UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 14, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-29 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment shall become effective 6 months after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1981.

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

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QUAD CITIES
DPR-29

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APPENDIX A

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II. DOSE EQUIVALENT I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

JJ. PROCESS CONTROL PROGRAM (PCP) - Contains the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

KK. OFFSITE DOSE CALCULATION MANUAL (ODCM) - Contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints.

LL. CHANNEL FUNCTIONAL TEST (RADIATION MONITOR) - Shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

MM. SOURCE CHECK - The qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

NN. MEMBER(S) OF THE PUBLIC - Shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

3.2/4.2 PROTECTIVE INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

SPECIFICATIONS

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2-1.

A. Primary Containment Isolation Functions

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2-2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Specification 3.5.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

2. The minimum number of operable instrument channels specified in Table 3.2-3 for the rod block monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. If this condition exists for more than 24 hours in a 30-day period, the system shall be tripped.

C. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

D. Refueling Floor Radiation Monitors

1. Except as specified in Specification 3.2.D.2, the two refueling floor radiation monitors shall be operable whenever irradiated fuel or components are present in the fuel storage pool and during refueling or fuel movement operations.

2. One of the two refueling floor radiation monitors may be inoperable for 24 hours. If the inoperable monitor is not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The trip setting for the refueling floor radiation monitors shall be set at a value of 100 mR/hr.

4. Upon loss of both refueling floor radiation monitors while in use, the reactor building ventilation system shall be isolated and the standby gas treatment operated.

D. Refueling Floor Radiation Monitors

The two refueling floor radiation monitors shall be functionally tested and calibrated as indicated in Table 4.2-1. Reactor building ventilation isolation and standby gas treatment system initiation shall be performed at least each operating cycle.

E. Postaccident Instrumentation

The limiting conditions for operation for the instrumentation which is read out in the control room, required for postaccident monitoring are given in Table 3.2-4.

E. Postaccident Instrumentation

Postaccident instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-2.

F. Control Room Ventilation System Isolation

The control room ventilation system is isolated from outside air on a signal of high drywell pressure, low water level, high main steamline flow, or high radiation in either of the reactor building ventilation exhaust ducts. Limiting conditions for operation shall be as indicated in Table 3.2-1 and Specification 3.2.H.1.

F. Control Room Ventilation System Isolation

Surveillance for instrumentation which initiates isolation of control room ventilation shall be as specified in Table 4.2-1.

G. Radioactive Liquid Effluent Instrumentation

The effluent monitoring instrumentation shown in Table 3.2-5 shall be operable with alarm setpoints set to ensure that the limits of Specification 3.8.B are not exceeded. The alarm setpoints shall be determined in accordance with the ODCM.

1. With a radioactive liquid effluent monitoring instrument alarm/trip setpoint less conservative than required, without delay suspend the release of radioactive liquid effluents monitored by the affected instrument, or declare the instrument inoperable, or change the setpoint so it is acceptably conservative.

2. With one or more radioactive liquid effluent monitoring instruments inoperable, take the action shown in Table 3.2-5. Exert best efforts to return the instrument to operable status within 30 days and, if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. This is in lieu of an LER.

G. Radioactive Liquid Effluent Instrumentation

Each radioactive liquid effluent monitoring instrument shown in Table 4.2-3 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequencies shown in Table 4.2-3.

3. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

H. Radioactive Gaseous Effluent Instrumentation

The effluent monitoring instrumentation shown in Table 3.2-6 shall be operable with alarm/trip setpoints set to ensure that the limits of Specification 3.8.A. are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

1. With a radioactive gaseous effluent monitoring instrument alarm/trip setpoint less conservative than required, without delay suspend the release of radioactive gaseous effluents monitored by the affected instrument, or declare the instrument inoperable, or change the setpoint so it is acceptably conservative.

2. With one or more radioactive gaseous effluent monitoring instruments inoperable, take the action shown in Table 3.2-6. Exert best efforts to return the instrument to operable status within 30 days and, if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. This is in lieu of an LER.

H. Radioactive Gaseous Effluent Instrumentation

Each radioactive gaseous radiation monitoring instrument in Table 4.2-4 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequency shown in Table 4.2-4.

3. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to the Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance and (2) to precribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations. Some of the settings on the instrumentation that initiates or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by the protective instrumentation which serves the condition for which isolation is required (this instrumentation is shown in Table 3.2.1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the basis for Specification 3.1 is applicable here.

The low reactor level instrumentation is set to trip at > 8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses*. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break: the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary isolation are initiated and in time to meet the above criteria. The instrumentation also covers the full spectrum of breaks and meets the above criteria.

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also Group 2 isolation valves include the drywell vent, purge and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

* Loss of coolant accident analysis for Dresden Units 2 & 3 and Quad Cities Units 1 & 2, NEDO-24146A, April, 1979

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with $\leq$100 CPS and the detector not full inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level block provide annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident, i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

The radioactive liquid and gaseous effluent instrumentation is provided to monitor the release of radioactive materials in liquid and gaseous effluents during releases. The alarm setpoints for the instruments are provided to ensure that the alarms will occur prior to exceeding the limits of 10 CFR 20.

Table 3.2-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels	Total No. of Channels	Parameter	Action (1)
1	1	Service Water Effluent Gross Activity Monitor	A
1	1	Liquid Radwaste Effluent Flow Rate Monitor	C
1	1	Liquid Radwaste Effluent Gross Activity Monitor	B
1	1	Spray Canal Discharge Blowdown Flow Rate Monitor	C

Notes:

Action A: With less than the minimum number of operable channels, releases via this pathway may continue, provided that at least once per 12 hours grab samples are collected and analyzed for beta or gamma activity at an LLD of less than or equal to 10^{-7} uCi/ml.

Action B: With less than the minimum number of operable channels, effluent releases via this pathway may continue, provided that prior to initiating a release, at least 2 independent samples are analyzed in accordance with Specification 4.8.B.1., and at least 2 members of the facility staff independently verify the release calculation and discharge valving. Otherwise, suspend release of radioactive effluents via this pathway.

Action C: With less than the minimum number of operable channels, 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 utilized to estimate flow.

Table 3.2-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels (1)	Total No. of Channels	Parameter	Action (2)
1	2	SJAE Radiation Monitors	D
1	2	Main Chimney Noble Gas Activity Monitor	A
1	1	Main Chimney Iodine Sampler	C
1	1	Main Chimney Particulate Sampler	C
1	1	Reactor Bldg. Vent Sampler Flow Rate Monitor	B
1	1	Reactor Bldg. Vent Iodine Sampler	C
1	1	Reactor Bldg. Vent Particulate Sampler	C
1	1	Main Chimney Sampler Flow Rate Monitor	B
1	1	Main Chimney Flow Rate Monitor	B
1	2	Reactor Bldg. Vent Noble Gas Monitor	E

Notes

(1) For SJAE monitors, applicable during SJAE operation. For other instrumentation, applicable at all times.

(2) Action A: With the number of operable channels less than the minimum requirement, effluent releases via this pathway may continue, provided grab samples are taken at least once per 8 hour shift and these samples are analyzed within 24 hours.

Action B: With the number of operable channels less than the minimum required, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per 4 hours.

Action C: With less than the minimum channels operable, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment, as required in Table 4.8-1.

Action D: With less than the minimum channels operable, gases from the main condenser off-gas system may be released to the environment for up to 72 hours provided the off-gas system is not bypassed and at least one chimney monitor is operable; otherwise, be in hot stand-by in 12 hours.

Action E: With less than the minimum channels operable, immediately suspend release of radioactive effluents via this pathway.

Table 4.2-1 (Cont'd)

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
HPCI Isolation			
1. Steamline high flow	(1)	Once/3 months	None
2. Steamline area high temperature	Refueling Outage	Refueling Outage	None
3. Low reactor pressure	(1)	Once/3 months	None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refuel Floor Rad. Monitors	(1)	Once/3 months	Once/day
Control Room Ventilation System Isolation			
1. Reactor low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Main steamline high flow	(1)	Once/3 months	Once/day

Notes

1. Initially once per month until exposure hours (M as defined on Figure 4.1-1) are 2.0×10^5; thereafter, according to Figure 4.1-1 with an interval not less than 1 month nor more than 3 months. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of Quad-Cities 1/2.
2. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable, or are tripped.
3. This instrumentation is excepted from the functional test definition. The functional test shall consist of injecting a simulated electrical signal into the measured channel.
4. This instrument channel is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every 3 months.
5. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week.
6. The positioning mechanism shall be calibrated every refueling outage.
7. Logic system functional tests are performed as specified in the applicable section for these systems.
8. Functional test shall include verification of operation of the degraded voltage 5-minute timer and 7-second inherent timer.
9. Verification of the time delay setting of $3 \leq \tau \leq 10$ seconds shall be performed during each refueling outage.

Table 4.2-3

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)	Calibration (1) (3)	Functional Test (1) (2)	Source Check (1)
Liquid Radwaste Effluent Gross Activity Monitor	D	R	Q (7)	(6)
Service Water Effluent Gross Activity Monitor	D	R	Q (7)	R
Liquid Radwaste Effluent Flow Rate Monitor	(4)	R	NA	NA
Blowdown Flow Rate Monitor Spray Canal Discharge	(4)	R	NA	NA

Notes:

(1) D = once per 24 hours
M = once per 31 days
Q = once per 92 days
R = once per 18 months
S = once per 6 months

(2) The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

a. Instrument indicates levels above the alarm setpoint.
b. Circuit failure.
c. Instrument indicates a downscale failure.
d. Instrument controls not set in OPERATE mode.

(3) Calibration shall include performance of a functional test.

(4) Instrument Check to verify flow during periods of release.

(5) Calibration shall include performance of a source check.

(6) Source check shall consist of observing instrument response during a discharge.

(7) Functional test may be performed by using trip check and test circuitry associated with the monitor chassis.

Table 4.2-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Mode(2)	Instrument Check(1)	Calibration(1)(4)	Functional Test(1)(3)	Source Check(1)
Main Chimney Noble Gas Activity Monitor	B	D	R	Q	M
Main Chimney Sampler Flow Rate Monitor	B	D	R	Q(6)	NA
Reactor Bldg. Vent Sampler Flow Rate Monitor	B	D	R	Q(6)	NA
Main Chimney Flow Rate Monitor	B	D	R	Q	NA
Reactor Bldg. Vent Activity Monitor	B	D	R	Q	Q
SJAE Activity Monitor	A	D	R	Q	R
Main Chimney Iodine and Particulate Sampler	B	D(5)	NA	NA	NA
Reactor Bldg. Vent Iodine and Particulate Sampler	B	D(5)	NA	NA	NA

Notes

(1) D = once per 24 hours
M = once per 31 days
Q = once per 92 days
R = once per 18 months

(2) A = during SJAE operation
B = at all times

(3) The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable:

a. Instrument indicates levels above the alarm setpoint
b. Circuit failure
c. Instrument indicates a downscale failure
d. Instrument controls not set in OPERATE mode

(4) Calibration shall include performance of a functional test.

(5) Instrument check to verify operability of sampler; that the sampler is in place and functioning properly

(6) Functional test shall be performed on local switches providing low flow alarm.

3.8/4.8 RADIOACTIVE EFFLUENTS

Limiting Conditions for Operation	Surveillance Requirements
Applicability:	Applicability:
Applies to the radioactive effluents from the plant.	Applies to the periodic measurements radioactive effluents.

Specifications

A. Gaseous Effluents	A. Gaseous Effluents
1. The dose rate in unrestricted areas (at or beyond the site boundary, Figure 4.8-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following:	1. The dose rates due to radioactive materials released in gaseous effluents from the site shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 4.8-1. The dose rates are calculated using methods prescribed in the Off-Site Dose Calculation Manual (ODCM).
a. For Noble Gases:	
(1) Less than 500 mrem/ year to the whole body.	
(2) Less than 3000 mrem/ year to the skin.	
b. For iodine-131, for iodine -133, and for all radionuclides in particulate form with half-lives greater than 8 days less than 1500 mrem/year.	
c. If the dose rates exceed the above limits, without delay decrease the release rates to bring the dose rates within the limits, and provide prompt notification to the Commission (6.6.B.1.)	
2. The air dose in unrestricted areas (at or beyond the site boundary) due to Noble Gases released in gaseous effluents from the unit shall be limited to the following:	2. The air dose due to releases of radioactive noble gases in gaseous effluents shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in sections A and B of Table 4.8-1. The allocation of effluents between units having shared effluent control systems and the air doses are determined using methods prescribed in the ODCM at least once every 31 days.
a. For gamma radiation:	
(1) Less than or equal to 5 mrad during any calendar quarter.	

