ENCLOSURE IV

PROPOSED RADIOLOGICAL EFFLUENT

TECHNICAL SPECIFICATIONS

For

Cooper Nuclear Station

(Note: This cover sheet is not part of the proposed Technical Specifications)

Revised June 7, 1982

RADIOLOGICAL TECHNICAL SPECIFICATIONS

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Thermal Parameters

1. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

2. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

3. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio corresponding to the most limiting fuel assembly in the core.

4. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 bundles.

5. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

D. Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 104.4% of Rated Power (105% of rated steam flow) and the power to which the safety analysis applies.

E. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required to maintain the consequences of postulated accidents within acceptable limits.

E.A Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be 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. They thyroid dose conversion factors used for this calculation shall be those listed in Regulatory Guide 1.109.

E.B Exhaust Ventilation Treatment System - An EXHAUST VENTILATION TREATMENT SYSTEM (EVTS) is a system intended to remove radioiodine or radioactive material in particulate form from gaseous effluent by passing exhaust ventilation air through charcoal absorbers and/or HEPA filters before exhausting the air to the environment. An EVTS is not intended to affect noble gas in gaseous effluent. Engineered Safety Feature (ESF) gaseous treatment systems are not considered to be EVTS. The Standby Gas Treatment System is an ESF and not an EVTS. EVTS are specifically identified in ODAM Figure 3-1.

F. Functional Test - A functional test is the manual operation or initiation of a system, subsystem or component to verify that it functions within design tolerances (e.g. the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

F.A Gaseous Radwaste Treatment System - A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

G. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1000 psig, and the mode switch in "Startup/Hot Standby".

H. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

I. Instrumentation

1. Instrument Functional Test - Analog instrument functional test means the injection of a simulated signal into the instrument as close to the sensor as practical to verify the proper instrument channel response, alarm and/or initiating action. Bistable channels - the injection of a simulated signal into the sensor the verify OPERABILITY including alarm and/or trip functions.

2. Instrument Calibration - An instrument calibration means the adjustment, as necessary, of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including sensor, alarm/or trip functions and shall include the functional test. The calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

3. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit a signal related to the plant parameter monitored by that instrument channel.

4. Instrument Check - An instrument check is the qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same parameter.

5. Logic System Functional Test - A logic system functional test means a test of relays and contacts of a logic circuit from sensor to activated device to ensure components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Protective Action - An action initiated by the protection system when a limiting safety system setting is reached. A protective action can be at a channel or system level.

7. Protection Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

8. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

8.A Source Check - A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

9. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

J. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performances necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

Limiting Conditions for Operation (LCO) shall be applicable during the operational conditions specified for each specification.

Adherence to the requirements of the LCO within the specified time interval shall constitute compliance with the specification. In the event the LCO is restored prior to expiration of the specified time interval, completion of the LCO action is not required.

In the event an LCO cannot be satisfied because of circumstances in excess of those addressed in the specification, the facility shall be placed in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the LCO for the specified time interval as measured from initial discovery. Exception to these requirements shall be stated in the individual specifications.

Entry into an operational condition shall not be made unless the conditions of the LCO are met without reliance on the actions specified in the LCO unless otherwise excepted. This provision shall not prevent passage through operational conditions required to comply with the specified actions of an LCO.

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

K.A

L. Mode - The reactor mode is established by the mode selector-switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RSM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position.

4. Startup/Hot Standby - In this mode the reactor protection scram trips initiated by the main steam line isolation valve closure are bypassed when reactor pressure is less than 1000 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with APRM (15% SCRAM) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

L.A Normal Ventilation - Normal ventilation is the controlled process of discharging and replacing air from/to a confinement to maintain temperature, humidity, or other conditions necessary for personnel safety and entry. The contents of the atmosphere being discharged from the confinement will have been established prior to establishing normal ventilation following a purging/venting operation.

L.B Offsite Dose Assessment Manual (ODAM) - An OFFSITE DOSE ASSESSMENT MANUAL (ODAM) shall be a manual containing 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.

M. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

N. Operating - Operating means that a system or component is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all or the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P.A Process Control Program - The Process Control Program outlines the solidification of radioactive waste from liquid systems. It does not substitute for station operating procedures, but provides a general description of equipment, controls, and practices to be considered during waste solidification to assure solid wastes.

P.B Purge - Purging - Purge or Purging is the controlled process of discharging air or gas from a confinement to establish temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

P.C Radiological Environmental Monitoring Manual (REMM) - This manual describes the radiological environmental monitoring program for CNS.

Q. Rated Power - Rated power refers to operation at a reactor power of 2381 megawatts thermal. This is also termed 100% power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 105% of rated power, which corresponds to 2500 megawatts thermal.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated power.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

T. Refueling Outage - Refueling Outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling.

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" position.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F and the reactor vessel vented.

W.A Solidification - SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a solid which is as uniformally distributed as reasonably achievable with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

X. Surveillance Frequency - Surveillance requirements shall be applicable during the operational conditions associated with individual LCO's unless otherwise stated in an individual Surveillance Requirement.

Each Surveillance Requirement shall be performed within the specified time interval with:

1. A maximum allowable extension not to exceed 25% of the surveillance interval.

2. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified interval.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO unless otherwise required by the specification.

Y. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable or as practicable following its return to service.

Z.A Venting - Venting is the controlled process of discharging air or gas from a confinement to establish temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.2.C (Cont'd)	4.2.C
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Main Condenser Air Ejector OffGas System	1. Main Condenser Air Ejector Off-Gas System
a. Operability of the Main Condenser Air Ejector Off-Gas System is defined in Table 3.21.A.2.	Instrumentation surveillance requirements are given on Table 4.21.A.2.
b. The time delay setting for closure of the steam jet air ejector isolation valves shall not exceed 15 minutes.	
c. Other limiting conditions for operation are given on Table 3.2.D and Specifications 3.21.A.2 and 3.21.C.6.	
2. Reactor Building Isolation and Standby Gas Treatment Initiation	2. Reactor Building Isolation and Standby Gas Treatment Initiation
The limiting conditions for operation are given on Table 3.2.D and Specification 3.21.A.2.	Instrumentation surveillance requirements are given on Table 4.2.D.
3. Liquid Radwaste Discharge Isolation	3. Liquid Radwaste Discharge Isolation
The limiting conditions for operation are given on Table 3.2.D and Specification 3.21.B.	Instrumentation surveillance requirements are given on Table 4.2.D and Table 4.21.A.1.
4. Main Control Room Ventilation Isolation	4. Main Control Room Ventilation Isolation
The limiting conditions for operation are given on Table 3.2.D and the Section entitled "Additional Safety Related Plant Capabilities."	The instrument surveillance requirements are given on Table 4.2.D.

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TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I.D. No.	Setting Limit	No. of Sensor Channels Provided by Design	Action (1)
Main Condenser Air Ejector Off-Gas System	RMP-RM-150 A & B	(3)	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	$\leq$ 100 mr/hr	2	B
Liquid Radwaste Discharge Isolation	RMP-RM-2	(2)	1	C
Main Control Room Ventilation Isolation	(RMV-RM-1)	4×10^3 CPM	1	D
Mechanical Vacuum Pump Isolation	RMP-RM-251 A-D	3 times normal full power background. Alarm at 1.5 times normal full power background.	4	E

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

   A. (1) If radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for a period greater than 15 consecutive minutes, the off-gas isolation valve shall close and reactor shutdown shall be initiated immediately and the reactor placed in a cold shutdown condition within 24 hours.

   A. (2) Refer to Specification 3.21.A.2.

   B. Cease refueling operations, isolate secondary containment and start SBGT.

   C. During release of radioactive wastes, the effluent control monitor shall be set to alarm and automatically close the waste discharge valve prior to exceeding the limits of Specification 3.21.B.1.

   D. Refer to Section entitled "Additional Safety Related Plant Capabilities".

   E. Refer to Section 3.2.d.5 and the requirements for Primary Containment Isolation on high main steam line radiation. Table 3.2.A.

2. Trip settings to correspond to Specification 3.21.B.1.

3. Trip settings to correspond to Specification 3.21.C.6.a.

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TABLE 4.2.D

MINIMUM TEST AND CALIBRATION FREQUENCIES FOR RADIATION MONITORING SYSTEMS

System	Instrument I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	(12)	(12)	(12)
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	(12)	(12)	(12)
Liquid Radwaste Discharge Isolation	(RMP-RM-2)	(11)	(11)	(11)
Main Control Room Ventilation Isolation	RMV-RM-1	Once/Month (1)	Once/3 Months	Once/Day
Mechanical Vacuum Pump Isolation	RMP-RM-251, A-D		See Tables 4.1.1 & 4.1.2	
Logic Systems				
SJAE Off-Gas Isolation		Once/Year		
Standby Gas Treatment Initiation		Once/6 Months		
Reactor Building Isolation		Once/6 Months		
Liquid Radwaste Disch. Isolation		Once/6 Months		
Main Control Room Vent Isolation		Once/6 Months		
Mechanical Vacuum Pump Isolation		Once/Operating Cycle		

NOTES FOR TABLES 4.2.A THROUGH 4.2.F

1. Initially once every month until exposure (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 (after NRC approval). 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 CNS.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of applying simulated inputs. Local alarm lights representing upscale and downscale trips will be verified but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2.A since they are tested on Table 4.1.2.

