establish a continuous firewatch whenever an auxiliary feedwater pump area sprinkler system is inoperable.

SURVEILLANCE REQUIREMENTS (continued)

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a manual test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

Note: The only open head spray/sprinkler nozzles are those associated with the Main Filter Banks, and the cable penetration area in containment.

PROPOSED WORDING

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.14.4 The fire hose stations in the following locations shall be OPERABLE.

a. Primary Auxiliary Building
b. Fuel Building
c. Intake Structure
d. Service Building (Safety Related Areas)
e. Safeguards Building (Pipe Tunnel Areas)
f. Containment

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the above fire hose stations inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour (4 hours for containment hose stations) if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.4 Each of the above fire hose stations shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2. Removing the hose for inspection and re-racking, and

   3. Replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig above maximum fire main operating pressure.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.5 The following Halon systems shall be OPERABLE.

a. Process Equipment Area Zone 1

b. Process Equipment Area Zone 2

c. Cable Tunnel (CV-3)

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within 1 hour establish a roving fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; such that the area is checked hourly. Restore the system to OPERABLE status within 14 days Or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 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.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.5 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight (or level) and pressure to be at least 90% of full charge pressure.

c. At least once per 18 months by:

   1. Verifying the system actuates manually and automatically, upon receipt of a simulated actuation signal.

   2. Visually inspect each header and nozzle to verify their integrity.

PROPOSED WORDING

3/4.7.15 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.15 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable and piping penetration seals and ventilation seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol until the functional capability of the barrier is restored.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.15.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assemblies.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.15.2 Each of the above required fire doors * shall be verified OPERABLE by inspecting the automatic holdopen, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The position of each closed Fire door at least once per 24 hours.

b. That doors with automatic holdopen and release mechanisms are free of obstructions at least once per 24 hours.

* Security alarm fire doors are not included in the above surveillance requirements, since they are checked per security requirements.

3/4.7.13 AUXILIARY RIVER WATER SYSTEM

The operability of the ARWS ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal RIVER WATER SYSTEM supply pumps inoperable.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinquish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

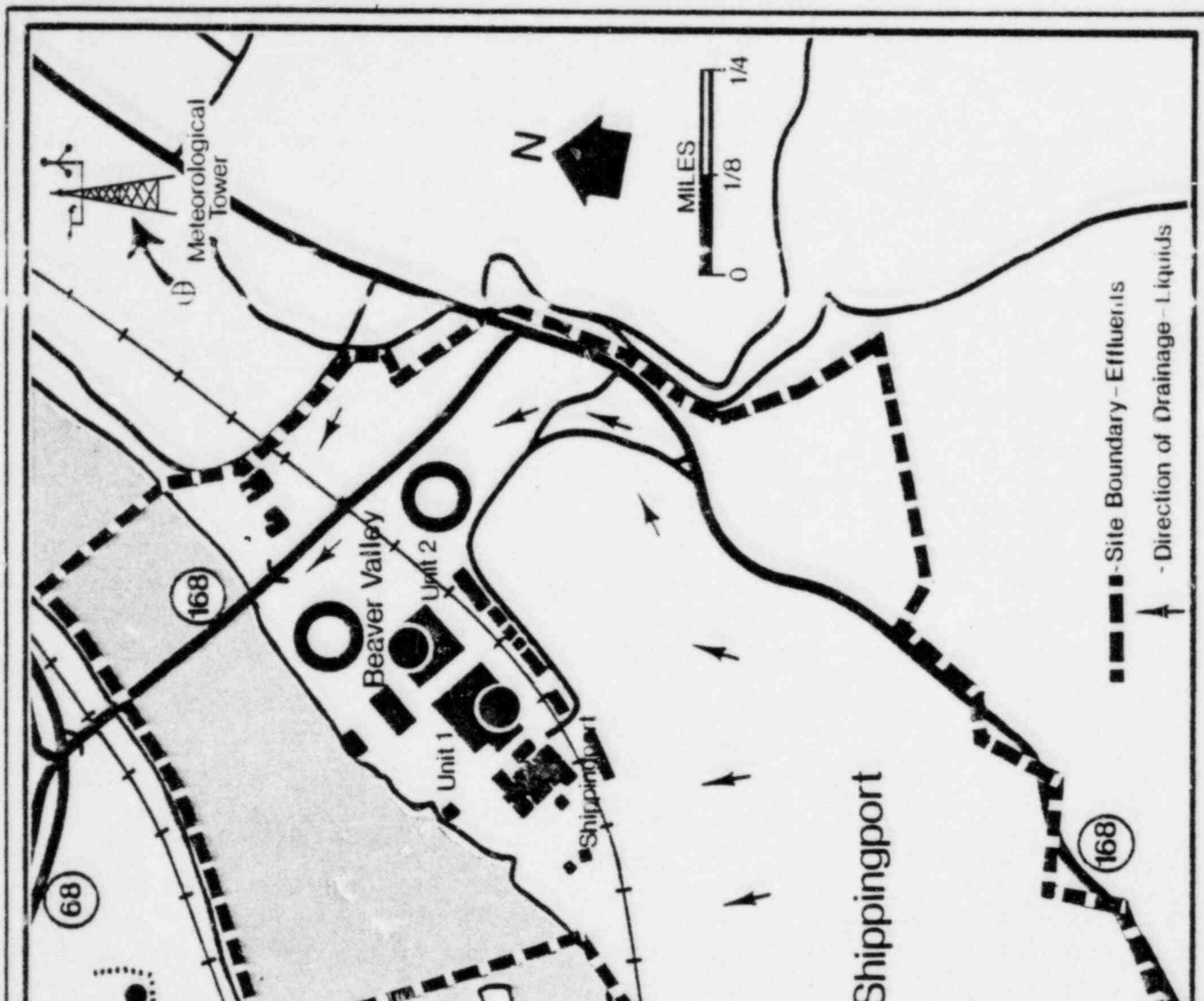
The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. The halon systems are indoor, underfloor cable area systems not susceptible to outdoor weather conditions. The systems are dry pipe (rust is not expected) gas suppression systems.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