(2) Less then or equal to 10 mrad during any calendar year.

b. For Beta radiation:

(1) Less than or equal to 10 mrad during any calendar quarter

(2) Less than or equal to 20 mrad during any calendar year.

c. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases are in compliance with 3.8.A.2.a. & b. This is in lieu of a Licensee Event Report.

d. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding the limits of Specification 3.8.A.2.a. or 3.8.A.2.b., prepare and submit a Special Report to the Commission within 30 days and limit the subsecuent releases such that the doses or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise, obtain a

variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use the methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

3. The dose to a member of the public in unrestricted areas (at or beyond the site boundary) from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the unit shall be limited to the following:

   a. Less than or equal to 7.5 mrem to any organ during any calendar quarter.

   b. Less than or equal to 15 mrem to any organ during any calendar year.

   c. With the calculated dose from the release of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken and the proposed actions to be taken to ensure that future releases are in compliance with 3.8.A.3. a. & b. This is in lieu of a Licensee Event Report.

3. The dose to a member of the public due to releases of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 4.8-1.

   For radionuclides not determined in each batch or weekly composite, the dose contribution to the current calendar quarter cumulative summation may be estimated by assuming an average monthly concentration based on the previous monthly or quarterly composite analyses. However, for reporting purposes, the calculated dose contributions shall be based on the actual composite analyses when possible.

   The allocation of effluents between units having shared effluent control systems and the doses are determined using the methods prescribed in the ODCM at least once every 31 days.

d. With the calculated dose from the release of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents exceeding the limits of Specification 3.8.A.3.a. or 3.8.A.3.b., prepare and submit a Special Report to the Commission within 30 days and limit subsequent releases such that the dose or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or equal to 25 mrem to the total body or organ (except the thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use the methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

4.	Off-Gas System a. At all times during processing for discharge to the environs, process and control equipment provided to reduce the amount or concentration of radioactive materials shall be operated. b. The above specification shall not apply for the Off-Gas Charcoal Absorber Beds below 30 percent of rated thermal power.	4. Off-Gas System Doses due to treated gases released to unrestricted areas at or beyond the site boundary shall be projected at least once per 31 days in accordance with the ODCM.
5.	Explosive Gas Mixture a. The concentration of hydrogen in the off-gas hold up system, downstream of the recombiner shall be limited by having a recombiner operable within the allowable band of the base-line plot of recombiner outlet temperature vs. reactor power, whenever the reactor is operating at a pressure greater than 900 psig. b. The recombiner may be inoperable for 48 hours.	5. Explosive Gas Mixture Once per 8 hours verification will be made that the unit is operating within the allowable band of the base-line plot of recombiner outlet temperature vs. reactor power.
6.	With either the recombiners inoperable, or all charcoal beds bypassed for more than 7 days in a calendar quarter while operating above 30 percent of rated thermal power, prepare and submit to the Commission within 30 days a special report which includes the following information:	

a. Identification of the defective equipment.

b. Cause of the defective equipment

c. Action(s) taken to restore the equipment to an operating status.

d. Length of time the above requirements were not satisfied.

e. Volume and curie content of the waste discharged which was not processed by the inoperable equipment but which required processing.

f. Action(s) taken to prevent a recurrance of equipment failures.

This is in lieu of a Licensee Event Report.

7. The release rate of the sum of the activities from the noble gases measured at the main condenser air ejector shall be limited to less than or equal to 100 microcuries/sec per MWt (after 30 minutes decay) at all times. With the release rate of the sum of the activities from noble gases at the main condenser air ejector exceeding 100 microcuries/sec per MWt (after 30 minutes decay), restore the release rate to within its limits within 72 hours, or be in at least HOT STANDBY within the next 12 hours.

7. The radioactivity rate of nobles ga at (near) the outlet of the main co denser air ejector shall be contin ously monitored in accordance with Specification 3.2.H. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be dete mined to be within the limits of Specification 3.8.H. at the follow frequencies by performing an isotop analysis of a representative sample of gases taken at the recombiner ou let, or at the air ejector outlet i the recombiner is bypassed.

a. At least once per 31 days:

b. Within 4 hours following an in- crease, as indicated by the mai condenser air ejector noble gas activity monitor, of greater th 50%, after factoring out increa due to changes in thermal power level and off-gas flow, in the nominal steady-state fission ga release from the primary coolan

B. Liquid Effluents	B. Liquid Effluents
1. The concentration of radioactive material released from the site to unrestricted areas (at or beyond the site boundary, figure 4.8-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 with the Table 4.8-2 values representing the MPC's for noble gases. With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits, without delay decrease the release rate of radioactive materials and/or increase the dilution flow rate to restore the concentration to within the above limits.	1. The concentration of radioactive material in unrestricted areas shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 4.8-3. The sample analysis results will be used with the calculational methods in the ODCM to determine that the concentrations are within the limits of Specification 3.8.B.1.

2. The dose or dose commitment above background to a member of the public from radioactive materials in liquid effluents released to unrestricted areas (at or beyond the site boundary) from the site shall be limited to the following:

   a. During any calendar quarter:

      (1) Less than or equal to 3 mrem to the whole body.

      (2) Less than or equal to 10 mrem to any organ.

   b. During any calendar year:

      (1) Less than or equal to 6 mrem to the whole body.

      (2) Less than or equal to 20 mrem to any organ.

   c. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken and the proposed actions to be taken to ensure that future releases are in compliance with 3.8.B.2.a. & b. This is in lieu of a Licensee Event Report.

   d. With the calculated dose from the release of radioactive materials in liquid effluents exceeding the limits of Specification 3.8.B.2.a. or 3.8.B.2.b., prepare and submit a Special Report to the Commission within 30 days and limit the subsequent releases such that the dose or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or

2. a. The dose contributions from measured quantities of radioactive material shall be determined by calculation at least once per 31 days and a cumulative summation of these total body and any organ doses shall be maintained for each calendar quarter.

   b. Doses computed at the nearest community water system will consider only the drinking water pathway and shall be projected using the methods prescribed in the ODCM at least once per 92 days.

equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise obtain a variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

e. With the projected annual whole body or any internal organ dose computed at the nearest downstream community water system is equal to or exceeds 2 mrem from all radioactive materials released in liquid effluents from the Station, prepare and submit a Special Report within 30 days to the operator of the community water system. The report is prepared to assist the operator in meeting the requirements of 40 CFR 141: EPA Primary Drinking Water Standards. A copy of this report will be sent to the NRC. This is in lieu of a Licensee Event Report.

3. At all times during processing prior to discharge to the environs, process and control equipment provided to reduce the amount or concentration of radioactive materials shall be operated when the projected dose due to liquid effluent releases to unrestricted areas (see Figure 4.8-1), when averaged over 31 days, exceeds 0.13 mrem to the total body or 0.42 mrem to any organ.

4. If liquid waste has to be or is being discharged without treatment as required above, prepare and submit to the Commission within 30 days, a report which includes the following information:

   a. Identification of the defective equipment.

   b. Cause of the defective equipment.

   c. Action(s) taken to restore the equipment to an operating status.

   d. Length of time the above requirements were not satisfied.

   e. Volume and curie content of the waste discharged which was not processed by the appropriate equipment but which required processing.

   f. Action(s) taken to prevent a recurrence of equipment failures.

   This is in lieu of a Licensee Event Report.

3. Liquid Waste Treatment

   a. Doses due to liquid releases to unrestricted areas (at or beyond the site boundary) shall be projected at least once per 31 days in accordance with the ODCM.

C. Mechanical Vacuum Pump	C. Mechanical Vacuum Pump
1. The mechanical vacuum shall be capable of being isolated and secured on a signal of main steam high radiation or shall be isolated and secured whenever the main steam isolation valves are open.	At least once during each operating cycle, automatic securing and isolation of the mechanical vacuum pump shall be verified.

D. Environmental Monitoring Program

1. The environmental monitoring program given in Table 4.8-4 shall be conducted except as specified below.

2. With the radiological environmental monitoring program not being conducted as specified in Table 4.8-4, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, contractor omission which is corrected as soon as discovered, malfunction of sampling equipment, or if a person who participates in the program goes out of business. If the equipment malfunctions, corrective actions shall be completed as soon as practical. If a person supplying samples goes out of business, a replacement will be found as soon as possible. All deviations from the sampling schedule shall be described in the annual report.

3. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in the ODCM exceeding the limits of Table 4.8-5 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter, a Special Report which includes an evaluation of any release conditions, environmental

D. Environmental Monitoring Program

1. The radiological environmental monitoring samples shall be collected pursuant to Table 4.8-4 from the locations specified in the ODCM, and shall be analyzed pursuant to the requirements of Table 4.8-6.

2. The results of analyses performed on radiological environmental monitoring samples shall be summarized in the Annual Radiological Environmental Operating Report.

3. The land use census shall be conducted at least once per twelve months between the dates of June 1 and October 1 by a door-to-door survey, aerial survey, road survey, or by consulting local agriculture authorities.

factors or other aspects which caused the limits of Table 4.8-5 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

4. With milk samples unavailable from one or more of the sample locations required by Table 4.8-4, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the Annual Radiological Environmental Operating report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

4. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

5. A census of nearest residences and of animals producing milk for human consumption shall be conducted annually (during the grazing season for animals) to determine their location and number with respect to the site. The nearest residence in each of the 16 meteorological sectors shall also be determined within a distance of five miles. The census shall be conducted under the following conditions:

   a. Within a 2-mile radius from the plant site, enumeration of animals and nearest residences by a door-to-door or equivalent counting technique.

5. The results of the analyses performed as part of the required crosscheck program shall be included in the Annual Radiological Environmental Operating Report. The analyses shall be done in accordance with the ODCM.

b. Within a 5-mile radius, enumeration of animals by using referenced information from county agricultural agents or other reliable sources.

6. With a land use census identifying location(s) of animals which yield(s) an ODCM calculated dose or dose commitment greater than the values currently being calculated in Specification 4.8.A.3, the new location(s) shall be added to the radiological environmental monitoring program with 30 days, if possible.

   The sampling location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

7. Radiological analyses shall be performed on samples representative of those in Table 4.8-4, supplied as a part of the Inter-laboratory Comparison Program which has been approved by the NRC.

8. With analyses not being performed as required, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

E. Solid Radioactive Waste

1. The solid radwaste system shall be used as applicable in accordance with the PCP to process wet radioactive wastes to meet shipping and burial ground requirements.

2. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive waste from the site.

E. Solid Radioactive Waste

1. The PCP shall specify the method and frequency to verify solidification of radioactive waste. Actions to be taken if solidification is not verified shall also be specified in the PCP.

F. Miscellaneous Radioactive Materials Sources

Source Leakage Test

Specification

Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and either decontaminated and repaired or disposed of in accordance with Commission Regulations.

A complete inventory of radioactive materials in the licensee's possession shall be maintained current at all times.

F. Miscellaneous Radioactive Materials Sources

Each sealed source shall be tested for leakage and/or contamination by the licensee or by other persons specifically authorized by the Commission or an Agreement state. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

Each category of sealed sources shall be tested at the frequency described below:

1. Sources in use (excluding startup previously subjected to core flux) - At least once per 6 months for all sealed sources containing radioactive material:

   a. With a half-life greater than 30 days (excluding Hydrogen 3), and

   b. In any form other than gas.

2. Stored sources not in use - Each sealed source shall be tested prior to the use or transfer to another licensee unless tested within the previous 6 months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.C.3 if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

G. In the event a limiting condition for operation and/or associated action requirements identified in sections 3.8.A. through 3.8.E., and 4.8.A. through 4.8.E. cannot be satisfied because of circumstances in excess of those addressed in the specifications, no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

3.8/4.8.A.1 GASEOUS EFFLUENTS - DOSE

This specification is provided to ensure that the dose at the unrestricted area boundary from gaseous effluents from the units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the unrestricted area boundary to less than or equal to 500 mrem/year to the total body or to not less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to not less than or equal to 1500 mrem/year for the nearest cow to the plant. For purposes of calculating doses resulting from airborne releases the main chimney is considered to be an elevated release point, and the reactor vent stack is considered to be a mixed mode release point.

3.8/4.8.A.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. The ODCM equations provide for determining the air doses at the unrestricted boundary based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

3.8/4.8.A.3 DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implements the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods approved by NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these specifications were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man and 3) deposition onto grassy areas where milk animals graze with consumption of the milk by man.

3.8/4.8.A.4 GASEOUS WASTE TREATMENT

The OPERABILITY of the gaseous waste treatment which reduces amounts or concentrations of radioactive materials ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be operable when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50.

3.8/4.8.A.5. EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the off gas system is maintained in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

LIQUID EFFLUENTS

3.8/4.8.B.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, column 2. The concentration limit for noble gases, MPC in air (submersion), was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

3.8/4.8.B.2. DOSE

This specification is provided to implement the requirements of Sections IIA, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A. of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April 1977. NUREG-0113 provides methods for dose calculations consistent with Reg Guide 1.109 and 1.113.

3.8/4.8.B.3 LIQUID WASTE TREATMENT

The operability of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50.