6. The logic system functional tests shall include an actuation of time delay relays and timers necessary for proper functioning of the trip systems.

7. These units are tested as part of the Core Spray System tests.

8. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verifying that it will produce a rod block during the operating cycle.

9. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

10. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

11. Surveillance requirements for this system are defined in Table 4.21.A.1.

12. Surveillance requirements for this system are defined in Table 4.21.A.2.

3.2 BASES (Cont'd)

Trip settings of <100 mr/hr for the monitors in the ventilation exhaust ducts 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 path but rather all the activity is processed by the standby gas treatment system.

Flow transmitters are used to record the flow of liquid from the drywell sumps. An air sampling system is also provided to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

The liquid radwaste monitor assures that all liquid discharged to the discharge canal does not exceed the limits of Specification 3.21.B of Environmental Technical Specifications. Upon sensing a high discharge level, an isolation signal is generated which closes of radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

The main control room ventilation isolation is provided by a detector monitoring the intake of the control room ventilation system. Automatic isolation of the normal supply and exhaust and the activation of the emergency filter system is provided by the radiation detector trip function at the predetermined trip level.

The mechanical vacuum pump isolation prevents the exhausting of radioactive gas thru the 1 minute holdup line upon receipt of a main steam line high radiation signal.

The operability of the reactor water level instrumentation in Tables 3/4.2.F ensures that sufficient information is available to monitor and assess accident situations.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS

A. Instrumentation

1. Liquid Effluent Monitoring

Applicability: As shown in Table 3.21.A.1.

Specification:

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.21.A.1 shall be OPERABLE with their alarm and trip setpoints set to insure that the limits of 3.21.B.1 are not exceeded.

b. With a radioactive liquid effluent monitoring instrumentation channel alarm and trip setpoint less conservative than required, reset without delay to meet Specification 3.21.A.1.a, suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

c. Radioactive liquid effluent monitoring instrumentation, with less than the minimum number of channels operable, take the ACTION shown in Table 3.21.A.1.

d. If the minimum number of instrument channels is not returned to OPERABLE status within 31 days, in lieu of any other report, explain in the next Semiannual Radioactive Effluent Report why the instrument was not repaired in a timely manner.

e. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.7.2.B.2 are not applicable.

4.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS

A. Instrumentation

1. Liquid Effluent Monitoring

a. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.21.A.1.

b. Radioactive liquid effluent monitor alarm and trip setpoints shall be determined in the manner described in the ODAM. Auditable records of the setpoints and setpoint calculations shall be maintained.

TABLE 3.21.A.1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY+	ACTION
1. Gross Beta or Gamma Radioactivity Monitor Providing Automatic Isolation			
a. Liquid Radwaste Effluent Line	1	*	18
2. Gross Beta or Gamma Radioactivity Monitors Not Providing Automatic Isolation			
a. Service Water Effluent Line	1	*	20
3. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	*	21

NOTES FOR TABLE 3.21.A.1

* During releases via this pathway.

\+ Channel(s) shall be OPERABLE and in service except that outages for maintenance and required tests, checks, or calibrations are permitted.

ACTION 18 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be resumed, providing that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specification 4.21.B.1.c and;

2. At least one technically qualified member of the Facility Staff independently verifies the release rate calculations and discharge valving which were determined by another qualified member.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 20 With the numbers of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided that at least once every day a grab sample is collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection not greater than 10^{-6} μCi/ml.

ACTION 21 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided the flow rate is estimated at least once per 4 hours during actual releases.

TABLE 4.21.A.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Liquid Radwaste Effluents Line	D*	P	R(3)	Q(1)
2. Gross Beta or Gamma Radioactivity Monitors Provinding Alarm but not Providing Automatic Isolation				
a. Service Water System Effluent Line	D*	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(4)*	NA	R	SA
4. Tank Level Monitor				
a. Condensate Storage	D**	NA	R	Q

NOTES FOR TABLE 4.21.A.1

* During releases via this pathway.
** During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs for Conditions 1 and 2 below and control room alarm annunciation occurs for Conditions 1, 2, and 3 below.

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) The CHANNEL CALIBRATION shall include the use of a known (traceable to the National Bureau of Standards radiation measurement system) radioactive source positioned in a reproducible geometry with respect to the sensor and emitting beta and gamma radiation in the range measured by the channel according to established station calibration procedures.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

FREQUENCY NOTATION:

S	=	At least once per 12 hours.
D	=	At least once per 24 hours.
W	=	At least once per 7 days.
M	=	At least once per 31 days.
Q	=	At least once per 92 days.
SA	=	At least once per 184 days.
A	=	At least once per year.
R	=	At least once per 18 months.
S/U	=	Prior to each reactor startup.
P	=	Completed prior to each release.
NA	=	Not applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.A (Cont'd)	4.21.A (Cont'd)
2. Gaseous Effluent Monitoring	2. Gaseous Effluent Monitoring
Applicability: As shown in Table 3.21.A.2.	a. The setpoints shall be determined in accordance with the method described in the ODAM.
Specification:	b. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.21.A.2.
a. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.21.A.2 shall be OPERABLE with their alarm setpoints set to ensure that the limits of Specification 3.21.C.1 are not exceeded.	c. Auditable records of the setpoints and setpoint calculations shall be maintained.
b. With a radioactive gaseous effluent monitoring instrumentation channel alarm setpoint less conservation than a value which will ensure that the limits of 3.21.C.1 are met, reset without delay to comply with Specification 3.21.A.2.a, declare the channel inoperable; or immediately suspend release.	
c. With less than the minimum required number of radioactive gaseous effluent monitoring instrumentation channels operable, take the ACTION shown in Table 3.21.A.2.	
d. If the minimum number of instrument channels are not returned to OPERABLE status within 31 days, in lieu of any other report, explain in the next Semiannual Radioactive Effluent Report why the instrument was not repaired in a timely manner.	
e. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.7.2.B.2 are not applicable.	

TABLE 3.21.A.2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY#	PARAMETER	ACTION
1. Main Condenser Air Ejector				
a. Noble Gas Activity Monitor	1	***	Noble Gas Radioactivity Rate Measurement	25
b. Effluent System Flow Rate Measuring Device	1	*	System Flow Rate Measurement	26
2. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor (2% monitor)	2	**	% Hydrogen	28
3. Reactor Building Ventilation Monitor System				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	27
b. Iodine Sampler Cartridge	1	*	Verify Presence of Cartidge	29
c. Particulate Sampler Filter	1	*	Verify Presence of Filter	29
d. Effluent System Flow Rate Measuring Device	1	*	System Flow Rate Measurement	26
e. Sampler Flow Rate Measurement Device	1	*	Sampler Flow Rate Measurement	26
4. (****)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	27
b. Iodine Sampler Cartridge	1	*	Verify Presence of Cartridge	29
c. Particulate Sampler Filter	1	*	Verify Presence of Filter	29
d. Effluent System Flow Rate Measuring Device	1	*	System Flow Rate Measurement	26
e. Sampler Flow Rate Measuring Device	1	*	Sampler Flow Rate Measurement	26

NOTES FOR TABLE 3.21.A.2

Channels shall be operable and in service except that outages are permitted for the purpose of required tests, checks, and calibrations.
* During releases via this pathway.
** During Augmented Offgas Treatment System Operation.
*** During operation of the Main Condenser Air Ejector.
**** Main Stack Monitoring System, Augmented Radwaste Building Ventilation Monitoring System, Radwaste Area (Building) Ventilation Monitoring System (b, c, and e only), Turbine Building Ventilation Monitoring System.

ACTION 25 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, gases from the main condenser offgas treatment system may be released to the environment for up to 72 hours provided:

1. The offgas delay system is not bypassed; and

2. The main stack system noble gas activity monitor is OPERABLE:

Otherwise, be in at least HOT STANDBY within 12 hours.

ACTION 26 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 24 hours. 24 hour estimates are adequate since the system design is constant flow and loss of flow is alarmed in the Control Room.

ACTION 27 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per day and these samples are analyzed for gross activity within 24 hours.

ACTION 28 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the augmented offgas treatment system may continue with one channel operable provided that the recombiner exhaust temperature is monitored. With only one of the preceeding methods operable, operation of the augmented offgas treatment system may continue provided gas samples are collected at least once per day and analyzed within the ensuing 4 hours.

ACTION 29 With the number of samplers OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.21.C.1.