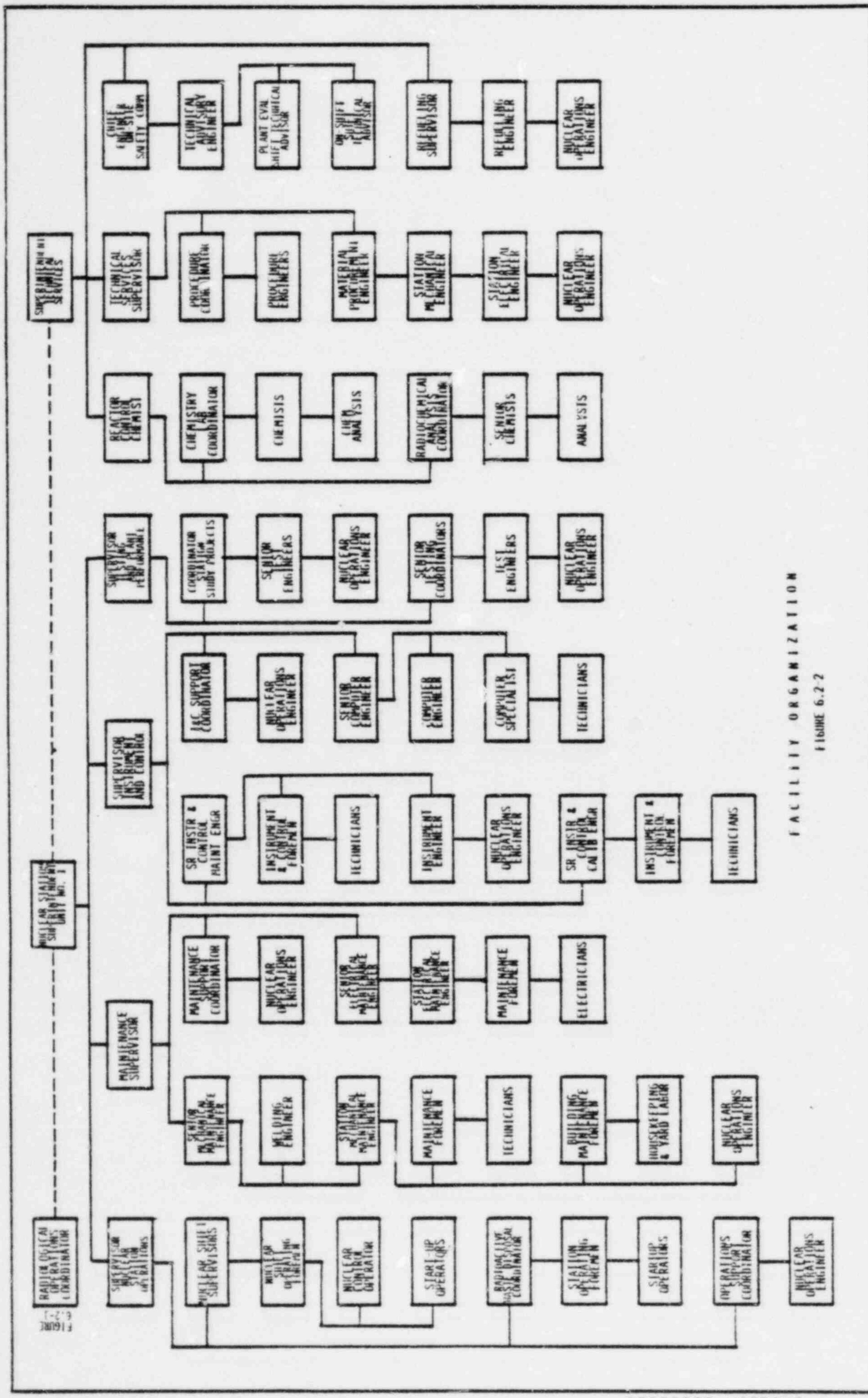
3/4.7.15 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their operability.