3.8/4.8.D.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. Program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.8-6 are state-of-the-art for routine environmental measurements in industrial laboratories. The specified lower limits of detection for I-131 in water, milk and other food products correspond to approximately one-quarter of the Appendix I to 10 CFR Part 50 design objective dose-equivalent of 15 mrem/year for atmospheric releases and 10 mrem/year for liquid releases to the most sensitive organ and individual. They are based on the assumptions given in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", October 1977, except the change for an infant consuming 330 liter/year of drinking water instead of 510 liters/year.

3.8/4.8.D.6 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

3.8/4.8.D.7 CROSSCHECK PROGRAM

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

3.8/4.8.C MECHANICAL VACUUM PUMP

The purpose of isolating the mechanical vacuum line is to limit release of activity from the main condenser. During an accident, fission products would be transported from the reactor through the main steamline to the main condenser. The fission product radioactivity would be sensed by the main steamline radioactivity monitors which initiate isolation.

3.8/4.8.F. MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

The objective of this specification is to assure that leakage from byproduct, source and special nuclear material sources does not exceed allowable limits. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium.

3.8/4.8.E. SOLID RADIOACTIVE WASTE

The operability of the solid radioactive waste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped off-site. This specification implements the requirements of 10 CFR 50.36a. and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

TABLE 4.8-1
RADIOACTIVE GASEOUS WASTE SAMPLING AND
ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (uci/ml)
A. Main Chimney Reactor Bldg. Vent Stack	M Grab Sample	M[b]	Principal Gamma Emitters[e]	1×10^{-4}
		M	Tritium	1×10^{-6}
B. All Release Types as Listed in A Above	Continuous[d]	W[c] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[d]	W[c] Particulate Sample	Principal Gamma Emitters[e] (I-131, others)	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	SR-89	1×10^{-11}
			SR-90	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
C. Main Chimney	Continuous[d]	Noble Gas Monitor	Noble Gases	1×10^{-6}
D. Reactor Bldg Vent Stack	Continuous[d]	Noble Gas Monitor	Noble Gases	1×10^{-4}

TABLE 4.8-1 (Continued)
TABLE NOTATION

a. The lower limit of detection (LLD) is defined in table notation A. of Table 4.8-6.

b. Sampling and analyses shall also be performed following shutdown, startup, or a thermal power change exceeding 20 percent of rated thermal power in 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least once per 7 days and the analyses completed within 48 hours after removal from the sampler. Sampling shall also be performed within 24 hours following each shutdown, startup, or thermal power level change exceeding 20% of rated thermal power in one hour. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

d. The ratio of sample flow rate to the sampled stream flow rate shall be known.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions, and Mn-54, Fe-59, Co-60, Zn-65, Co-58, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. Other peaks which are measurable and identifiable by gamma ray spectrometry, together with the above nuclides, shall be also identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

TABLE 4.8-2

MAXIMUM PERMISSIBLE CONCENTRATION OF
DISSOLVED OR ENTRAINED NOBLE GASES
RELEASED FROM THE SITE TO UNRESTRICTED AREAS
IN LIQUID WASTE

NUCLIDE	MPC (uCi/ml)*
Kr-58m	2×10^{-4}
Kr-85	5×10^{-4}
Kr-87	4×10^{-5}
Kr-88	9×10^{-5}
Ar-41	7×10^{-5}
Xe-131m	7×10^{-4}
Xe-133m	5×10^{-4}
Xe-133	6×10^{-4}
Xe-135m	2×10^{-4}
Xe-135	2×10^{-4}

* Computed from Equation 20 of ICRP Publication 2 (1959), adjusted for infinite cloud submersion in water, and R = 0.01 rem/week, density = 1.0 g/cc abd Pw/Pt = 1.0.

TABLE 4.8-3

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uci/ml)
A. Batch Waste Release Tanks	Prior to Each Batch	Prior to Each Batch	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	Prior to Each Batch	M Composite[b]	Gross Alpha	1×10^{-7}
			H-3	1×10^{-5}
	Prior to Each Batch	Q Composite[b]	Fe-55	1×10^{-6}
			Sr-89, Sr-90	5×10^{-8}
	Prior to One Batch/M	M	Dissolved & Entrained Gases[f] (Gamma Emitters)	1×10^{-5}
B. Plant Continuous Releases[d]	M[c] (Grab Sample)	M[c]	I-131	1×10^{-6}
			Principal Gamma Emitters[e]	5×10^{-7}
			Dissolved & Entrained Gases[f] (Gamma emmiters)	1×10^{-5}
			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Q[c] (Grab Sample)	Q[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.8-3 (Continued)
TABLE NOTATION

a. The LLD is defined in Notation **A** of Table 4.8-6.

b. A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. If the alarm setpoint of the service water effluent monitor as determined in the ODCM is exceeded, the frequency of analysis shall be increased to daily until the condition no longer exists.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the release.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-60, Zn-65, Co-58, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. Other peaks which are measurable and identifiable by gamma ray spectrometry together with the above nuclides, shall be also identified and reported when the actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

f. The dissolved and entrained gases (gamma emitters) for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138. Other dissolved and entrained gases (gamma emitters) which are measurable and identifiable by gamma-ray spectrometry, together with the above nuclides, shall also be identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

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TABLE 4.8-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Particulates	16 locations	Continuous operation of sampler for a week	Gross beta and gamma isotopic as specified in ODCM.
b. Radioiodine	16 locations	Continuous operation of sampler for two weeks	I-131 as specified in ODCM.
2. DIRECT RADIATION	Forty Locations (Minimum of two TLDs per packet)	Quarterly	

*Sample locations are described in the ODCM.

TABLE 4.8-4 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Public Water	2 Locations	Monthly composite of weekly collected samples	Gamma isotopic analysis of each composite sample
b. Sediment	1 downstream location in receiving body of water	Annually	Gamma isotopic analysis of each sample
c. Plant Cooling Water	Intake, Discharge	Weekly composite	Gross Beta analysis of each sample

*Sample locations are described in the ODCM

TABLE 4.8-4 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INJESTION			
a. Milk	2 Locations	At least once weekly when animals are on pasture; at least once per month at other times.	I-131 analysis of each sample
b. Fish	1 location in receiving body of water	Semi-annually	Gamma isotopic analysis on edible portions

*Sample locations are described in the ODCM

TABLE 4.8-5

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

alysis	Water	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg,wet)	Milk (pCi/l)	Food Products (pCi/Kg,wet)
	Reporting Levels				
3	2×10^4(a)				
-54	1×10^3		3×10^4		
-59	4×10^2		1×10^4		
-58	1×10^3		3×10^4		
-60	3×10^2		1×10^4		
-65	3×10^2		2×10^4		
-Nb-95	4×10^2				
131	2	0.9		3	1×10^2
-134	30	10	1×10^3	60	1×10^3
-137	50	20	1×10^3	70	2×10^3
-La-140	2×10^2			3×10^2	

) for drinking water samples. This is 40 CFR Part 141 value.

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TABLE 4.8-6

PRACTICAL LOWER LIMITS OF DETECTION (LLD)
FOR STANDARD ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

Sample Media	Analysis	LLD[A,B] (4.66 ☛)	Units
Airborne "Particulate"	Gross Beta +	0.01	pCi/m^3
	Gamma Isotopic	0.01	pCi/m^3
Airborne I-131	Iodine-131	0.10	pCi/m^3
Milk/Public Water	I-131	5°	pCi/l
	Cs-134	10	pCi/l
	Cs-137	10 △	pCi/l
	Tritium	200	pCi/l
	Gross Beta +	5	pCi/l
	Gamma Isotopic	20	pCi/l/nuclide
Sediment	Gross Beta +	2	pCi/g dry
	Gamma Isotopic	0.2	pCi/g dry
Fish Tissue	I-131 - Thyroid	0.1	pCi/g wet
	Cs-134, 137	0.1	pCi/g wet
	Gross Beta +	1.0	pCi/g wet
	γ Isotopic	0.2	pCi/g wet

° 0.5 pCi/l on milk samples collected during the pasture season.

\+ Referenced to Cs-137

△ 5.0 pCi/l on milk samples

TABLE 4.8-6 (Continued)
TABLE NOTATION

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

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

$$LLD = \frac{4.55 \cdot s_b}{A \cdot E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t) \cdot t}$$

Where:

LLD is the "a priori" lower limit of detection for a blank sample or background analysis as defined above (as pCi per unit mass or volume).

s_b is the square root of the background count or of a blank sample count; is the estimated standard error of a background count or a blank sample count as appropriate (in units of counts).

E is the counting efficiency (as counts per disintegration).

A is the number of gamma-rays emitted per disintegration for gamma-ray radionuclide analysis (A = 1.0 for gross alpha and tritium measurements).

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

2.22 is the number of disintegrations per minute per picocurie.

Y is the fractional radio-chemical yield when applicable (otherwise Y = 1.0).

λ is the radioactive decay constant for the particular radionuclide (in units of reciprocal minutes).

Δt is the elapsed time between the midpoint of sample collection and the start time of counting. (Δt = 0.0 for environmental samples and for gross alpha measurements).

t is the duration of the count (in units of minutes).

The value of "Sb" used in the calculation of the LLD for a detection system shall be based on an actual observed background count or a blank sample count (as appropriate) rather than on an unverified theoretically predicted value. Typical values of "E", "V", "Y", "t", and "Δt" shall be used in the calculation.

For gamma-ray radionuclide analyses the background counts are determined from the total counts in the channels which are within plus or minus one FWHM (Full Width at Half Maximum) of the gamma-ray photopeak energy normally used for the quantitative analysis for that radionuclide. Typical values of the FWHM shall be used in the calculation.

TABLE 4.8-6 (Continued)
TABLE NOTATION

The LLD for all measurements is defined as an "A priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular sample measurement.

B. Other radionuclides which are measureable and identifiable by gamma-ray spectrometry, together with the nuclides indicated in Table 4.8-6, shall also be identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

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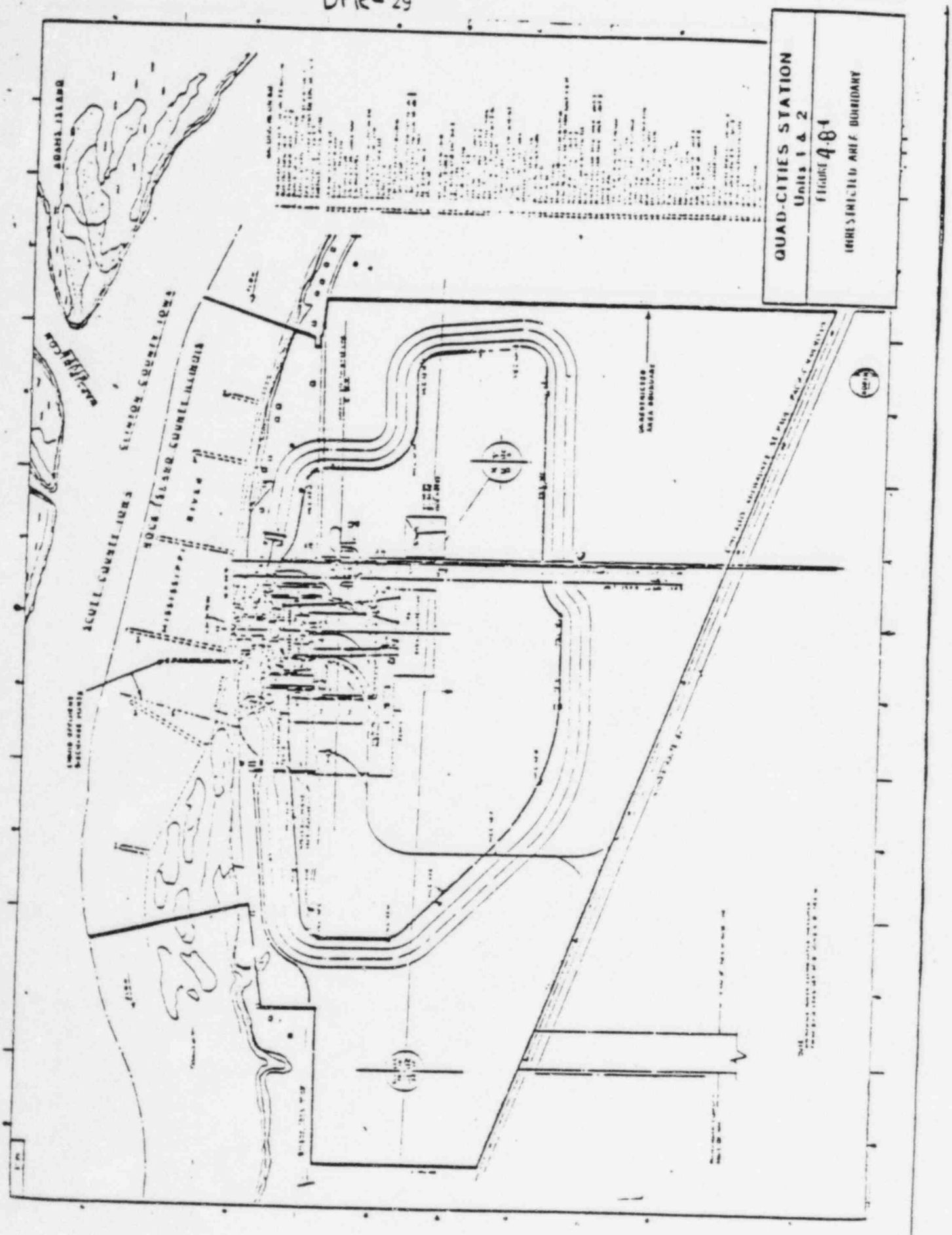

50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance.