TABLE 4.21.A.2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Main Condenser Air Ejector				
a. Noble Gas Activity Monitor	D***	M	R(3)	Q(2) R(1)
b. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
2. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor (2% Monitor)	D**	NA	Q(4)	M
3. Reactor Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor (NMC Monitor)	D*	M	R(3)	Q(2)
b. Iodine Sampler Cartridge	W*	NA	NA	NA
c. Particulate Sampler Filter	W*	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
e. Sampler Flow Rate Measuring Device	D*	NA	R	Q
f. Isolation Monitor (GE)	D*	Q	R(3)	R(1)
4. (****)				
a. Noble Gas Activity Monitor	D*	M	R(3)	Q(2)
b. Iodine Sampler	W*	NA	NA	NA
c. Particulate Sampler	W*	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
e. Sampler Flow Rate Monitor	D*	NA	R	Q

NOTES FOR TABLE 4.21.A.2

* During releases via this pathway.
** During augmented offgas treatment system operation.
*** During operation of the main condenser air ejector.
**** Main Stack Monitoring System, Augmented Radwaste Ventilation Monitoring System, Radwaste Area (Building) Ventilation Monitoring System (b, c, and e only), Turbine Building Ventilation Monitoring System

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The CHANNEL CALIBRATION shall include the use of known (traceable to the National Bureau of Standards radiation measurement system) radioactive source positioned in a reproducible geometry with respect to the sensor and emitting beta and/or gamma radiation in the range measured by the channel in accordance with established station calibration procedures.

(4) The CHANNEL CALIBRATION shall include the use of a standard gas sample containing a percentage of hydrogen to verify accuracy of the monitoring channel in its operating range.

FREQUENCY NOTATION:

S	=	At least once per 12 hours.
D	=	At least once per 24 hours.
W	=	At least once per 7 days.
M	=	At least once per 31 days.
Q	=	At least once per 92 days.
SA	=	At least once per 184 days.
A	=	At least once per year.
R	=	At least once per 18 months.
S/U	=	Prior to each reactor startup.
P	=	Completed prior to each release.
NA	=	Not applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
B. Liquid Effluents	B. Liquid Effluents
Applicability: At all times.	1. Concentration
Specification:	a. Radioactive liquid wastes shall be sampled and analyzed according to Table 4.21.B.1.
1. Concentration	b. The analytical results shall be used with methods in the ODCM to verify that the average concentration at the restricted area boundary does not exceed Specification 3.21.B.1.2.
a. The concentration of radioactive material released from the site to the unrestricted area (Figure 4.20.B.2) shall not exceed the concentration specified in 10 CFR Part 20.106 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} µCi/ml total activity.	
b. With the concentration of radioactive material released from the site to the unrestricted area exceeding the limit, restore the concentration within the above limit and provide prompt notification to the Commission pursuant to Specification 6.7.2.A.	

TABLE 4.21.B.1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)(1)
1. Batch Waste Release Tanks(5)	P Each Batch	P Each Batch	Principal Gamma Emitters(7)(8)	5×10^{-7}(2)
			I-131	1×10^{-6}
	P One Batch/M	M(9)	Dissolved and Entrained Gases	1×10^{-5}
	P Each Batch	M Composite(3)(9)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q(9) Composite(3)(9)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2.A. Plant Service Water Effluent(6)	W Grab Sample	W(9)	Principal Gamma Emitters(7)(8)	5×10^{-7}(2)
2.B. Plant Continuous Discharge(10)	Proportional(4)	W(9) Composite(4)	Principal Gamma Emitters(7)(8)	5×10^{-7}(2)
			I-131	1×10^{-6}
	M Grab Sample	M(9)	Dissolved and Entrained Gases	1×10^{-5}
	Proportional(4)	M(9) Composite(4)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Proportional(4)	Q(9) Composite(4)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

NOTES FOR TABLE 4.21.B.1

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

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

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

Where:

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

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

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

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

(2) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportionally to the magnitude of the gamma yield (i.e., $5 \times 10^{-7}/I$, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the MPC value specified in 10 CFR 20, Appendix B, Table II, Column 2.

(3) A composite sample is one in which the quantity of liquid sampled 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.

(4) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

NOTES FOR TABLE 4.21.B.1 (Continued)

(5) 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.

(6) A grab sample of plant service water effluent shall be analyzed at least once each week in accordance with Table Item 2.A. In the event the radioactivity concentration in a sample exceeds 3×10^{-6} μCi/ml, or in the event the plant service water effluent monitor indicates the presence of an activity concentration greater than 3×10^{-6} μCi/ml, sampling and analysis according to Table Item 2.B. shall commence and shall be performed as long as the condition persists.

(7) The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semiannual Radioactive Effluent Release Report.

(8) If an isotopic analysis is unavailable, batch releases may be made for up to 14 days provided the gross beta/gamma concentration to the unrestricted area is $\leq 1 \times 10^{-7}$ μc/ml and the sample is analyzed when the instrumentation is once again available.

(9) Analysis may be performed after release.

(10) A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

FREQUENCY NOTATION:

S	= At least once per 12 hours.	A	= At least once per year.
D	= At least once per 24 hours.	R	= At least once per 18 months.
W	= At least once per 7 days.	S/U	= Prior to each reactor startup.
M	= At least once per 31 days.	P	= Completed prior to each release.
Q	= At least once per 92 days.	NA	= Not applicable.
SA	= At least once per 184 days.		