SITE BOUNDARY FOR GASEOUS EFFLUENTS
FOR THE BEAVER VALLEY POWER STATION



PROPOSED WORDING



FACILITY ORGANIZATION

FIGURE 6.2-2

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility and Radiation Protection staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiological Operations Coordinator who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Director Nuclear Division Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A Training program for the Emergency Squad shall be maintained under the direction of the Director Nuclear Division Training and shall meet or exceed the requirements of Section 27 of the NEPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the Station Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The OSC shall be composed of the:

Chairman:	Chief Engineer
Member:	Senior Licensed Operator
Member:	Radiation Control Foreman
Member:	Maintenance Engineer
Member:	Project Engineer - Nuclear Engineering Dept.
Member:	Senior Testing or Study Projects Coordinator
Member:	Shift Technical Advisor
Member:	Chemist
Member:	Quality Control Engineer
Member:	I & C supervisor

NOTE: The chairman of the OSC shall appoint an individual from each of the above listed job categories to serve as a member of the OSC for a period of at least 6 months.

NOTE: OSC members shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The SRO shall meet the qualifications of Section 4.2.2 and the Maintenance Engineer will meet the qualifications of Section 4.2.3.

ATTACHMENT B

SAFETY EVALUATION

Proposed Change Request No. 95 amends the Beaver Valley Power Station, Unit No. 1 Technical Specifications, Appendix A concerning various administrative changes.

Description and Purpose of Change

1. Table 3.3-6 Radiation Monitoring Instrumentation, and Table 4.3-3 Radiation Monitoring Instrumentation Surveillance Requirements: delete "Purge and Exhaust Isolation (RM-215A and RM-215B)" from item 2.a.i and 2.a.ii on both tables.

2. Table 3.3-12 Action statement 23: correct a typographical error, the referenced Surveillance Requirement 4.11.1.1.3 should be 4.11.1.1.1.

3. Section 3.4.1.5 Isolated Loop Startup: add a note to clarify the APPLICABILITY statement * With fuel in the vessel. .

4. Section 3.4.6.1 Leakage Detection Systems: add the narrow range sump level instrument to the Action statement; Action statement c: "The provisions of specification 3.0.4 are not applicable", was added; Surveillance Requirement 4.4.6.1.c: added a requirement to log the narrow range level indication once per 12 hours.