2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

4) Proposed changes in Technical Specification NRC operating licenses.

5) Noncompliance with NRC requirements, or of internal procedures, or instructions having nuclear safety significance.

6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

7) Reportable occurrences requiring 24-hour notification to the NRC.

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change.

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Vice-President – Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The Audit Function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance for Operating and to the Staff Assistant to the Manager of Quality Assurance for maintenance quality assurance activities.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within a period of 2 years or less as designated below.

1) Audit of the conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

2) Audit of the adherence to procedures, training and qualification of the station staff at least once per year.

3) Audit of the results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months.

4) Audit of the performance of activities required by the Quality Assurance Program to meet the Criteria of Appendix "B" 10 CFR 50.

5) Audit of the Facility Emergency Plan and implementing procedures.

6) Audit of the Facility Security Plan and implementing procedures.

7) Audit onsite and offsite reviews.

8) Audit the Facility Fire Protection Program and implementing procedures at least once per 24 months.

9) The radiological environmental monitoring program and the results thereof at least once per 12 months.

10) The ODCM and implementing procedures at least once per 24 months.

11) The PCP and implementing procedures for solidification of radioactive waste at least once per 24 months.

12) Report all findings of noncompliance with NRC requirements and recommendations and results each audit to the Station Superintendent, Division Vice President Nuclear Stations, Manager of Quality Assurance, Vice President of Nuclear Operations, and to the Executive Vice President of Construction, Production, and Engineering.

c. Authority

The Manager of Quality Assurance reports to the Executive Vice-President and the Supervisor of the Offsite Review and Investigative Function reports to the General Superintendent of Production Systems Analysis. Either the Manager of Quality Assurance or the Supervisor of the Offsite Review and Investigative Function has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

d. Records

1) Reviews, audits, and recommendations shall be documented and distributed as covered in 6.1.G.1.a and 6.1.G.1.b.

2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the offsite reviews and investigative functions described in Specifications 6.1.G.1.a. Those procedures shall cover the following:

1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function.

2) Use of committees and consultants.

3) Review and approval.

4) Detailed listing of items to be reviewed.

5) Method of (1) appointing personnel, (2) performing reviews, investigations, (3) reporting findings and recommendations of reviews and investigations, (4) approving reports, and (5) distributing reports.

6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor the Offsite Review and Investigative Function, shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated:

a) nuclear power plant technology,
b) reactor operations,
c) utility operations,
d) power plant design,
e) reactor engineering,
f) radiological safety,
g) reactor safety analysis,
h) instrumentation and control,
i) metallurgy,
j) any other appropriate disciplines required by unique characteristics of the facility.

the Supervisor of the Offsite Review and Investigative Function; and (6) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

The responsibilities of the Personnel performing this function are stated below:

1) Review of (1) procedures required by Specification 6.2 and changes thereto and (2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

2) Review of all proposed tests and experiments that affect nuclear safety.

3) Review of all proposed changes to the Technical Specifications.

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

5) Investigation of all noncompliance with NRC requirements and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Vice President-Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function.

6) Review of facility operations to detect potential safety hazards.

7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function.

8) Review of the Station Security Plan and shall submit recommended changes to the Division Vice President-Nuclear Stations.

9) Review of the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Vice President-Nuclear Stations.

10) Review of reportable occurences and actions taken to prevent recurrence.

11) Review of any unplanned on-site release of radioactive material to the environs, including the preparation and forwarding of reports covering evaluation recommendations and disposition of the corrective action to prevent recurrence to the Division Vice President-Nuclear Stations, and to the Supervisor of the Offsite Review and Investigative Function.

12) Review of changes to the PCP and ODCM, and major changes to the radwaste treatment systems.

b. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigations, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Vice President-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

c. Records

1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Vice President-Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent, and the Manager of Quality Assurance.

2) Copies of all records and documentation shall be kept on file at the station.

d. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative function. These procedures shall include the following:

1) Content and method of submission and presentation to the Station Superintendent, Division Vice President-Nuclear Stations, and the Supervisor of the Offsite Review and Investigative Function.

6.2 PLANT OPERATING PROCEDURES

A. Detailed written procedures, including applicable checkoff lists covering items listed below shall be prepared, approved, and adhered to:

1. Normal startup, operation, and shutdown of the reactor, and other systems and components involving nuclear safety of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks, and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity - "Generating Station Emergency Plan" and station emergency and abnormal procedures.

5. Instrumentation operation which could have an affect on the safety of the facility.

6. Preventive and corrective maintenance operations which could have an affect on the safety of the facility.

7. Surveillance and testing requirements.

8. Tests and experiments.

9. Procedure to ensure safe shutdown of the plant.

10. Station Security Plan and implementation procedures.

11. Fire Protection Program implementation.

12. ODCM implementation.

13. PCP implementation.

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. The procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR-20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. 1. Procedures for items identified in Specification 6.2.A and any changes to such procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in the areas of operation of fuel handling and by Maintenance Asst. Supt. and Technical Staff Supervisor in the areas of plant maintenance and plant inspection. Procedures for items identified in Specification 6.2.B and any changes to such procedures shall be reviewed and approved by the Technical Staff Supervisor and the Radiation Chemical Supervisor. At least one person approving each of the above procedures shall hold a valid Senior Operator's License. In addition, these procedures and changes thereto, must have authorization by the Station Superintendent before being implemented.

2. Work and instruction type procedures which implement approved maintenance or modification procedures shall be approved and authorized by the Maintenance Asst. Supt. where the written authority has been provided by the Station Superintendent. The "Maintenance Modification Procedures" utilized for safety related work shall be so approved only if procedures referenced in the "Maintenance/Modification Procedure" have been approved as required by 6.2.A. Procedures which do not fall within the requirement of 6.2.A or 6.2.B. may be approved by the Department Heads.

D. Temporary changes to procedures 6.2.A. and 6.2.B. above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed by the Onsite Review and Investigative Function and approved by the Station Superintendent within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted in accordance with the ODCP Manual.

2. A tabulation shall be submitted on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job function (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report. In addition, any changes to the ODCM shall be submitted with the Monthly Operating Report within 90 days of the effective date of the change.

A report of major change to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the onsite review function. If such change is re-evaluated and not installed, notification of cancellation of the change should be provided to the NRC.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. In general, the importance of an occurrence with respect to safety significance determines the immediacy of reporting required. In some cases, however, the significance of an event may not be obvious at the time of its occurrence. In such cases, the NRC shall be informed promptly of an increased significance in the licensee's assessment of the event. In addition, supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1. Prompt Notification with Written Followup

The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the director of the appropriate regional office or his designate no later than the first working day following the event, with a written followup report within 2 weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented as needed by additional narrative material to provide

Note: This item is intended to provide for reporting of potentially generic problems.

2. Thirty-Day Written Reports

The reportable occurrences discussed below have lesser immediate importance than those described under B.1. above. Such events shall be the subject of written reports to the director of the appropriate regional office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in Items B.2.a. and B.2.b. need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in Item B.1.c. above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

C. Unique Reporting Requirements

1. Radioactive Effluent Release Report (Semi-Annual)

A semi-annual report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June, 1974. Any changes to the PCP shall be included in this report.

2. Environmental Program Data (Annual Report)

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-4) shall be submitted prior to May 1 of each year. The content of the report shall include:

a. Results of all environmental measurements summarized in the format of Regulatory Guide 4.8 Table 1 (December 1975). (Individual sample results will be retained at the Station). In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. Summaries, interpretations, and analysis of trends of the results are to be provided.

b. An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emmissions of radioactivity including the maximum noble gas gamma and beta air doses in the unrestricted area. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

c. Results of the census to determine the locations of nearest residences and of nearby animals producing milk for human consumption.

(Table 4.8-4).

d. The reason for the omission if the nearest dairy to the station is not in the monitoring program (Table 4.8-4).

e. An annual summary of meteorological conditions concurrent with the releases of gaseous effluents in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

f. The results of the Interlaboratory Comparison Program described in section 3.8.D.7.

g. The results of the 40 CFR 190 uranium fuel cycle dose analysis for each calendar year.

h. A summary of the monitoring program, including maps showing sampling locations and tables giving distance and direction of sampling locations from the Station.

3. If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-5 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Director of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter.

   a. When more than one of the radionuclides in Table 4.8-5 are detected in the medium, the reporting level shall have been exceeded if

$$\sum \frac{Ci}{R.L.i} \geq 1$$

   where Ci is the average quarterly concentration of the i^{th} radionuclide in the medium and RL is the reporting level of radionuclide i.

   b. If radionuclides other than those in Table 4.8-5 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

   c. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

4. Special Reports

   Special Reports shall be submitted as indicated in Table 6.6-1.

TABLE 6.6-1
SPECIAL REPORTS

Area	Specification Reference	Submittal Date
a. Secondary containment leak rate test (1)	4.7.C	Upon completion of each test
b. Summary status of fuel performance	1.1 Bases	After each refueling outage.
c. Materials radiation surveillance specimens	4.6.B.2	After each specimen removal and completion of analyses.
d. Evaluation of EGC operation	3.3.F Bases	Upon completion of initial testing.
e. Radioactive Source Leak Testing (2)	4.8.F	Annual Report
f. Special Effluents Reports	3.8.A. 3.8.B. 3.8.D. 6.6.C.3.	30 days following occurrence

Notes

1. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

2. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.8 Offsite Dose Calculation Manual (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

B. Licensee initiated changes to the ODCM may be made provided the change:

1. Shall be submitted to the Commission by inclusion in the Monthly Operating Report pursuant to Specification 6.6.A.3. within 90 days of the date the change(s) was made effective and shall contain:

   a. Sufficiently detailed information to support the change. Information submitted should consist of a package of those pages of the ODCM to be changed together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy of reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the onsite review functions.

2. Shall become effective upon review and acceptance by the onsite review function.

6.9 Process Control Program (PCP)

A. The PCP shall contain the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

B. The PCP shall be approved by the Commission prior to implementation.

C. Licensee initiated changes may be made to the PCP provided the change:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change was made and and shall contain:

a. Sufficiently detailed information to support the change;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation that the change has been reviewed and found acceptable by the onsite review function.

2. Shall become effective upon review and acceptance by the onsite review function.

6.10 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, Solid)

A. Licensee initiated major changes to the radioactive waste systems may be made provided:

1. The change is reported in the Monthly Operating Report for the period in which the evaluation was reviewed by the onsite review function. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to support the reason for the change;

   c. A detailed description of the equipment, components, and process involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and (or quantity of solid waste that differ from those previously predicted in the license application and amendments);

   e. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

   f. An estimate of the exposure to plant operating personnel as a result of the change; and

   g. Documentation of the fact that the change was reviewed and found acceptable by the onsite review function.

2. The change shall become effective upon review and acceptance by by onsite review function.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 14, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-30 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 84, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment shall become effective 6 months after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix "A" Technical Specifications by replacement, addition or deletion of the following pages, as indicated:

i
ii
iii
iv
v
vi
1.0-5*
3.2/4.2-1
3.2/4.2-2
3.2/4.2-3
3.2/4.2-4
3.2/4.2-5
3.2/4.2-5a
3.2/4.2-7
3.2/4.2-8
3.2/4.2-15b*
3.2/4.2-15c*
3.2/4.2-15d
3.2/4.2-17
3.2/4.2-19
3.2/4.2-20
3.8/4.8-1
3.8/4.8-2
3.8/4.8-3
3.8/4.8-4
3.8/4.8-5
3.8/4.8-6
3.8/4.8-6a*
3.8/4.8-7
3.8/4.8-7a
3.8/4.8-8
3.8/4.8-9
3.8/4.8-10
3.8/4.8-11
3.8/4.8-12
3.8/4.8-13
3.8/4.8-14
3.8/4.8-15
3.8/4.8-16

ATTACHMENT TO DPR-30, CONTINUED

3.8/4.8-17
3.8/4.8-18
3.8/4.8-18a - to be deleted
3.8/4.8-19*
3.8/4.8-20*
3.8/4.8-21*
3.8/4.8-21a*
3.8/4.8-22*
3.8/4.8-23*
3.8/4.8-24*
3.8/4.8-25*
3.8/4.8-26*
3.8/4.8-27*
3.8/4.8-28*
3.8/4.8-29*
3.8/4.8-29a*
3.8/4.8-30*
6.1-2
6.1-3
6.1-5
6.2-1
6.6-2
6.6-4
6.6-4a - to be deleted
6.6-5
6.6-5a - to be deleted
6.6-6
6.6-7*
6.7-1
6.8-1*
6.9-1*
6.10-1*

* - New Page

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TECHNCIAL SPECIFICATIONS

APPENDIX A

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TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF TABLES

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

II. DOSE EQUIVALENT I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

JJ. PROCESS CONTROL PROGRAM (PCP) - Contains the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

KK. OFFSITE DOSE CALCULATION MANUAL (ODCM) - Contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints.