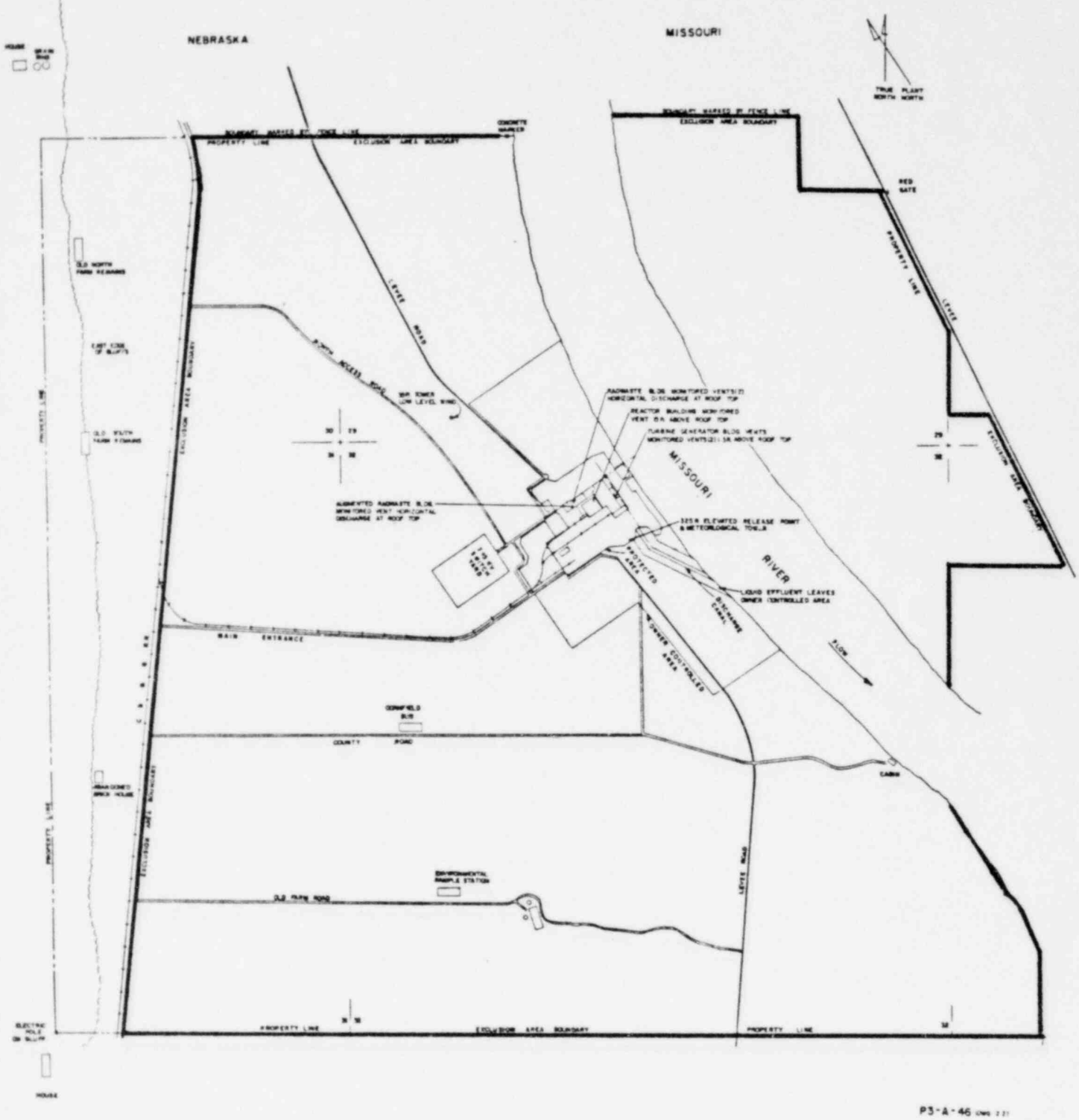

Figure 4.21.B.2

Exclusion Area Boundary
For
Gaseous and Liquid Effluents

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.B (Cont'd)	4.21.B (Cont'd)
2. Liquid Dose	2. Liquid Dose
a. The dose or dose commitment to an individual from radioactive materials in liquid effluents released to unrestricted areas (see Figure 4.21.B.2) shall not exceed 1.5 mrem to the total body or 5 mrem to any organ during any calendar quarter.	a. Dose Calculation - Cumulative dose contributions from liquid effluents shall be determined in accordance with the Offsite Dose Assessment Manual (ODAM) at least once per 31 days.
b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding the above limit, prepare and submit to the Commission within 31 days after the end of the quarter in which the limit was exceeded, pursuant to Specification 6.7.3.B, a Special Report in lieu of any other report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken.	b. In any quarter in which radioactive liquid releases are made and the radwaste system is not operable, doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days in accordance with the ODAM.
c. Appropriate parts of the system shall be used to reduce the concentration of radioactive materials in liquid wastes prior to their discharge when the pre-release analysis indicates a radioactivity concentration, excluding tritium and noble gases, in excess of 0.01 uCi/ml.	
d. With radioactive liquid waste being discharged without treatment in excess of the above limit, prepare and submit to the Commission within 31 days after the end of the quarter in which the limit was exceeded, pursuant to Specification 6.7.3.B, a Special Report in lieu of any other report which includes the following information: 1) Indentification of equipment or subsystems not OPERABLE and the reason for nonoperability. 2) Action(s) taken to restore the nonoperable equipment to OPERABLE status. 3) Summary description of action(s) taken to prevent a recurrence.	
e. The provisions of Definition J are not applicable.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.B (Cont'd)	4.21.B (Cont'd)
3. Condensate Storage Tank and Outside Temporary Tanks	3. Condensate Storage Tank and Outside Temporary Tanks
This specification deleted.	This specification deleted.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
C. Gaseous Effluents	C. Gaseous Effluents
Applicability: At all times.	
Specification:	
1. Concentration	1. Concentration
a. The concentration of radioactive noble gas in air offsite due to gaseous effluents shall not exceed the concentration specified in 10 CFR Part 20.106.	a. The release rate of radioactive noble gas shall be monitored according to Specification 3.21.A.2.
b. With the concentration exceeding the limit in 3.21.C.1.a, decrease the release rate to comply with the limit and provide prompt notification to the Commission pursuant to Specification 6.7.2.A.	b. A radioactive noble gas effluent monitor shall be set to cause automatic alarm when the concentration exceeds the monitor alarm setpoint, determined as specified in the ODAM.
c. The provisions of Definition J are applicable.	
2. Noble Gases Dose	2. Noble Gases Dose
a. The air dose in unrestricted areas (see Figure 4.21.B.2) due to noble gases released in gaseous effluents shall not exceed 5 mrad from gamma radiation and 10 mrad from beta radiation during any calendar quarter.	a. Dose Calculations - Cumulative dose contributions during each calendar quarter shall be determined in accordance with the method in the ODAM at least once every 31 days.
b. 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 31 days, pursuant to Specification 6.7.3.B, a Special Report (in lieu of any other report) which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken.	
c. The provisions of Definition J are not applicable.	

TABLE 4.21.C.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) (μCi/ml)(1)
A.	Main Stack, Reactor Bldg Vent, Augmented Radwaste Bldg Vent, Turbine Bldg Vent (Gaseous)	M(3) Grab Sample	M(3)	Principal Gamma Emitters(6)	1×10^{-4}(2)
		Q(9) Grab Sample	Q	H-3	1×10^{-6}
B.	All Release Types as Listed in A Above, & Radwaste Bldg Vent (Iodine & Particulate)	Continuous(5)	W(4) Charcoal Sample	I131 I-133	1×10^{-12} 1×10^{-10}
		Continuous(5)	W(4) Particulate Sample	Principal Gamma Emitters(6) (I-131, Others)	1×10^{-11}(2)
		Continuous(5)	Q Composite Particulate Sample(7)	Sr89, Sr-90 Gross Alpha	1×10^{-11} 1×10^{-11}
		Continuous(5)	Noble Gas Monitor	Gross Noble Gases Beta and Gamma(8)	5×10^{-6}

NOTES FOR TABLE 4.21.C.1

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

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

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

Where:

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

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

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

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

(2) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportionally to the magnitude of the gamma yield (i.e., $1 \times 10^{-4}/I$, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the MPC value specified in 10 CFR 20, Appendix B, Table II, Column 1.

(3) Analyses shall also be performed following an increase as indicated by the gaseous release monitor of greater than 50% in the steady state release, after factoring out increases due to power changes or other operational occurrences, which could alter the mixture of radionuclides.

(4) Analyses shall also be performed following an increase as indicated by the gaseous release monitor of greater than 50% in the steady state release, after factoring out increases due to power changes or other operational occurrences, which could alter the mixture of radionuclides. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(5) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.21.C.1, 3.21.C.2 and 3.21.C.3.

(6) The principal gamma emitters for which the LLD specification will apply are exclusively 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-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances cause LLD's higher than required for more than 31 days, the reasons shall be documented in the semi-annual effluent report.

(7) A quarterly composite particulate sample shall include a portion of each weeks particulate samples collected during the quarter.

(8) The noble gas continuous monitor shall be calibrated using laboratory analysis of the grab samples from A and B on Table 4.21.C.1 or using reference sources.

(9) A H-3 grab sample will also be taken when the reactor vessel head is removed. This sample will be taken at the ERP or Reactor Building vent whichever will be representative dependent upon the head removal vaccum procedure.

FREQUENCY NOTATION:

S	=	At least once per 12 hours.
D	=	At least once per 24 hours.
W	=	At least once per 7 days.
M	=	At least once per 31 days.
Q	=	At least once per 92 days.
SA	=	At least once per 184 days.
A	=	At least once per year.
R	=	At least once per 18 months.
S/U	=	Prior to each reactor startup.
P	=	Completed prior to each release.
NA	=	Not applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C (Cont'd)
3. Iodine and Particulate	3. Iodine and Particulate
a. The dose to an individual from I-131 Tritium (H-3), and radioactive material in particulate form having a half-life greater than 8 days in gaseous effluents released to unrestricted areas (see Figure 4.21.B.2) shall not exceed 7.5 mrem to any organ during any calendar quarter.	a. Dose Calculations - Cumulative dose contributions during each quarter shall be determined in accordance with the ODAM at least once every 31 days.
b. With the calculated dose from the release of radionuclides, radioactive materials in particulate form, or radionuclides other than noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 31 days following the end of the calendar quarter in which the release occurred pursuant to Specification 6.7.3.B, in lieu of any other report, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken.	
c. The provisions of Definition J are not applicable.	
4. Gaseous Releases	4. Gaseous Releases
a. Every reasonable effort shall be made to maintain at least one train of the gaseous radwaste treatment system and the exhaust ventilation treatment system OPERABLE.	a. In any month in which radioactive material in gaseous effluent is being released without treatment, doses due to gaseous releases to unrestricted areas shall be projected at least once per 31 days using calculational methods in the ODAM.
b. The gaseous radwaste treatment system shall be operated to reduce radioactive materials in gaseous wastes prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases via the ERP to unrestricted areas (see Figure 4.21.B.2) when averaged over 31 days would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation.	
c. The ventilation exhaust treatment system shall be operated to reduce the radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent doses due to gaseous effluent releases via vent exhaust to unrestricted areas (see Figure 4.21.B.2) when averaged over 31 days would exceed 0.3 mrem to any organ.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C (Cont'd)
d. With gaseous wastes being discharged for more than 31 days without treatment and in excess of the above limits, prepare and submit to the Commission within 31 days, pursuant to Specification 6.7.3.B, in lieu of any other report, a Special Report which includes the following information: 1) Identification of equipment of subsystems not OPERABLE and the reason for nonoperability. 2) Action(s) taken to restore the non-operable equipment to OPERABLE STATUS. 3) Summary description of action(s) taken to prevent a recurrence. e. The provisions of Definition J are not applicable.	
5. Hydrogen Concentration	5. Hydrogen Concentration
a. The concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners shall be limited to ≤ 2% by volume. b. With the concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners exceeding the limit, restore the concentration to within the limit within 48 hours. c. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.7.2.B.2 are not applicable.	a. The concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners shall be determined by continuously monitoring the waste gases in the main condenser offgas treatment system with the (hydrogen) monitors required OPERABLE by Table 3.21.A.2.
6. Air Ejector	6. Air Ejector
a. The gross radioactivity (beta and/or gamma) rate of noble gases measured at the main condenser air ejector shall be limited to ≤ 1 Ci/sec at the air ejector. b. With the gross radioactivity (beta and/or gamma) rate of noble gases at the main condenser air ejector exceeding Specification 3.21.C.6.a, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.	a. The gross radioactivity (beta and/or gamma) rate of noble gases from the main condenser air ejector shall be determined at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the main condenser air ejector:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C.6 (Cont'd)
7. Containment a. Whenever the primary containment is vented/purged, it shall be vented/purged through the Standby Gas Treatment System. With this specification not satisfied, suspend all venting/purging of the containment. This specification does not apply to Normal Ventilation. b. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.7.2.B.2 are not applicable.	1) At least once per 31 days during normal operation. 2) Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant. b. The radioactivity rate of noble gases at or near the outlet of the main condenser air ejector shall be continuously monitored in accordance with Table 3.21.A.2.
D. Effluent Dose Liquid/Gaseous Applicability: At all times. Specification: 1. The dose or dose commitment to a (actual) member of the public due to radiation and radioactive releases from Cooper Station shall not exceed 25 mrem to the total body or any organ except the thyroid and shall not exceed 75 mrem to the thyroid during a calendar year. In the event the calculated dose from radioactive material in liquid or gaseous effluents exceeds two times the limit of Specification 3.21.B.2, 3.21.C.2, or 3.21.C.3, prepare and submit a Special Report, in lieu of any other report, to the Commission pursuant to Specification 6.7.3.B within 31 days which 1) defines actions to be taken to reduce releases and prevent recurrence and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a member of the public due to radiation and radioactive releases from Cooper Station during the calendar year through the period covered by the calculation was less than limits stated in this Specification. If the estimated dose exceeds the limits stated herein, and if the condition resulting in doses exceeding these limits has not already been corrected, submission of the Special Report shall be deemed a timely request for a variance in accord with provisions of 40 CFR Part 190, provided information specified in 40 CFR Part 190.11(b) is included. In that event, a variance is granted until NRC Staff action on the item is complete.	D. Effluent Dose Liquid/Gaseous 1. Dose Calculations - The cumulative dose to an individual contributed by radioactive material in gaseous and liquid effluents shall be calculated at least once per year in accordance with the ODAM in order to verify compliance with Specification 3.21.D.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.D (Cont'd)	4.21 (Cont'd)
2. The provisions of Definition J are not applicable.	
E. Solid Radioactive Waste	E. Solid Radioactive Waste
Applicability: During solid radwaste processing.	
Specification:	
1. The appropriate equipment of the solid radwaste system shall be operated in accordance with the Process Control Program to solidify and package radioactive waste and meet the requirements of 10 CFR Part 20 and 10 CFR Part 71 prior to shipment of radioactive wastes from the site.	1. Operating parameters and limits for the solidification of radioactive waste were established during preparational testing of the system. Radioactive waste solidification shall be performed in accordance with established parameters and limits and in accordance with the Process Control Program. In addition, every 10th batch of dewatered waste will be sampled prior to solidification and analyzed for pH.
2. When a package does not comply with 10 CFR Part 20 or 10 CFR Part 71, suspend shipment of the defective packaged waste.	2. Each drum of solidified radioactive waste will be visually inspected, prior to capping, to insure that there is no free standing liquid on top of the solidified waste.
	3. The Semiannual Radioactive Material Release Report in Specification 6.7.1.F shall include the following information for each type of solid waste shipped offsite during the report period:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
3. The provisions of Definition J are not applicable.	a. Container burial volume, b. Total curie quantity (determined by measurement or estimate), c. Principal gamma radionuclides (determined by measurement or estimate), d. Type of waste, e. Type of container, f. Solidification agent.
F. Monitoring Program	F. Monitoring Program
Applicability: At all times.	
Specification:	
1. As a minimum the radiological environmental monitoring program shall be conducted as specified in Table 3.21.F.1. Analytical techniques used shall be such that the detection capabilities in Table 3.21.F.2 are achieved.	1. Radiological environmental samples shall be collected and analyzed as specified in Table 3.21.F.1.
2. In the event the radiological environmental monitoring program is not conducted as specified in Table 3.21.F.1, prepare and submit to the Commission in the Annual Operating Report the reasons for not conducting the program in accordance with Table 3.21.F.1 and the plans for preventing a recurrence.	2. A land use census shall be conducted annually and shall identify the location of the nearest garden that is greater than 500 square feet in area and that yields edible leafy vegetables, the location of the nearest milk animal, and the location of the nearest resident in each of the 16 meteorological sectors within three miles of the Station. The land use census shall be conducted at least once per 12 months.
3. When the radioactivity in a sampled environmental medium, averaged over a calendar quarter, exceeds an appropriate value stated in Table 6.7-2, prepare and submit to the Commission within 31 days from the end of the affected calendar quarter a Special Report in accordance with 6.7.3.B which includes an evaluation of any release conditions, environmental factors or other conditions which caused the value(s) of Table 6.7-2 to be exceeded. If the radioactivity in environmental sample(s) is not attributable to release from the	3. The results of sample analyses performed shall be summarized in the Annual Radiological Environmental Report.
	4. The results of the land use census shall be included in the Annual Radiological Environmental Report.

3.21.F (Cont'd)

Station, the Special Report is not required; instead the sample(s) result(s) shall be reported and explained in the Annual Radiological Environmental Report.

4. When environmental sampling medium is not available from a sampling location designated in Table 3.21.F.1, the cause and the location where replacement samples were obtained shall be reported in the Annual Radiological Environmental Report.

5. In the event a location is identified at which the calculated personal dose associated with one or more exposure pathways exceed 120% of the calculated dose at the maximum dose location associated with like pathways at a location where sampling is conducted as specified in Table 3.21.F.1, then the pathways having maximum exposure potential at the newly identified location will be added to the radiological monitoring program and to Table 3.21.F.1 at the next SRAB meeting if samples are reasonably attainable at the new location. Like pathways monitored (sampled) at a location, excluding the control station location(s), having the lowest associated calculated personal dose may be deleted from Table 3.21.F.1 at the time the new pathway(s) and location are added.

6. A change in Table 3.21.F.1 shall be described in the Annual Radiological Environmental Report.

7. The provisions of Definition J are not applicable.

4.21 (Cont'd)

TABLE 3.21.F.1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Stations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. Airborne			
a. Radioiodine and Particulate	At least 5 locations in accordance with the Radiological Environmental Monitoring Manual (REMM).	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioidine canister: Analyze at least once per 7 days for I-131. Particulate sample: Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample in which gross beta activity is >10 times the yearly mean of control samples. Perform gamma isotopic[b] analysis on composite (by location) sample at least once per 92 days.
2. Direct Radiation	At least 32 locations in accordance with the REMM, with 2 dosimeters at each location.	Thermoluminescent Dosimeters (TLD) exchange and read-out at least once per 92 days.	Gamma dose: At least once per 92 days.
3. Waterborne			
a. River Water	At least 2 locations in accordance with the REMM.	Collect a one (1) gallon grab sample at least once per 31 days.	Gamma isotopic[b] analysis of each sample. Composite grab sample for tritium analysis at least once per 92 days.
b. Ground Water	At least 2 locations in accordance with the REMM.	Collect a one (1) gallon grab sample at least once per 92 days.	Gamma isotopic[b] and tritium analysis of each sample.
c. Sediment from Shoreline	At least 1 location in accordance with the REMM.	Two (2) times a year, once in the spring and once in the fall.	Gamma isotopic[b] analysis of each sample.

TABLE 3.21.F.1 (CONTINUED)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Stations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. Ingestion			
a. Milk	At least 4 locations in accordance with the REMM.	At least once per 15 days during Peak Pasture Period[c]; at least once per 31 days at other times.	Gamma isotopic[b] and I-131 analysis of each sample.
b. Fish	At least 2 locations in accordance with the REMM.	Two times per year (once in the summer and once in the fall). Attempt to include the following: 1. Bottom feeding species 2. Middle-Top feeding species	Gamma isotopic[b] analysis on edible portions.
c. Food Products (Vegetables)	At least 3 locations in accordance with the REMM.	At time of harvest. Sample one of the following classes of food products at each location. 1. Flowers & fruits[d] 2. Tubers[d] 3. Roots[d]	Gamma isotopic[b] analysis on edible portion.
	At least 1 location in accordance with the REMM.	At time of harvest. One sample of broad-leaf[d] vegetation.	I-131 analysis.