5. Table 4.4-3, Reactor Vessel Material Irradiation Surveillance Schedule, has been revised to reflect the revised capsule removal schedule recommended by WCAP-9860. The Bases have also been revised to reference 10CFR50 Appendix H for capsule removal and evaluation.

6. Section 4.4.10, Surveillance Requirements for ASME Code Class 1, 2 and 3 Components, has been revised to delete the augmented inservice inspection program requirements. The augmented inservice inspection program was completed following the third refueling outage, therefore, the surveillance requirements of 4.4.10 have been satisfied and no longer apply.

7. Section 3.7.5.1 Ultimate Heat Sink-Ohio River: correct a typographical error, Surveillance Requirement 4.7.6.1 should be 4.7.5.1.

8. Section 3.7.14.2 - Spray and/or Sprinkler Systems, has been updated for application to the following additional areas:

   Containment (RHR Area)
   Containment (Cable Penetration Area)
   Auxiliary Feedwater Pump Area
   CCR Pump Area

Section 3.7.14.4 - Fire Hose Stations, has been updated for application to the following additional areas:

Service Building (Safety Related Areas)
Safeguards Building (Pipe Tunnel Areas)
Containment

Section 3.7.14.5 - Halon Systems, has been added for application to the following areas:

Process Equipment Area Zone 1
Process Equipment Area Zone 2
Cable Tunnel (CV-3)

Section B 3/4.7.14 - Fire Suppression Systems, the applicable bases have been revised to reflect the above changes.

Section 3.7.15 - Penetration Fire Barriers, has been retitled to "Fire Rated Assemblies" applicable to all fire rated assemblies separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and sealing devices in fire rated assembly penetrations.

Section B 3/4.7.15 - Penetration Fire Barriers, has been retitled to "Fire Rated Assemblies" applicable to the above specification.

9. Figure 5.1-1 Site Boundary for Gaseous Effluents for the Beaver Valley Power Station: correct this figure by adding the "Meteorological Tower" as referenced in Section 5.8.1.

10. Figure 6.2-2, Facility Organization, has also been revised to reflect the reorganization of Station Engineering. The functions deleted from Station Engineering responsibility have been incorporated into the Nuclear Engineering Department.

11. Section 6.5.1.2 revises the title of one of the OSC members from "Senior Engineer - Station Engineering" to "Project Engineer - Nuclear Engineering Department" and adds an additional member "I&C Supervisor".

<u>Basis</u>

1. Is the probability of an occurrence of the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Updated Final Safety Analysis Report (UFSAR) increased? <u>No</u>.

<u>Reason</u>:

1. "Purge and Exhaust Isolation (RM-215A and RM-215B)" has been

deleted to correct Table 3.3-6 and 4.3-3 because these radiation monitors do not actuate isolation of the Purge and Exhaust system as indicated on the tables. These monitors provide control room alarm indication and this change is consistent with UFSAR Section 11.3.4. The automatic isolation of the Purge and Exhaust System is actuated by RM-VS-104A and RM-VS-104B as stated in the tables under item 1.b.i and is consistent with UFSAR Section 11.3.1.

2. Table 3.3-12 has been revised to correct a typographical error on the Surveillance Requirement that should have been referenced. This change is not a safety concern and does not affect the UFSAR.

3. Section 3.4.1.5 Isolated Loop Startup has been revised to add a note to clarify the Applicability statement "With fuel in the vessel". This specification is intended to prevent a reactivity transient due to the injection of cool water from the startup of a idle loop. Therefore, with no fuel in the vessel there can be no reactivity transient, no safety concern and no need for this specification requirements. This change is not a safety concern and does not affect the UFSAR.

4. Section 3.4.6.1 Leakage Detection Systems, has been revised to add the narrow range sump level instrument. The narrow range instrument provides an additional method of leakage indication and as discussed in the UFSAR Section 7.3.1.3.1 monitors the sump level under normal operating conditions. The addition of Action statement c; exception to specification 3.0.4 when in Modes 1, 2 and 3; permits the plant to be restarted following a plant trip. Alternate and diverse methods of monitoring RCS leakage are available, i.e., containment sump discharge flow measurement or narrow range sump level, therefore, should one of these systems become inoperable the other system will be available to monitor and detect RCS leakage. Plant startup can therefore continue without compromising the ability to monitor for RCS leakage.