LL. CHANNEL FUNCTIONAL TEST (RADIATION MONITOR) - Shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

MM. SOURCE CHECK - The qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

NN. MEMBER(S) OF THE PUBLIC - Shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

3.2/4.2 PROTECTIVE INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

SPECIFICATIONS

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2-1.

A. Primary Containment Isolation Functions

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2-2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Specification 3.5.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

2. The minimum number of operable instrument channels specified in Table 3.2-3 for the rod block monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. If this condition exists for more than 24 hours in a 30-day period, the system shall be tripped.

C. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

D. Refueling Floor Radiation Monitors

1. Except as specified in Specification 3.2.D.2, the two refueling floor radiation monitors shall be operable whenever irradiated fuel or components are present in the fuel storage pool and during refueling or fuel movement operations.

2. One of the two refueling floor radiation monitors may be inoperable for 24 hours. If the inoperable monitor is not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The trip setting for the refueling floor radiation monitors shall be set at a value of 100 mR/hr.

4. Upon loss of both refueling floor radiation monitors while in use, the reactor building ventilation system shall be isolated and the standby gas treatment operated.

D. Refueling Floor Radiation Monitors

The two refueling floor radiation monitors shall be functionally tested and calibrated as indicated in Table 4.2-1. Reactor building ventilation isolation and standby gas treatment system initiation shall be performed at least each operating cycle.

E. Postaccident Instrumentation

The limiting conditions for operation for the instrumentation which is read out in the control room, required for postaccident monitoring are given in Table 3.2-4.

E. Postaccident Instrumentation

Postaccident instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-2.

F. Control Room Ventilation System Isolation

The control room ventilation system is isolated from outside air on a signal of high drywell pressure, low water level, high main steamline flow, or high radiation in either of the reactor building ventilation exhaust ducts. Limiting conditions for operation shall be as indicated in Table 3.2-1 and Specification 3.2.H.1.

F. Control Room Ventilation System Isolation

Surveillance for instrumentation which initiates isolation of control room ventilation shall be as specified in Table 4.2-1.

G. Radioactive Liquid Effluent Instrumentation

The effluent monitoring instrumentation shown in Table 3.2-5 shall be operable with alarm setpoints set to ensure that the limits of Specification 3.8.B are not exceeded. The alarm setpoints shall be determined in accordance with the ODCM.

1. With a radioactive liquid effluent monitoring instrument alarm/trip setpoint less conservative than required, without delay suspend the release of radioactive liquid effluents monitored by the affected instrument, or declare the instrument inoperable, or change the setpoint so it is acceptably conservative.

2. With one or more radioactive liquid effluent monitoring instruments inoperable, take the action shown in Table 3.2-5. Exert best efforts to return the instrument to operable status within 30 days and, if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. This is in lieu of an LER.

G. Radioactive Liquid Effluent Instrumentation

Each radioactive liquid effluent monitoring instrument shown in Table 4.2-3 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequencies shown in Table 4.2-3.

3. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

H. Radioactive Gaseous Effluent Instrumentation

The effluent monitoring instrumentation shown in Table 3.2-6 shall be operable with alarm/trip setpoints set to ensure that the limits of Specification 3.8.A. are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

1. With a radioactive gaseous effluent monitoring instrument alarm/trip setpoint less conservative than required, without delay suspend the release of radioactive gaseous effluents monitored by the affected instrument, or declare the instrument inoperable, or change the setpoint so it is acceptably conservative.

2. With one or more radioactive gaseous effluent monitoring instruments inoperable, take the action shown in Table 3.2-6. Exert best efforts to return the instrument to operable status within 30 days and, if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. This is in lieu of an LER.

H. Radioactive Gaseous Effluent Instrumentation

Each radioactive gaseous radiation monitoring instrument in Table 4.2-4 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequency shown in Table 4.2-4.

3. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to the Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations. Some of the settings on the instrumentation that initiates or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by the protective instrumentation which serves the condition for which isolation is required (this instrumentation is shown in Table 3.2.1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the basis for Specification 3.1 is applicable here.

The low reactor level instrumentation is set to trip at > 8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses (NEDO-24146A, April 1979). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break: the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary isolation are initiated and in time to meet the above criteria. The instrumentation also covers the full spectrum of breaks and meets the above criteria.

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also Group 2 isolation valves include the drywell vent, purge and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with $\leq$ 100 CPS and the detector not full inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level block provide annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident, i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

The radioactive liquid and gaseous effluent instrumentation is provided to monitor the release of radioactive materials in liquid and gaseous effluents during releases. The alarm setpoints for the instruments are provided to ensure that the alarms will occur prior to exceeding the limits of 10 CFR 20.

Table 3.2-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels	Total No. of Channels	Parameter	Action (1)
1	1	Service Water Effluent Gross Activity Monitor	A
1	1	Liquid Radwaste Effluent Flow Rate Monitor	C
1	1	Liquid Radwaste Effluent Gross Activity Monitor	B
1	1	Spray Canal Discharge Blowdown Flow Rate Monitor	C

Notes:

Action A: With less than the minimum number of operable channels, releases via this pathway may continue, provided that at least once per 12 hours grab samples are collected and analyzed for beta or gamma activity at an LLD of less than or equal to 10^{-7} uCi/ml.

Action B: With less than the minimum number of operable channels, effluent releases via this pathway may continue, provided that prior to initiating a release, at least 2 independent samples are analyzed in accordance with Specification 4.9.8.1., and at least 2 members of the facility staff independently verify the release calculation and discharge valving. Otherwise, suspend release of radioactive effluents via this pathway.

Action C: With less than the minimum number of operable channels, 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 utilized to estimate flow.

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Table 3.2-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels (1)	Total No. of Channels	Parameter	Action (2)
1	2	SJAE Radiation Monitors	D
1	2	Main Chimney Noble Gas Activity Monitor	A
1	1	Main Chimney Iodine Sampler	C
1	1	Main Chimney Particulate Sampler	C
1	1	Reactor Bldg. Vent Sampler Flow Rate Monitor	B
1	1	Reactor Bldg. Vent Iodine Sampler	C
1	1	Reactor Bldg. Vent Particulate Sampler	C
1	1	Main Chimney Sampler Flow Rate Monitor	B
1	1	Main Chimney Flow Rate Monitor	B
1	2	Reactor Bldg. Vent Noble Gas Monitor	E

Notes

(1) For SJAE monitors, applicable during SJAE operation. For other instrumentation, applicable at all times.

(2) Action A: With the number of operable channels less than the minimum requirement, effluent releases via this pathway may continue, provided grab samples are taken at least once per 8 hour shift and these samples are analyzed within 24 hours.

Action B: With the number of operable channels less than the minimum required, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per 4 hours.

Action C: With less than the minimum channels operable, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment, as required in Table 4.8-1.

Action D: With less than the minimum channels operable, gases from the main condenser off-gas system may be released to the environment for up to 72 hours provided the off-gas system is not bypassed and at least one chimney monitor is operable; otherwise, be in hot stand-by in 12 hours.

Action E: With less than the minimum channels operable, immediately suspend release of radioactive effluents via this pathway.

Table 4.2-1 (Cont'd)

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
HPCI Isolation			
1. Steamline high flow	(1)	Once/3 months	None
2. Steamline area high temperature	Refueling Outage	Refueling Outage	None
3. Low reactor pressure	(1)	Once/3 months	None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refuel Floor Rad. Monitors	(1)	Once/3 months	Once/day
Control Room Ventilation System Isolation			
1. Reactor low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Main steamline high flow	(1)	Once/3 months	Once/day

Notes

1. Initially once per month until exposure hours (M as defined on Figure 4.1-1) are 2.0×10^5; thereafter, according to Figure 4.1-1 with an interval not less than 1 month nor more than 3 months. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of Quad-Cities 1/2.
2. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable, or are tripped.
3. This instrumentation is excepted from the functional test definition. The functional test shall consist of injecting a simulated electrical signal into the measured channel.
4. This instrument channel is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every 3 months.
5. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week.
6. The positioning mechanism shall be calibrated every refueling outage.
7. Logic system functional tests are performed as specified in the applicable section for these systems.
8. Functional test shall include verification of operation of the degraded voltage 5-minute timer and 7-second inherent timer.
9. Verification of the time delay setting of $3 \leq \tau \leq 10$ seconds shall be performed during each refueling outage.

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Table 4.2-3

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)	Calibration (1) (3)	Functional Test (1) (2)	Source Check (1)
Liquid Radwaste Effluent Gross Activity Monitor	D	R	Q (7)	(6)
Service Water Effluent Gross Activity Monitor	D	R	Q (7)	R
Liquid Radwaste Effluent Flow Rate Monitor	(4)	R	NA	NA
Blowdown Flow Rate Monitor Spray Canal Discharge	(4)	R	NA	NA

Notes:

(1) D = once per 24 hours
M = once per 31 days
Q = once per 92 days
R = once per 18 months
S = once per 6 months

(2) The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

a. Instrument indicates levels above the alarm setpoint.
b. Circuit failure.
c. Instrument indicates a downscale failure.
d. Instrument controls not set in OPERATE mode.

(3) Calibration shall include performance of a functional test.

(4) Instrument Check to verify flow during periods of release.

(5) Calibration shall include performance of a source check.

(6) Source check shall consist of observing instrument response during a discharge.

(7) Functional test may be performed by using trip check and test circuitry associated with the monitor chassis.

QUAD-CITIES
DPR-30

Table 4.2-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Mode(2)	Instrument Check(1)	Calibration(1)(4)	Functional Test(1)(3)	Source Check(1)
Main Chimney Noble Gas Activity Monitor	B	D	R	Q	M
Main Chimney Sampler Flow Rate Monitor	B	D	R	Q(6)	NA
Reactor Bldg. Vent Sampler Flow Rate Monitor	B	D	R	Q(6)	NA
Main Chimney Flow Rate Monitor	B	D	R	Q	NA
Reactor Bldg. Vent Activity Monitor	B	D	R	Q	Q
SJAE Activity Monitor	A	D	R	Q	R
Main Chimney Iodine and Particulate Sampler	B	D(5)	NA	NA	NA
Reactor Bldg. Vent Iodine and Particulate Sampler	B	D(5)	NA	NA	NA

Notes

(1) D = once per 24 hours
M = once per 31 days
Q = once per 92 days
R = once per 18 months

(2) A = during SJAE operation
B = at all times

(3) The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable:

a. Instrument indicates levels above the alarm setpoint
b. Circuit failure
c. Instrument indicates a downscale failure
d. Instrument controls not set in OPERATE mode.

(4) Calibration shall include performance of a functional test.

(5) Instrument check to verify operability of sampler; that the sampler is in-place and functioning properly.

(6) Functional test shall be performed on local switches providing low flow alarm.

3.8/4.8 RADIOACTIVE EFFLUENTS

Limiting Conditions for Operation	Surveillance Requirements
Applicability:	Applicability:
Applies to the radioactive effluents from the plant.	Applies to the periodic measurements radioactive effluents.

Specifications

A. Gaseous Effluents

1. The dose rate in unrestricted areas(at or beyond the site boundary, Figure 4.8-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following:

   a. For Noble Gases:

      (1) Less than 500 mrem/ year to the whole body.

      (2) Less than 3000 mrem/ year to the skin.

   b. For iodine-131, for iodine -133, and for all radionuclides in particulate form with half-lives greater than 8 days less than 1500 mrem/year.

   c. If the dose rates exceed the above limits, without delay decrease the release rates to bring the dose rates within the limits, and provide prompt notification to the Commission (6.6.B.1.)

2. The air dose in unrestricted areas(at or beyond the site boundary)due to Noble Gases released in gaseous effluents from the unit shall be limited to the following:

   a. For gamma radiation:

      (1) Less than or equal to 5 mrad during any calendar quarter.

A. Gaseous Effluents

1. The dose rates due to radioactive materials released in gaseous effluents from the site shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 4.8-1. The dose rates are calculated using methods prescribed in the Off-Site Dose Calculation Manual (ODCM).