NOTES FOR TABLE 3.21.F.1

a. Sample station locations are shown on Figure 1.F.1 of the Radiological Environmental Monitoring Manual (REMM) maintained by the Environmental Affairs Division of the Power Operations Group.

b. Ge(Li) gamma isotopic analysis refers to high resolution Ge(Li) gamma spectrum analysis as follows: the sample is scanned for gamma-ray activity. If no activity is found for a selected nuclide, the detection sensitivity for that nuclide will be calculated using the counting time, detector efficiency, gamma energy, geometry, and detector background appropriate to the particular sample in question. The following nineteen (19) nuclides shall be analyzed for and routinely reported:

Be-7	Ru-103	Ce-144
K-40	Ru-106	Ra-226
Mn-54	I-131	Th-228
Fe-59	Cs-134	
Co-58	Cs-137	
Co-60	BaLa-140	
Zn-65	Ce-141	
Zr-95		
Nb-95		

Any nuclide detected, having a concentration greater than the LLD shall be reported quantitatively whether or not it is one of the above 19 nuclides.

c. Peak Pasture Period is June 1 through September 30 of each year.

d. Vegetables are classified as follows:

- Flowers and fruits: Artichoke, broccoli, cauliflower, corn, cucumber; eggplant, okra, pepper, pumpkin, squash, and tomato.

- Tubers: Potato.

- Roots: Beet, carrot, parsnip, radish, rutabaga, sweet potato, and turnip.

- Leaves (broad leaf): Cabbage, lettuce, spinach.

TABLE 3.21.F.2
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

	Lower Limit of Detection (LLD)[a]					
Anaysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}				
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	1[b]	7×10^{-2}		1	60	
^{134}Cs	15	5×10^{-2}	130	15	60	150
^{137}Cs	18	6×10^{-2}	150	18	80	180
^{140}Ba	60			60		
^{140}La	15			15		

Note: This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

NOTES FOR TABLE 3.21.F.2

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

For a particular measurement system (which may include radio-chemical separation):

$$LLD = \frac{4.66\ s^b}{E\ .\ V\ .\ 2.22\ .\ Y\ .\ \exp(-\lambda\Delta t)}$$

Where

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

s^b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per transformation)

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

2.22 is the number of transformation per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

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

The value of s^b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radio-nuclides normally present in the samples (e.g., potassium-40 in milk samples).

Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLD's unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

b. LLD for drinking water.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
G. Interlaboratory Comparison Program	G. Interlaboratory Comparison Program
Applicability: Applicable at all times to Radiological Environmental Monitoring Program.	1. A brief summary of results obtained as part of the Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Report, pursuant to Specification 6.7.1.A.E.
Specification:	
1. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the NRC.	
2. With analyses not being performed as required in Specification 3.21.G.1, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Report.	
3. The provisions of Definition J are not applicable.	

3.14-3.19/4.14-4.19 BASES

3.14/4.14 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3.15-3.18/4.15-4.18 FIRE SUPPRESSION SYSTEMS

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

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

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

3.19/4.19 FIRE BARRIER PENETRATION SEALS

The functional integrity of the fire barrier penetration seals ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected seal until the seal is restored to functional status.

Fire barrier penetration seals include cable penetration barriers, fire doors, and fire dampers.

3.21 & 4.21 BASES

3.21.A & 4.21.A INSTRUMENTATION

3.21.A.1 & 4.21.A.1 Liquid Effluent Monitoring

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive material in liquid effluents. The OPERABILITY and use of these instruments implements the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64. The alarm and/or trip setpoints for these instruments are calculated in the manner described in the ODAM to assure that the alarm and/or trip will occur before the limit specified in 10 CFR Part 20.106 is exceeded. Control of the normal liquid discharge pathway is assured by station procedures governing locked discharge valves and valve line-up verification.

3.21.A.2 & 4.21.A.2 Gaseous Effluent Monitoring

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The location of this instrumentation is indicated by a Figure in the ODAM, a simplified flow diagram showing gaseous effluent treatment and monitoring equipment. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in the ODAM, which have been reviewed by NRC, to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The process monitoring instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the augmented offgas treatment system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3.21.B & 4.21.B LIQUID EFFLUENTS

3.21.B.1 & 4.21.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 20.106. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within (1) the Section IV.A guides on technical specifications in Appendix I, 10 CFR Part 50, for an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Since service water is not a normal or expected source of significant radioactive release, routine sampling and monitoring for radioactivity is precautionary. An activity concentration of 3×10^{-6} μCi/ml in service water effluent is diluted in the discharge canal to about 1.5% of the 10 CFR 20 Appendix B Table 2 Column 2 concentration with only one circulating water pump operating. During normal Station operation the dilution would be even greater. By monitoring service water effluent continuously for radioactivity and by confirmatory sampling weekly, reasonable assurance that its activity concentration can be kept to a small fraction of the 10 CFR Part 20.106 limit and within the Specification 3.21.B.2.a limit is provided.

By monitoring service water continuously and liquid radwaste continuously during discharge with the monitor set to alarm or trip before the limit specified in 10 CFR 20.106 is exceeded, reasonable assurance of compliance with Specification 3.21.B.1.2 is provided. Verification that radioactivity in liquid effluent averaged only a small fraction of the concentration limit is provided by calculations demonstrating compliance with Specification 3.21.B.2.a.

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.2 & 4.21.B.2 Liquid Dose

Specifications 3.21.B.2, 3.21.C.2 and 3.21.C.3 implement the requirements of 10 CFR Part 50.36a and of 10 CFR Part 50, Appendix I, Section IV. These specifications state limiting conditions for operation (LCO) to keep levels of radioactive materials in LWR effluents as low as is reasonably achievable. Compliance with these specifications will also keep average releases of radioactive material in effluents at small percentages of the limits specified in 10 CFR Part 20.106. Surveillance Requirements provide for the measurement of releases and calculation of doses to verify compliance with the Specifications. Action statements in these Specifications implement the requirements of 10 CFR Part 50.36(c)(2) and 10 CFR Part 50, Appendix I, Section IV.A in the event an LCO is not met.

10 CFR Part 50 contains two distinctly separate statements of requirements pertaining to effluents from nuclear power reactors. The first concerns a description of equipment to maintain control over radioactive materials in effluents, determination of design objectives, and means to be employed to keep radioactivity in effluents ALARA. This requirement is stated in Part 50, Section 34a and Appendix I, Section II. Appendix I, Section III stipulates that conformance with the guidance on design objectives be demonstrated by calculations (since demonstration is expected to be prospective). The other is a requirement for developing limiting conditions for operation in technical specifications. It is stated in 10 CFR Part 50, Section 36a and Appendix I, Section IV. Both the intent of the Commission and the requirement are clearly stated in the Opinion of the Commission;[1] relevant paragraphs from that document follow:

> Section 50.36a(b) of 10 CFR Part 50 provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications which take into account the need for operating flexibility and at the same time ensure that the licensee will exert his best efforts to keep levels of radioactive materials in effluents as low as practicable. The Appendix I that we adopt provides more specific guidance to licensees in this respect.
>
> A. The Rule
>
> Section IV of Appendix I specifies action levels for the licensee. If, for any individual light water cooled nuclear power reactor, the quantity of radioactive material actually released in effluents to unrestricted areas during any calendar quarter is such as to cause radiation exposure, calculated on the same basis as the design objective exposure, which would exceed one-half the annual design objective exposure, the licensee shall make an investigation to identify the causes of these high release rates, define and initiate a program of action to correct the situation, and report these actions to the Commission within 30 days of the end of the calendar quarter.

The conclusion of the NRC Staff in the Appendix I Rulemaking Hearing[2] agrees with that of the Commission. The Staff recommended, "...that the limiting conditions for operation described in Appendix I, Section IV be applicable upon publication to technical specifications included in any license authorizing operation of a light water cooled nuclear power reactor..." (p. 73). (Cont'd)

3.21 & 4.21 BASES (Cont'd)

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.2 & 4.21.B.2 Liquid Dose (Cont'd)

The action to be taken by a licensee in the event a limiting condition is exceeded, is stated in Appendix I, Section IV.A and in the Opinion of the Commission.[3] Technical Specifications 3.21.B.2, 4.21.B.2, 3.21.C.2, 4.21.C.2, 3.21.C.3 and 4.21.C.3 for Cooper Station conform to this requirement.

Guidance for developing technical specifications for surveillance and monitoring is included in Appendix I, Section IV.B.

Although "it is expected that the annual releases of radioactive material in effluents from lightwatercooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II" (Appendix I, Section IV), no recommendation was made by either the Staff in its Concluding Statement[4] or by the Commission in its Opinion[5] that design objective values should appear as technical specification limits. The Opinion of the Commission and the statement of Appendix I are clear. Limiting conditions of operation (LCO) related to the quantity of radioactive material in effluents released to an unrestricted area stated in technical specifications shall conform to Appendix I, Section IV.A. Licensee action in the event an LCO is exceeded should be in accord with Section IV.A. Finally, surveillance and monitoring of effluents and the environment should conform to Section IV.B.

With the implementation of Specification 3.21.B.2 and 4.21.B.2 there is reasonable assurance that Station operation will not cause a radionuclide concentration in public drinking water taken from the River that exceeds the standard for anthropogenic radioactivity in community drinking water. The equations in the ODAM for calculating doses due to measured releases of radioactive material in liquid effluent will be consistent with the methodology in Regulatory Guides 1.109 and 1.113. The assessment of personal doses will examine potential exposure pathways including consumption of fish and water taken from the River downstream of the discharge canal.