5. The basis for the Reactor Vessel Material Irradiation Surveillance Schedule change has been incorporated into the Updated FSAR Section 4.5.1.2 by reference to ASTM E185-79, based upon recommendations in WCAP-9860. ASTM E185-82, referenced in 10CFR50 Appendix H, is identical to ASTM E185-79 except for a minor editorial change. This change is not a safety concern and is consistent with the basis described in the UFSAR.

6. Section 4.4.10, Surveillance Requirements for ASME Code Class 1, 2 and 3 Components for the augmented inservice inspection program have been satisfied and therefore can be deleted. This change is not a safety concern and does not affect the UFSAR.

7. Section 3.7.5.1 Ultimate Heat Sink-Ohio River, the correction of the typographical error is not a safety concern and does not affect the UFSAR.

8. The probability of an occurrence or the consequence of an accident caused by fire or the fire induced malfunction of safety-related equipment are not evaluated in the UFSAR Section 9.10. The proposed technical specifications changes are administrative in nature and incorporate additional requirements applicable to the equipment added to the Fire Protection System by Design Changes 268 and 553. The Fire Protection System has been upgraded by the design changes and will enhance the effectiveness of the system to comply with the standards of the National Fire Protection Association and the intent of 10CFR50 Appendix R.

9. Figure 5.1-1 Site Boundary For Gaseous Effluents for the Beaver Valley Power Station, has been revised to identify the location of the Meteorological Tower as referenced by Section 5.8.1. This figure had been revised by the RETS (Amendment 66) and the location of the Meteorological Tower had been omitted. This change is not a safety concern and does not affect the UFSAR.

10. Figure 6.2-2, Facility Organization revisions do not affect the safety functions performed by onsite organization personnel, are not a safety concern and do not affect the UFSAR.

11. Section 6.5.1.2, OSC membership changes, are consistent with the minimum qualifications for OSC membership. This change is not a safety concern and does not affect the UFSAR.

2. Is the possibility 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. These changes provide clarification of existing specifications or add additional requirements and will enhance the overall effectiveness of the technical specifications.

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

   Reason: The bases for specification 3/4 4.9, Pressure/Temperature

Limits have been revised to reference 10CFR50 Appendix H for the removal and evaluation of reactor vessel material irradiation specimens. The bases for specification 3/4 7.14, Fire Suppression Systems have been revised to reflect the additional requirements imposed by changes to the applicable specifications. The bases for specification 3/4 7.15 has been changed from Penetration Fire Barriers to Fire Rated Assemblies to reflect the revised specifications in accordance with the standard technical specifications. These changes reflect additional requirements and provide additional assurance that the safety of the plant will be maintained, therefore, the margin of safety inherent in the applicable bases will not be reduced. The other changes do not affect the bases, therefore, the margin of safety as defined in the 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, intended to update or clarify existing specifications or add additional requirements to enhance the effectiveness of the Technical Specifications.

The OSC membership and Facility Organization Chart changes reflect the reorganization of the Nuclear Group. The Reactor Vessel Material Irradiation Surveillance schedule change complies with 10CFR50 Appendix H. Two major changes resulting from the capsule removal schedule revision are (1) all removal times are changed from operating years to Effective Full Power Years, and (2) the samples to be tested will not be moved to the position of highest neutron fluence levels after the previous test sample is removed. Relocation of the sample capsules is no longer recommended due to the increased complexity of the calculation and the uncertainty of the assigned lead factor as a result of the neutron fluence differences at the several locations.

The Fire Protection System changes reflect plant modifications incorporated by Design Change 268, "Fire Protection System Modifications", and 553, "Halon System for the Cable Tunnel (CV-3)". Incorporating these changes reflect additional requirements in accordance with the Standard Technical Specifications, and comply with the intent of 10CFR50 Appendix R and Branch Technical Position 9.5.1.

The changes do not involve physical change to any 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.

Based on the considerations addressed above the proposed revisions have been determined to be safe and do not involve an unreviewed safety question.