2. The air dose due to releases of radioactive noble gases in gaseous effluents shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in sections A and B of Table 4.8-1. The allocation of effluents between units having shared effluent control systems and the air doses are determined using methods prescribed in the ODCM at least once every 31 days.

(2) Less then or equal to 10 mrad during any calendar year.

b. For Beta radiation:

(1) Less than or equal to 10 mrad during any calendar quarter

(2) Less than or equal to 20 mrad during any calendar year.

c. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases are in compliance with 3.8.A.2.a. & b. This is in lieu of a Licensee Event Report.

d. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding the limits of Specification 3.8.A.2.a. or 3.8.A.2.b., prepare and submit a Special Report to the Commission within 30 days and limit the subsequent releases such that the doses or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise, obtain a

variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use the methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

3. The dose to a member of the public in unrestricted areas (at or beyond the site boundary) from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the unit shall be limited to the following:

   a. Less than or equal to 7.5 mrem to any organ during any calendar quarter.

   b. Less than or equal to 15 mrem to any organ during any calendar year.

   c. With the calculated dose from the release of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken and the proposed actions to be taken to ensure that future releases are in compliance with 3.8.A.3. a. & b. This is in lieu of a Licensee Event Report.

3. The dose to a member of the public due to releases of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified Table 4.8-1.

   For radionuclides not determined in each batch or weekly composite, the dose contribution to the current calendar quarter cumulative summation may be estimated by assuming an average monthly concentration based on the previous monthly or quarterly composite analyses. However, for reporting purposes, the calculated dose contributions shall be based on the actual composite analyses when possible.

   The allocation of effluents between units having shared effluent control systems and the doses are determined using the methods prescribed in the ODCM at least once every 31 days.

d. With the calculated dose from the release of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents exceeding the limits of Specification 3.8.A.3.a. or 3.8.A.3.b., prepare and submit a Special Report to the Commission within 30 days and limit subsequent releases such that the dose or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or equal to 25 mrem to the total body or organ (except the thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use the methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

4.	Off-Gas System a. At all times during processing for discharge to the environs, process and control equipment provided to reduce the amount or concentration of radioactive materials shall be operated. b. The above specification shall not apply for the Off-Gas Charcoal Absorber Beds below 30 percent of rated thermal power.	4. Off-Gas System Doses due to treated gases released to unrestricted areas at or beyond the site boundary shall be projected at least once per 31 days in accordance with the ODCM.
5.	Explosive Gas Mixture a. The concentration of hydrogen in the off-gas hold up system, downstream of the recombiner shall be limited by having a recombiner operable within the allowable band of the base-line plot of recombiner outlet temperature vs. reactor power, whenever the reactor is operating at a pressure greater than 900 psig. b. The recombiner may be inoperable for 48 hours.	5. Explosive Gas Mixture Once per 8 hours verification will be made that the unit is operating within the allowable band of the base-line plot of recombiner outlet temperature vs. reactor power.
6.	With either the recombiners inoperable, or all charcoal beds bypassed for more than 7 days in a calendar quarter while operating above 30 percent of rated thermal power, prepare and submit to the Commission within 30 days a special report which includes the following information:	

a. Identification of the defective equipment.

b. Cause of the defective equipment

c. Action(s) taken to restore the equipment to an operating status.

d. Length of time the above requirements were not satisfied.

e. Volume and curie content of the waste discharged which was not processed by the inoperable equipment but which required processing.

f. Action(s) taken to prevent a recurrence of equipment failures.

This is in lieu of a Licensee Event Report.

7. The release rate of the sum of the activities from the noble gases measured at the main condenser air ejector shall be limited to less than or equal to 100 microcuries/sec per MWt (after 30 minutes decay) at all times. With the release rate of the sum of the activities from noble gases at the main condenser air ejector exceeding 100 microcuries/sec per MWt (after 30 minutes decay), restore the release rate to within its limits within 72 hours, or be in at least HOT STANDBY within the next 12 hours.

7. The radioactivity rate of nobles ga at (near) the outlet of the main co denser air ejector shall be contin ously monitored in accordance with Specification 3.2.H. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be dete mined to be within the limits of Specification 3.8.H. at the follow frequencies by performing an isotop analysis of a representative sample of gases taken at the recombiner ou let, or at the air ejector outlet i the recombiner is bypassed.

a. At least once per 31 days.

b. Within 4 hours following an in- crease, as indicated by the mai condenser air ejector noble gas activity monitor, of greater th 50%, after factoring out increa due to changes in thermal power level and off-gas flow, in the nominal steady-state fission ga release from the primary coolan

B. Liquid Effluents

1. The concentration of radioactive material released from the site to unrestricted areas (at or beyond the site boundary, figure 4.8-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 with the Table 4.8-2 values representing the MPC's for noble gases.

   With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits, without delay decrease the release rate of radioactive materials and/or increase the dilution flow rate to restore the concentration to within the above limits.

B. Liquid Effluents

1. The concentration of radioactive material in unrestricted areas shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 4.8-3. The sample analysis results will be used with the calculational methods in the ODCM to determine that the concentrations are within the limits of Specification 3.8.B.1.

2. The dose or dose commitment above background to a member of the public from radioactive materials in liquid effluents released to unrestricted areas (at or beyond the site boundary) from the site shall be limited to the following:

   a. During any calendar quarter:

      (1) Less than or equal to 3 mrem to the whole body.

      (2) Less than or equal to 10 mrem to any organ.

   b. During any calendar year:

      (1) Less than or equal to 6 mrem to the whole body.

      (2) Less than or equal to 20 mrem to any organ.

   c. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken and the proposed actions to be taken to ensure that future releases are in compliance with 3.8.B.2.a. & b. This is in lieu of a Licensee Event Report.

   d. With the calculated dose from the release of radioactive materials in liquid effluents exceeding the limits of Specification 3.8.B.2.a. or 3.8.B.2.b., prepare and submit a Special Report to the Commission within 30 days and limit the subsequent releases such that the dose or dose commitment to a member of the public from all uranium fuel cycle sources is limited to less than or

2. 'a. The dose contributions from measured quantities of radioactive material shall be determined by calculation at least once per 31 days and a cumulative summation of these total body and any organ doses shall be maintained for each calendar quarter.

   b. Doses computed at the nearest community water system will consider only the drinking water pathway and shall be projected using the methods prescribed in the ODCM at least once per 92 days.

equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise obtain a variance from the Commission to permit releases which exceed the 40 CFR Part 190 Standard. The radiation exposure analysis contained in the Special Report shall use methods prescribed in the ODCM. This report is in lieu of a Licensee Event Report.

e. With the projected annual whole body or any internal organ dose computed at the nearest downstream community water system is equal to or exceeds 2 mrem from all radioactive materials released in liquid effluents from the Station, prepare and submit a Special Report within 30 days to the operator of the community water system. The report is prepared to assist the operator in meeting the requirements of 40 CFR 141: EPA Primary Drinking Water Standards. A copy of this report will be sent to the NRC. This is in lieu of a Licensee Event Report.

3. At all times during processing prior to discharge to the environs, process and control equipment provided to reduce the amount or concentration of radioactive materials shall be operated when the projected dose due to liquid effluent releases to unrestricted areas (see Figure 4.8-1), when averaged over 31 days, exceeds 0.13 mrem to the total body or 0.42 mrem to any organ.

4. If liquid waste has to be or is being discharged without treatment as required above, prepare and submit to the Commission within 30 days, a report which includes the following information:

   a. Identification of the defective equipment.

   b. Cause of the defective equipment.

   c. Action(s) taken to restore the equipment to an operating status.

   d. Length of time the above requirements were not satisfied.

   e. Volume and curie content of the waste discharged which was not processed by the appropriate equipment but which required processing.

   f. Action(s) taken to prevent a recurrence of equipment failures.

   This is in lieu of a Licensee Event Report.

3. Liquid Waste Treatment

   a. Doses due to liquid releases to unrestricted areas (at or beyond the site boundary) shall be projected at least once per 31 days in accordance with the ODCM.

C. Mechanical Vacuum Pump	C. Mechanical Vacuum Pump
1. The mechanical vacuum shall be capable of being isolated and secured on a signal of main steam high radiation or shall be isolated and secured whenever the main steam isolation valves are open.	At least once during each operating cycle, automatic securing and isolation of the mechanical vacuum pump shall be verified.

D. Environmental Monitoring Program

1. The environmental monitoring program given in Table 4.8-4 shall be conducted except as specified below.

2. With the radiological environmental monitoring program not being conducted as specified in Table 4.8-4, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, contractor omission which is corrected as soon as discovered, malfunction of sampling equipment, or if a person who participates in the program goes out of business. If the equipment malfunctions, corrective actions shall be completed as soon as practical. If a person supplying samples goes out of business, a replacement will be found as soon as possible. All deviations from the sampling schedule shall be described in the annual report.

3. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in the ODCM exceeding the limits of Table 4.8-5 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter, a Special Report which includes an evaluation of any release conditions, environmental

D. Environmental Monitoring Program

1. The radiological environmental monitoring samples shall be collected pursuant to Table 4.8-3/4 from the locations specified in the ODCM, and shall be analyzed pursuant to the requirements of Table 4.8-6.

2. The results of analyses performed on radiological environmental monitoring samples shall be summarized in the Annual Radiological Environmental Operating Report.

3. The land use census shall be conducted at least once per twelve months between the dates of June 1 and October 1 by a door-to-door survey, aerial survey, road survey, or by consulting local agriculture authorities.

factors or other aspects which caused the limits of Table 4.8-5 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

4. With milk samples unavailable from one or more of the sample locations required by Table 4.8-4, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the Annual Radiological Environmental Operating report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

4. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

5. A census of nearest residences and of animals producing milk for human consumption shall be conducted annually (during the grazing season for animals) to determine their location and number with respect to the site. The nearest residence in each of the 16 meteorological sectors shall also be determined within a distance of five miles. The census shall be conducted under the following conditions:

   a. Within a 2-mile radius from the plant site, enumeration of animals and nearest residences by a door-to-door or equivalent counting technique.

5. The results of the analyses performed as part of the required crosscheck program shall be included in the Annual Radiological Environmental Operating Report. The analyses shall be done in accordance with the ODCM.

c. Within a 5-mile radius, enumeration of animals by using referenced information from county agricultural agents or other reliable sources.

6. With a land use census identifying location(s) of animals which yield(s) ar ODCM calculated dose or dose commitment greater than the values currently being calculated in Specification 4.8.A.3, the new location(s) shall be added to the radiological environmental monitoring program with 30 days, if possible.

   The sampling location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

7. Radiological analyses shall be performed on samples representative of those in Table 4.8-4, supplied as a part of the Inter-laboratory Comparison Program which has been approved by the NRC.

8. With analyses not being performed as required, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

E. Solid Rad'oactive Waste

1. The solid radwaste system shall be used as applicable in accordance with the PCP to process wet radioactive wastes to meet shipping and burial ground requirements.

2. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive waste from the site.

E. Solid Radioactive Waste

1. The PCP shall specify the method and frquency to verify solidification of radioactive waste. Actions to be taken if solidification is not verified shall also be specified in the PCP.

F. Miscellaneous Radioactive Materials Sources

Source Leakage Test

Specification

Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and either decontaminated and repaired or disposed of in accordance with Commission Regulations.

A complete inventory of radioactive materials in the licensee's possession shall be maintained current at all times.

F. Miscellaneous Radioactive Materials Sources

Each sealed source shall be tested for leakage and/or contamination by the licensee or by other persons specifically authorized by the Commission or an Agreement state. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

Each category of sealed sources shall be tested at the frequency described below:

1. Sources in use (excluding startup previously subjected to core flux) - At least once per 6 months for all sealed sources containing radioactive material:

   a. With a half-life greater than 30 days (excluding Hydrogen 3), and

   b. In any form other than gas.

2. Stored sources not in use - Each sealed source shall be tested prior to the use or transfer to another licensee unless tested within the previous 6 months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.C.3 if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

G. In the event a limiting condition for operation and/or associated action requirements identified in sections 3.8.A. through 3.8.E., and 4.8.A. through 4.8.E. cannot be satisfied because of circumstances in excess of those addressed in the specifications, no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

3.8/4.8.A.1 GASEOUS EFFLUENTS - DOSE

This specification is provided to ensure that the dose at the unrestricted area boundary from gaseous effluents from the units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the unrestricted area boundary to less than or equal to 500 mrem/year to the total body or to not less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to not less than or equal to 1500 mrem/year for the nearest cow to the plant. For purposes of calculating doses resulting from airborne releases the main chimney is considered to be an elevated release point, and the reactor vent stack is considered to be a mixed mode release point.

3.8/4.8.A.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational prodecures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. The ODCM equations provide for determining the air doses at the unrestricted boundary based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

3.8/4.8.A.3 DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implements the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods approved by NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these specifications were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man and 3) deposition onto grassy areas where milk animals graze with consumption of the milk by man.