Specification 3.21.B.2.c implements the requirements of 10 CFR Part 50.36a(a)(1) that operating procedures be established and followed and that equipment be maintained and used to keep releases to the environment as low as is reasonably achievable. The OPERABILITY of the liquid radwaste treatment system ensures that the appropriate portions will be available for use whenever liquid effluents require treatment prior to release to the environment. The specification that the portions of the system which were used to establish compliance with the design objectives in 10 CFR Part 50, Appendix I, Section II be used when specified provides reasonable assurance that releases of radioactive material in liquid effluent will be kept as low as is reasonably achievable. The activity concentration, 0.01 μCi/ml, below which liquid radwaste treatment would not be costbeneficial, and therefore not required, is demonstrated below:

The quantity of radioactive material in liquid effluent released annually from Cooper Station has been calculated to be[6]

total iodines	3.65 curies	
total others (less H^3)	0.7	
total	4.35 curies	(Cont'd)

3.21 & 4.21 BASES (Cont'd)

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.2 & 4.21.B.2 Liquid Dose (Cont'd)

The population dose commitment resulting from the radioactive material in liquid effluent released annually has been calculated to be

thyroid		1.95 manrem
total body		0.56
	total	2.5 manrem

Therefore, population doses are about 0.5 manrem per curie of iodine released and about 0.8 manrem per curie of other radionuclides (less H^3) released in liquids. It would be conservative to assume one manrem committed per curie released in liquid effluent.

The volume of liquid waste processed and intended for discharge is estimated to be:

Low Purity Waste	5700 gal/day	1.8×10^6 gal/yr
Chemical Waste + Demin Regenerant Waste	4000 gal/day	1.2×10^6 gal/yr

The annual costs to operate the radwaste processing equipment, neglecting credit for capital recovery, are estimated according to Regulatory Guide 1.110 to be:

Dirty Waste Ionex	\$ 88,000/yr
Evaporator	\$114,000/yr

Unit volume operating costs are about:

$$\text{Cost to ion exchanger} = \frac{\$\ 88{,}000}{1.8E{+}6\ \text{gal}} = \$0.05/\text{gal}$$

$$\text{Cost to evaporate} = \frac{\$114{,}000}{1.2E{+}6\ \text{gal}} = \$0.10/\text{gal}$$

Assuming the costbenefit balance is \$1,000 expenditure per manrem reduction and assuming teatment removes all radioactivity from the liquid, then

(1) the activity concentration in a batch below which treatment is not cost-beneficial is

$$C = \frac{\$\ 88{,}000}{1.8E{+}6\ \text{gal} \times 3785\ \frac{\text{ml}}{\text{gal}}} \times \frac{1\ \text{curie}}{\text{manrem}} \times \frac{10^6\ \mu\text{Ci}}{\text{curie}} \times \frac{1\ \text{manrem}}{\$1{,}000}$$

$$C = 0.013\ \mu\text{Ci/ml}$$

(Cont'd)

3.21 & 4.21 BASES (Cont'd)

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.2 & 4.21.B.2 Liquid Dose (Cont'd)

(2) the activity concentration below which evaporation is not cost-beneficial is

$$C = \frac{\$114{,}000}{1.2E{+}6 \text{ gal} \times 3785 \frac{\text{ml}}{\text{gal}}} \times \frac{1 \text{ curie}}{\text{manrem}} \times \frac{10^6 \ \mu\text{Ci}}{\text{curie}} \times \frac{1 \text{ manrem}}{\$1{,}000}$$

$$C = 0.025 \ \mu\text{Ci/ml}$$

Therefore, to one significant digit, radwaste treatment of liquids containing less than 0.01 μCi/ml is not justified.

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[1] NRC Commissioners, "Opinion of the Commission," in the Appendix I Rulemaking Hearing, Docket Rm502, p. 101102, April 30, 1975.

[2] NRC Staff, "Concluding Statement of the Regulatory Staff," in the Appendix I Rulemaking Hearing, Docket RM502, pp. 17, 69, 73, 115, February, 1974.

[3] NRC Commissioners, p. 101.

[4] NRC Staff, op. cit.

[5] NRC Commissioners, op. cit.

[6] Demonstration of Compliance with 10 CFR 50 Appendix I, Revision 1 and Supplement 2, Nebraska Public Power District, Cooper Nuclear Station, January 9, 1978.

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.3 & 4.21.B.3 Condensate Storage Tank and Outside Temporary Tanks

Restricting the quantity of radioactive material contained in the Condensate Storage Tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting dose commitment to an individual in an unrestricted area will not exceed 0.5 rem. Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

3.21.C & 4.21.C GASEOUS EFFLUENTS

3.21.C.1 & 4.21.C.1 Total Dose

Specification 3.21.C.1.a is included to assure that a measure of control is provided over the concentration of radionuclides in air entering the unrestricted area. Radioactive noble gases are monitored by instruments that provide a measure of release rate and cause automatic alarm when the noble gas concentration offsite is expected to exceed the unrestricted area limit specified in 10 CFR Part 20, Appendix B. With prompt action to reduce the radioactive noble gas concentration in effluent following alarm initiation, it can be maintained at a small fraction of the technical specification limit. The specified release rate limits restrict the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to $\leq$ (500) mrem/year to the total body or to $\leq$ (3000) mrem/year to the skin.

Radioiodines and radionuclides in particulate form are sampled with integrating samplers that permit assessment of the average release rate during each sample collection period. By complying with Specifications 3.21.C.2 and 3.21.C.3 the average offsite concentration will be maintained at a small fraction of the 10 CFR Part 20.106 concentration limit.

3.21.C.2 & 4.21.C.2 Noble Gases

Assessments of dose required by Specifications 4.21.C.2 and 4.21.C.3 to verify compliance with Appendix I, Section IV is based on measured radioactivity in gaseous effluent and on calculational methods stated in the ODAM. Pathways of exposure and location of individuals are selected such that the dose to a nearby resident is unlikely to be underestimated. Dose assessment methodology described in the ODAM for gaseous effluent will be consistent with the methodology in Regulatory Guides 1.109 and 1.111. Cumulative and projected assessments of dose made during a quarter are based on historical average, or reference (the same period of record used in the design objective Appendix I evaluation) atmospheric conditions. Assessments made for the annual radiological environmental report will be based on quarterly and annual averages of atmospheric conditions during the period of release.

The bases for Specifications 3.21.C.2 and 4.21.C.2 are also discussed in the bases for Specifications 3.21.B.2 and 4.21.B.2

3.21.C.3 & 4.21.C.3 Iodine and Particulates

The bases for Specifications 3.21.C.3 and 4.21.C.3 are discussed in the bases for Specifications 3.21.B.2 and 4.21.B.2.

3.21.C & 4.21.C GASEOUS EFFLUENTS (Cont'd)

3.21.C.4 & 4.21.C.4 Gaseous Radwaste System

The OPERABILITY of the gaseous radwaste treatment system and the ventilation exhaust treatment systems ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used 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 IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems are specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3.21.C.5 & 4.21.C.5 Hydrogen Concentration

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. While the Augmented Treatment System is in service the hydrogen and oxygen concentrations are prevented from reaching the flammability limits. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3.21.C.6 & 4.21.C.6 Air Ejector

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3.21.C.7 & 4.21.C.7 Containment

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas.

3.21.D & 4.21.D EFFLUENT DOSE LIQUID/GASEOUS

This specification is provided to meet the reporting requirements of 40 CFR Part 190.

3.21 & 4.21 BASES (Cont'd)

3.21.D & 4.21.D EFFLUENT DOSE LIQUID/GASEOUS (Cont'd)

In the event an analysis is required to determine compliance with 40 CFR 190, the dose to a member of the public due to radiation direct from the station will be estimated with the aid of environmental TLD, PIC, or similar environmental radiation dosimetry. A contribution from another fuel cycle facility is not added since there is no licensed fuel cycle facility within 50 miles of Cooper Station.

3.21.E & 4.21.E SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require materials processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

3.21.F & 4.21.F MONITORING PROGRAM

The radiological environmental monitoring program, including the land use census, is conducted to satisfy the requirements of 10 CFR Part 50, Appendix I, Section IV.B.2 and 3. 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.

The environmental monitoring program described in Table 3.21.F.1 is the minimum program which will be maintained. The Radiological Environmental Monitoring Manual (REMM) is an internal control document which describes in detail the actual monitoring program which is performed to ensure compliance with the specified minimum program. Control of the radiological environmental monitoring program, including the REMM, rests with the Environmental Affairs Division of the Power Operations and not the Cooper Nuclear Station organization.

The land use census is conducted annually to identify changes in use of the unrestricted area in order to recommend modifications in monitoring programs for evaluating individual doses from principal exposure pathways.