3.8/4.8.A.4 GASEOUS WASTE TREATMENT

The OPERABILITY of the gaseous waste treatment which reduces amounts or concentrations of radioactive materials ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be operable when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50.

3.8/4.8.A.5. EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the off gas system is minimized in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

LIQUID EFFLUENTS

3.8/4.8.B.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, column 2. The concentration limit for noble gases, MPC in air (submersion), was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

3.8/4.8.B.2. DOSE

This specification is provided to implement the requirements of Sections IIA, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A. of Appendix I. The statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April 1977. NUREG-0113 provides methods for dose calculations consistent with Reg Guide 1.109 and 1.113.

3.8/4.8.B.3 LIQUID WASTE TREATMENT

The operability of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50.

3.8/4.8.D.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. Program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.8-6 are state-of-the-art for routine environmental measurements in industrial laboratories. The specified lower limits of detection for I-131 in water, milk and other food products correspond to approximately one-quarter of the Appendix I to 10 CFR Part 50 design objective dose-equivalent of 15 mrem/year for atmospheric releases and 10 mrem/year for liquid releases to the most sensitive organ and individual. They are based on the assumptions given in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", October 1977, except the change for an infant consuming 330 liter/year of drinking water instead of 510 liters/year.

3.8/4.8.D.6 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

3.8/4.8.D.7 CROSSCHECK PROGRAM

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

3.8/4.8.C MECHANICAL VACUUM PUMP

The purpose of isolating the mechanical vacuum line is to limit release of activity from the main condenser. During an accident, fission products would be transported from the reactor through the main steamline to the main condenser. The fission product radioactivity would be sensed by the main steamline radioactivity monitors which initiate isolation.

3.8/4.8.F. MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

The objective of this specification is to assure that leakage from byproduct, source and special nuclear material sources does not exceed allowable limits. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium.

3.8/4.8.E. SOLID RADIOACTIVE WASTE

The operability of the solid radioactive waste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped off-site. This specification implements the requirements of 10 CFR 50.36a. and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

TABLE 4.8-1
RADIOACTIVE GASEOUS WASTE SAMPLING AND
ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (uci/ml)
A. Main Chimney Reactor Bldg.. Vent Stack	M Grab Sample	M[b]	Principal Gamma Emitters[e]	1×10^{-4}
		M	Tritium	1×10^{-6}
B. All Release Types as Listed in A Above	Continuous[d]	W[c] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[d]	W[c] Particulate Sample	Principal Gamma Emitters[e] (I-131, others)	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	SR-89	1×10^{-11}
			SR-90	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
C. Main Chimney	Continuous[d]	Noble Gas Monitor	Noble Gases	1×10^{-6}
D. Reactor Bldg Vent Stack	Continuous[d]	Noble Gas Monitor	Noble Gases	1×10^{-4}

TABLE 4.8-1 (Continued)
TABLE NOTATION

a. The lower limit of detection (LLD) is defined in table notation A. of Table 4.8-6.

b. Sampling and analyses shall also be performed following shutdown, startup, or a thermal power change exceeding 20 percent of rated thermal power in 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least once per 7 days and the analyses completed within 48 hours after removal from the sampler. Sampling shall also be performed within 24 hours following each shutdown, startup, or thermal power level change exceeding 20% of rated thermal power in one hour. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

d. The ratio of sample flow rate to the sampled stream flow rate shall be known.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions, and Mn-54, Fe-59, Co-60, Zn-65, Co-58, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. Other peaks which are measurable and identifiable by gamma ray spectrometry, together with the above nuclides, shall be also identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

TABLE 4.8-2

MAXIMUM PERMISSIBLE CONCENTRATION OF
DISSOLVED OR ENTRAINED NOBLE GASES
RELEASED FROM THE SITE TO UNRESTRICTED AREAS
IN LIQUID WASTE

NUCLIDE	MPC(uCi/ml)*
Kr-58m	2×10^{-4}
Kr-85	5×10^{-4}
Kr-87	4×10^{-5}
Kr-88	9×10^{-5}
Ar-41	7×10^{-5}
Xe-131m	7×10^{-4}
Xe-133m	5×10^{-4}
Xe-133	6×10^{-4}
Xe-135m	2×10^{-4}
Xe-135	2×10^{-4}

* Computed from Equation 20 of ICRP Publication 2 (1959), adjusted for infinite cloud submersion in water, and R = 0.01 rem/week, density = 1.0 g/cc abd Pw/Pt = 1.0.

TABLE 4.8-3

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uci/ml)
A. Batch Waste Release Tanks	Prior to Each Batch	Prior to Each Batch	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	Prior to Each Batch	M Composite[b]	Gross Alpha	1×10^{-7}
			H-3	1×10^{-5}
	Prior to Each Batch	Q Composite[b]	Fe-55	1×10^{-6}
			Sr-89, Sr-90	5×10^{-8}
	Prior to One Batch/M	M	Dissolved & Entrained Gases[f] (Gamma Emitters)	1×10^{-5}
B. Plant Continuous Releases[d]	M[c] (Grab Sample)	M[c]	I-131	1×10^{-6}
			Principal Gamma Emitters[e]	5×10^{-7}
			Dissolved & Entrained Gases[f] (Gamma emmiters)	1×10^{-5}
			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Q[c] (Grab Sample)	Q[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.8-3 (Continued)
TABLE NOTATION

a. The LLD is defined in Notation **A** of Table 4.8-6.

b. A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. If the alarm setpoint of the service water effluent monitor as determined in the ODCM is exceeded, the frequency of analysis shall be increased to daily until the condition no longer exists.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the release.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-60, Zn-65, Co-58, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. Other peaks which are measurable and identifiable by gamma ray spectrometry together with the above nuclides, shall be also identified and reported when the actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

f. The dissolved and entrained gases (gamma emitters) for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138. Other dissolved and entrained gases (gamma emitters) which are measurable and identifiable by gamma-ray spectrometry, together with the above nuclides, shall also be identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.

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TABLE 4.8-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Particulates	16 locations	Continuous operation of sampler for a week	Gross beta and gamma isotopic as specified in ODCM.
b. Radioiodine	16 locations	Continuous operation of sampler for two weeks	I-131 as specified in ODCM.
2. DIRECT RADIATION	Forty Locations (Minimum of two TLDs per packet)	Quarterly	

*Sample locations are described in the ODCM.

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TABLE 4.8-4 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Public Water	2 Locations	Monthly composite of weekly collected samples	Gamma isotopic analysis of each composite sample
b. Sediment	1 downstream location in receiving body of water	Annually	Gamma isotopic analysis of each sample
c. Plant Cooling Water	Intake, Discharge	Weekly composite	Gross Beta analysis of each sample

*Sample locations are described in the ODCM

TABLE 4.8-4 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples and Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INJESTION			
a. Milk	2 Locations	At least once weekly when animals are on pasture; at least once per month at other times.	I-131 analysis of each sample
b. Fish	1 location in receiving body of water	Semi-annually	Gamma isotopic analysis on edible portions

*Sample locations are described in the ODCM

TABLE 4.8-5

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

alysis	Water	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg,wet)	Milk (pCi/l)	Food Products (pCi/Kg,wet)
	Reporting Levels				
3	2×10^4(a)				
-54	1×10^3		3×10^4		
-59	4×10^2		1×10^4		
-58	1×10^3		3×10^4		
-60	3×10^2		1×10^4		
-65	3×10^2		2×10^4		
-Nb-95	4×10^2				
131	2	0.9		3	1×10^2
-134	30	10	1×10^3	60	1×10^3
-137	50	20	1×10^3	70	2×10^3
-La-140	2×10^2			3×10^2	

) for drinking water samples. This is 40 CFR Part 141 value.

TABLE 4.8-6

PRACTICAL LOWER LIMITS OF DETECTION (LLD)
FOR STANDARD ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

Sample Media	Analysis	LLD[A,B] (4.66σ)	Units
Airborne "Particulate"	Gross Beta +	0.01	pCi/m^3
	Gamma Isotopic	0.01	pCi/m^3
Airborne I-131	Iodine-131	0.10	pCi/m^3
Milk/Public Water	I-131	5°	pCi/l
	Cs-134	10	pCi/l
	Cs-137	10 △	pCi/l
	Tritium	200	pCi/l
	Gross Beta +	5	pCi/l
	Gamma Isotopic	20	pCi/l/nuclide
Sediment	Gross Beta +	2	pCi/g dry
	Gamma Isotopic	0.2	pCi/g dry
Fish Tissue	I-131 - Thyroid	0.1	pCi/g wet
	Cs-134, 137	0.1	pCi/g wet
	Gross Beta +	1.0	pCi/g wet
	γ Isotopic	0.2	pCi/g wet

° 0.5 pCi/l on milk samples collected during the pasture season.

\+ Referenced to Cs-137

△ 5.0 pCi/l on milk samples

TABLE 4.8-6 (Continued)
TABLE NOTATION

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

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

$$LLD = \frac{4.55 \cdot s_b}{A \cdot E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t) \cdot t}$$

Where:

LLD is the "a priori" lower limit of detection for a blank sample or background analysis as defined above (as pCi per unit mass or volume).

s_b is the square root of the background count or of a blank sample count; is the estimated standard error of a background count or a blank sample count as appropriate (in units of counts).

E is the counting efficiency (as counts per disintegration).

A is the number of gamma-rays emitted per disintegration for gamma-ray radionuclide analysis (A = 1.0 for gross alpha and tritium measurements).

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

2.22 is the number of disintegrations per minute per picocurie.

Y is the fractional radio-chemical yield when applicable (otherwise Y = 1.0).

λ is the radioactive decay constant for the particular radionuclide (in units of reciprocal minutes).

Δt is the elapsed time between the midpoint of sample collection and the start time of counting. (Δt = 0.0 for environmental samples and for gross alpha measurements).

t is the duration of the count (in units of minutes).

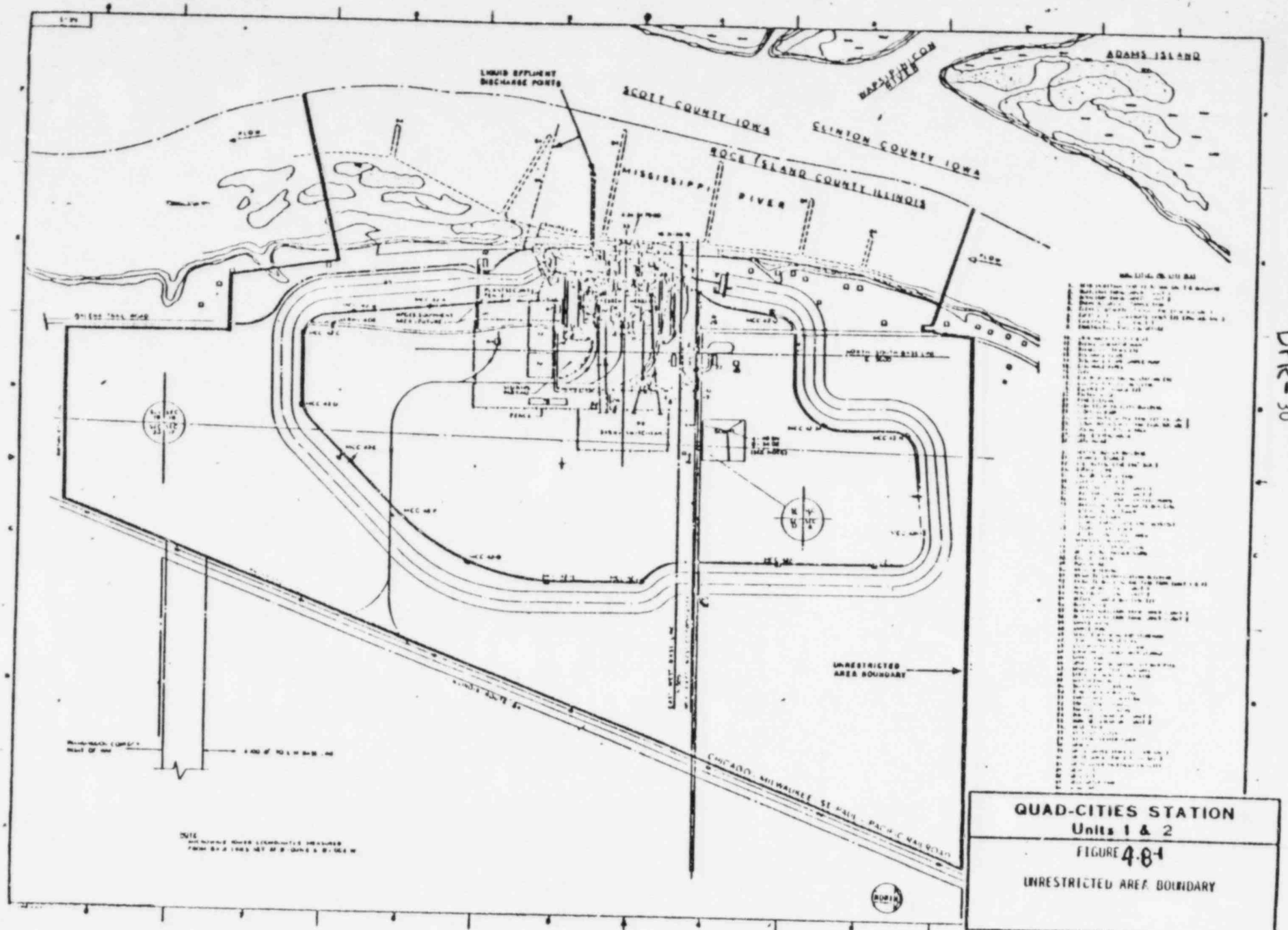
The value of "Sb" used in the calculation of the LLD for a detection system shall be based on an actual observed background count or a blank sample count (as appropriate) rather than on an unverified theoretically predicted value. Typical values of "E", "V", "Y", "t", and "Δt" shall be used in the calculation.

For gamma-ray radionuclide analyses the background counts are determined from the total counts in the channels which are within plus or minus one FWHM (Full Width at Half Maximum) of the gamma-ray photopeak energy normally used for the quantitative analysis for that radionuclide. Typical values of the FWHM shall be used in the calculation.

TABLE 4.8-6 (Continued)
TABLE NOTATION

The LLD for all measurements is defined as an "A priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular sample measurement.

B. Other radionuclides which are measureable and identifiable by gamma-ray spectrometry, together with the nuclides indicated in Table 4.8-6, shall also be identified and reported when an actual analysis is performed on a sample. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide.



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Units 1 & 2

FIGURE 4.8-1

UNRESTRICTED AREA BOUNDARY

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50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance.

2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

4) Proposed changes in Technical Specification NRC operating licenses.

5) Noncompliance with NRC requirements, or of internal procedures, or instructions having nuclear safety significance.

6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

7) Reportable occurrences requiring 24-hour notification to the NRC.

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change.

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Vice-President - Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The Audit Function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance for Operating and to the Staff Assistant to the Manager of Quality Assurance for maintenance quality assurance activities.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within a period of 2 years or less as designated below.

1) Audit of the conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

2) Audit of the adherence to procedures, training and qualification of the station staff at least once per year.

3) Audit of the results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months.

4) Audit of the performance of activities required by the Quality Assurance Program to meet the Criteria of Appendix "B" 10 CFR 50.

5) Audit of the Facility Emergency Plan and implementing procedures.

6) Audit of the Facility Security Plan and implementing procedures.

7) Audit onsite and offsite reviews.

8) Audit the Facility Fire Protection Program and implementing procedures at least once per 24 months.

9) The radiological environmental monitoring program and the results thereof at least once per 12 months.

10) The ODCM and implementing procedures at least once per 24 months.

11) The PCP and implementing procedures for solidification of radioactive waste at least once per 24 months.

12) Report all findings of noncompliance with NRC requirements and recommendations and results each audit to the Station Superintendent, Division Vice President Nuclear Stations, Manager of Quality Assurance, Vice President of Nuclear Operations, and to the Executive Vice President of Construction, Production, and Engineering.

c. Authority

The Manager of Quality Assurance reports to the Executive Vice-President and the Supervisor of the Offsite Review and Investigative Function reports to the General Superintendent of Production Systems Analysis. Either the Manager of Quality Assurance or the Supervisor of the Offsite Review and Investigative Function has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

d. Records

1) Reviews, audits, and recommendations shall be documented and distributed as covered in 6.1.G.1.a and 6.1.G.1.b.

2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the offsite reviews and investigative functions described in Specifications 6.1.G.1.a. Those procedures shall cover the following:

1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function.

2) Use of committees and consultants.

3) Review and approval.

4) Detailed listing of items to be reviewed.

5) Method of (1) appointing personnel, (2) performing reviews, investigations, (3) reporting findings and recommendations of reviews and investigations, (4) approving repor and (5) distributing reports.

6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor the Offsite Review and Investigative Function, shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated:

a) nuclear power plant technology,
b) reactor operations,
c) utility operations,
d) power plant design,
e) reactor engineering,
f) radiological safety,
g) reactor safety analysis,
h) instrumentation and control,
i) metallurgy,
j) any other appropriate disciplines required by unique characteristics of the facility.

the Supervisor of the Offsite Review and Investigative Function; and (6) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

The responsibilities of the Personnel performing this function are stated below:

1) Review of (1) procedures required by Specification 6.2 and changes thereto and (2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

2) Review of all proposed tests and experiments that affect nuclear safety.

3) Review of all proposed changes to the Technical Specifications.

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

5) Investigation of all noncompliance with NRC requirements and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Vice President-Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function.

6) Review of facility operations to detect potential safety hazards.

7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function.

8) Review of the Station Security Plan and shall submit recommended changes to the Division Vice President-Nuclear Stations.

9) Review of the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Vice President-Nuclear Stations.

10) Review of reportable occurences and actions taken to prevent recurrence.

11) Review of any unplanned on-site release of radioactive material to the environs, including the preparation and forwarding of reports covering evaluation recommendations and disposition of the corrective action to prevent recurrence to the Division Vice President-Nuclear Stations, and to the Supervisor of the Offsite Review and Investigative Function.

12) Review of changes to the PCP and ODCM, and major changes to the radwaste treatment systems.

b. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigations, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Vice President-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

c. Records

1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Vice President-Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent, and the Manager of Quality Assurance.

2) Copies of all records and documentation shall be kept on file at the station.

d. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative function. These procedures shall include the following:

1) Content and method of submission and presentation to the Station Superintendent, Division Vice President-Nuclear Stations, and the Supervisor of the Offsite Review and Investigative Function.

6.1 PLANT OPERATING PROCEDURES

A. Detailed written procedures, including applicable checkoff lists covering items listed below shall be prepared, approved, and adhered to:

1. Normal startup, operation, and shutdown of the reactor, and other systems and components involving nuclear safety of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks, and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity - "Generating Station Emergency Plan" and station emergency and abnormal procedures.

5. Instrumentation operation which could have an affect on the safety of the facility.

6. Preventive and corrective maintenance operations which could have an effect on the safety of the facility.

7. Surveillance and testing requirements.

8. Tests and experiments.

9. Procedure to ensure safe shutdown of the plant.

10. Station Security Plan and implementation procedures.

11. Fire Protection Program implementation.

12. ODCM implementation.

13. PCP implementation.

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. The procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR-20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. 1. Procedures for items identified in Specification 6.2-A and any changes to such procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in the areas of operation or fuel handling, and by Maintenance Asst. Supt. and Technical Staff Supervisor in the areas of plant maintenance and plant inspection. Procedures for items identified in Specification 6.2.B and any changes to such procedures shall be reviewed and approved by the Technical Staff Supervisor and the Radiation Chemical Supervisor. At least one person approving each of the above procedures shall hold a valid senior operator's license. In addition, these procedures and changes thereto, must have authorization by the Station Superintendent before being implemented.

2. Work and instruction type procedures which implement approved maintenance or modification procedures shall be approved and authorized by the Maintenance Asst. Supt. where the written authority has been provided by the Station Superintendent. The "Maintenance/Modification Procedures" utilized for safety related work shall be so approved only if procedures referenced in the "Maintenance/Modification Procedure" have been approved as required by 6.2.A. Procedures which do not fall within the requirement of 6.2.A or 6.2.B. may be approved by the Department Heads.

D. Temporary changes to procedures 6.2.A. and 6.2.B. above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed by the Onsite Review and Investigative Function and approved by the Station Superintendent within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4. shall be conducted in accordance with the GSEP Manual.

2. A tabulation shall be submitted on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job function (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report. In addition, any changes to the ODCM shall be submitted with the Monthly Operating Report within 90 days of the effective date of the change.

A report of major change to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the onsite review function. If such change is re-evaluated and not installed, notification of cancellation of the change should be provided to the NRC.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. In general, the importance of an occurrence with respect to safety significance determines the immediacy of reporting required. In some cases, however, the significance of an event may not be obvious at the time of its occurrence. In such cases, the NRC shall be informed promptly of an increased significance in the licensee's assessment of the event. In addition, supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1. Prompt Notification with Written Followup

The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the director of the appropriate regional office or his designate no later than the first working day following the event, with a written followup report within 2 weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented as needed by additional narrative material to provide

Note: This item is intended to provide for reporting of potentially generic problems.

2. Thirty-Day Written Reports

The reportable occurrences discussed below have lesser immediate importance than those described under B.1. above. Such events shall be the subject of written reports to the director of the appropriate regional office within 30 days of occurrence of the event. The written report shall include, as a minimum: a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in Items B.2.a. and B.2.b. need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in Item B.1.c. above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set fort in technical specifications need not be reported under this item.

C. [illegible] Reporting Requirements

1. Radioactive Effluent Release Report (Semi-Annual)

[illegible] annual report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June, 1974. Any changes to the PCP shall be included in this report.

2. Environmental Program Data (Annual Report)

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-4) shall be submitted prior to May 1 of each year. The content of the report shall include:

a. Results of all environmental measurements summarized in the format of Regulatory Guide 4.8 Table 1 (December 1975). (Individual sample results will be retained at the Station). In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. Summaries, interpretations, and analysis of trends of the results are to be provided.

b. An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emmissions of radioactivity including the maximum noble gas gamma and beta air doses in the unrestricted area. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

c. Results of the census to determine the locations of nearest residences and of nearby animals producing milk for human consumption, and the pasture season feeding practices at dairies in the monitoring program (Table 4.8-4).

d. The reason for the emission if the nearest dairy to the station is not in the monitoring program (Table 4.8-4).

e. An annual summary of meteorological conditions concurrent with the releases of gaseous effluents in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

f. The results of the Interlaboratory Comparison Program described in section 3.8.D.7.

g. The results of the 40 CFR 190 uranium fuel cycle dose analysis for each calendar year.

h. A summary of the monitoring program, including maps showing sampling locations and tables giving distance and direction of sampling locations from the Station.

3. If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-5 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Director of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter.

   a. When more than one of the radionuclides in Table 4.8-5 are detected in the medium, the reporting level shall have been exceeded if

$$\sum \frac{Ci}{R.L.i} \geq 1$$

   where Ci is the average quarterly concentration of the i^{th} radionuclide in the medium and RL is the reporting level of radionuclide i.

   b. If radionuclides other than those in Table 4.8-5 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

   c. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

4. Special Reports

   Special Reports shall be submitted as indicated in Table 6.6-1.

TABLE 6.6-1
SPECIAL REPORTS

Area	Specification Reference	Submittal Date
a. Secondary containment leak rate test (1)	4.7.C	Upon completion of each test
b. Summary status of fuel performance	1.1 Bases	After each refueling outage.
c. Materials radiation surveillance specimens	4.6.B.2	After each specimen removal and completion of analyses
d. Evaluation of EGC operation	3.3.F Bases	Upon completion of initial testing.
e. Radioactive Source Leak Testing (2)	4.8.F	Annual Report
f. Special Effluents Reports	3.8.A. 3.8.B. 3.8.D. 6.6.C.3.	30 days following occurrence.

Notes

1. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

2. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.8 Offsite Dose Calculation Manual (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

B. Licensee initiated changes to the ODCM may be made provided the change:

1. Shall be submitted to the Commission by inclusion in the Monthly Operating Report pursuant to Specification 6.6.A.3. within 90 days of the date the change(s) was made effective and shall contain:

   a. Sufficiently detailed information to support the change. information submitted should consist of a package of those pages of the ODCM to be changed together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy of reliability of dose calculations or set-point determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the onsite review functions.

2. Shall become effective upon review and acceptance by the onsite review function.

6.9 Process Control Program (PCP)

A. The PCP shall contain the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

B. The PCP shall be approved by the Commission prior to implementation.

C. Licensee initiated changes may be made to the PCP provided the change:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change was made and and shall contain:

a. Sufficiently detailed information to support the change;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation that the change has been reviewed and found acceptable by the onsite review function.

2. Shall become effective upon review and acceptance by the onsite review function.

6.10 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, Solid)

A. Licensee initiated major changes to the radioactive waste systems may be made provided:

1. The change is reported in the Monthly Operating Report for the period in which the evaluation was reviewed by the onsite review function. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to support the reason for the change;

   c. A detailed description of the equipment, components, and process involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and (or quantity of solid waste that differ from those previously predicted in the license application and amendments);

   e. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

   f. An estimate of the exposure to plant operating personnel as a result of the change; and

   g. Documentation of the fact that the change was reviewed and found acceptable by the onsite review function.

2. The change shall become effective upon review and acceptance by by onsite review function.