The need to adjust the program to current conditions and to assure that the integrity of the program is maintained are thereby provided. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3.21.G & 4.21.G INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in a Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid. Participation in an Interlaboratory Comparison Program is contingent upon availability of samples supplied by the NRC or samples approved by the NRC.

tary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Director of Power Supply within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. NPPD Safety Review and Audit Board.

The board must: verify that operation of the plant is consistent with company policy and rules, approved operating procedures and operating license provisions; review safety related plant changes, proposed tests and procedures; verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

Audits of selected aspects of plant operation shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all nuclear safety related activities is completed within a period of two years. Periodic review of the audit programs should be performed by the Board at least twice a year to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. The audits shall be performed in accordance with appropriate written instructions or procedures and should include verification of compliance with internal rules, procedures (for example, normal, off-normal, emergency, operating, maintenance, surveillance, test and radiation control procedures and the emergency and security plans), regulations involving nuclear safety and operating license provisions; training, qualification and performance of operating staff; and corrective actions following abnormal occurrences or unusual events. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. Written reports of such audits shall be reviewed at a scheduled meeting of the Board and by appropriate members of management including those having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

In addition to the above, the Safety Review and Audit Board will audit the facility Fire Protection Program, Radiological Environmental Monitoring Program, Offsite Dose Calculation Manual and their implementing procedures at least once every 24 months.

6.3 Station Operating Procedures

6.3.1 Station personnel shall be provided detailed written procedures to be used for operation and maintenance of system components and systems that could have an effect on nuclear safety.

6.3.2 Written integrated and system procedures and instructions including applicable check off lists shall be provided and adhered to for the following:

A. Normal startup, operation, shutdown and fuel handling operations of the station including all systems and components involving nuclear safety.

B. Actions to be taken to correct specific and forseen potential or actual malfunctions of safety related systems or components including responses to alarms, primary system leaks and abnormal reactivity changes.

C. Emergency conditions involving possible or actual releases of radioactive materials.

D. Implementing procedures of the Security Plan and the Emergency Plan.

E. Implementing procedures for the fire protection program.

F. Implementing procedures for the Offsite Dose Assessment Manual.

6.3.3 The following maintenance and test procedures will be provided to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

A. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

B. Routine testing of standby and redundant equipment.

C. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

D. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

E. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

6.3.4 Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR 20.

6.7 Station Reporting Requirements

6.7.1 Routine Reports

A. Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the NRC Region Administrator unless otherwise noted.

B. Startup Report

1. A summary report of plant startup and power escalation testing shall be submitted following:

   a. Receipt of an operating license.

   b. Amendment to the license involving a planned increase in power level.

   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.

   d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

   The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

2. Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

C. Annual Reports

Routine reports covering the subjects noted in 6.7.1.C.1 6.7.1.C.2, 6.7.1.C.3 and 6.7.1.C.4 for the previous calendar year shall be submitted prior to March 1 of each year.

6.7.1.C (Cont'd)

1. A tabulation 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 functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment 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.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, with a copy to the appropriate Regional Office, no later than the 15th of each month following the calendar month covered by the report.

1/ This tabulation supplements the requirements of §20.407 of 10CFR Part 20.

6.7.1 (Cont'd)

E. Annual Radiological Environmental Report

1. Routine radiological environmental reports covering the surveillance activities related to the Station operation during the previous calendar year shall be submitted to the NRC before May 1 of each year.

2. The Annual Radiological Environmental Report shall include the following:

   a. Summaries, interpretations, and an analysis of trends of results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment.

   b. A summary of the results of the land use census required in Specification 4.21.F.2.

   c. Summarized and tabulated results in the format of Table 6.7-1 of analyses of samples required by the radiological environmental monitoring program, and taken during the report period. 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. The missing data shall be submitted as soon as possible in a supplementary report.

   d. A summary description of the radiological environmental monitoring program including any changes; a map of all sampling locations keyed to a table giving distances and directions from the reactor; and the results of participation in the Interlaboratory Comparison Program, required by Specification 3.21.G.

   e. A summary of meteorological data collected during the year shall either be included in the Annual Radiological Environmental Report or retained by NPPD and made available to the NRC upon request.

TABLE 6.7-1
ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility Cooper Nuclear Station Docket No. 50-298

Location of Facility Nemaha, Nebraska (County, State) Reporting Period _____

Medium of Pathways Sampled (Unit of Measurement)	Type & Total No. of Analyses Performed	Lower Limit of Detection(1) (LLD)	All Indicator Locations Mean[](2) Range (2)	Location with Highest Annual Mean: Name, Distance & Direction	Location with Highest Annual Mean: Mean[](2) Range (2)	Control Locations Mean[](2) Range (2)	No. of Reportable Occurrences

Table Notes:

(1) Nominal Lower Limit of Detection (LLD) as defined in Definition K.A.
(2) Mean and Range based upon detectable measurements only. Fraction of detectable measurements at specified locations indicated in brackets [].

6.7.1 (Cont'd)

F. Semiannual Radioactive Material Release Report

1. A report of radioactive materials released from the Station shall be submitted to the NRC within 60 days after January 1 and July 1 of each year*. Each report shall include the information specified in Specification 6.7.1.F.2 covering the preceeding six months.

2. A Semiannual Radioactive Material Release Report shall include a summary by calendar quarter of the quantities of radioactive liquid and gaseous effluents and radioactive solid waste released from the Station. The data should be reported in the format recommended in Regulatory Guide 1.21, Appendix B, Tables 1, 2, and 3.

3. A Semiannual Radioactive Material Release Report shall include the following information related to each unplanned release radioactive material in gaseous or liquid effluent to offsite environs:

   a. A description of the event and equipment involved.

   b. Cause(s) of the unplanned release.

   c. Actions taken to prevent recurrence.

   d. Consequences of the unplanned release.

4. The report submitted within 60 days after January 1 of each year shall contain an assessment of off-site radiation doses due to radioactive liquid and gaseous effluents released from the Station during each calendar quarter of the year and during the year. The dose assessment shall be performed in accordance with methods compatible with the ODAM.

6.7.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of 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.

*It should be noted that this data has not normally been available to the District within 60 days and a verbal extension has typically been required from the NRC CNS Project Manager.

4. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of any unplanned criticality.

5. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

6. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For items 6.7.2.A.5 and 6.7.2.A.6 reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.7.2.B.2 and 6.7.2.B.3 below.

7. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

8. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

9. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

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

6.7.2 (Cont'd)

B. Thirty Day Written Reports

The reportable occurrences discussed below shall be the subject of written reports to the NRC Regional Administrator within thirty 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.

1. 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.

2. 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 6.7.2.B.1 and 6.7.2.B.2 need not be reported except where test results themselves reveal a degraded mode as described above.

3. 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.

4. Abnormal degradation of systems other than those specified in item 6.7.2.A.3 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.

5. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

   a. A description of the event and equipment involved.

   b. Cause(s) for the unplanned release.

   c. Actions taken to prevent recurrence.

   d. Consequences of the unplanned release.

6.7.2.B (Cont'd)

6. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 6.7-2 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 6.7-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 6.7-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.21.B.2.a, 3.21.C.2.a, and 3.21.C.3.a. 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 Report.

6.7.3 Unique Reporting Requirements

A. Testing Reports

Reports shall be submitted to the Director, Nuclear Reactor Regulation, USNRC, Washington, D.C. 20555, as follows:

Reports on the following area shall be submitted as noted:

Area	Reference	Submittal Date
1. Secondary Containment Leak Rate Testing (1)	4.7.C.1	90 Days After Completion of Each Test.

Note: (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.

B. Special Reports

Special reports (in lieu of Licensee Event Reports) may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

TABLE 6.7-2
REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	Reporting Levels				
Analysis	Water pCi/l	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, Wet)	Milk (pCi/l)	Food Products (pCi/Kg, Wet)
H-3	2E + 4(a)				
Mn-54	1E + 3		3E + 4		
Fe-59	4E + 2		1E + 4		
Co-58	1E + 3		3E + 4		
Co-60	3E + 2		1E + 4		
Zn-65	3E + 2		2E + 4		
Zr-Nb-95	4E + 2(b)				
I-131	2	0.9		3	1E + 2
Cs-134	30	10	1E + 3	60	1E + 3
Cs-137	50	20	2E + 3	70	2E + 3
Ba-La-140	2E + 2(b)			3E + 2	

(a) For drinking water samples. This is the 40 CFR 141 valve.
(b) Total for parent and daughter.

6.8 Environmental Qualification

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-46 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.9 Systems Integrity Monitoring Program

A program shall be established to reduce leakage from systems outside the primary containment that would or could contain highly radioactive fluids during a serious accident to as low as practical levels. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

6.10 Iodine Monitoring Program

A program shall be established to ensure that capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

6.11 PROCESS CONTROL PROGRAM (PCP)

6.11.1 The PCP shall be a manual detailing the program of sampling, analysis and formulation determination by which SOLIDIFICATION of radioactive waste from liquid systems is assured consistent with Specification 3.21.E and the surveillance requirements of these Technical Specifications.

6.11.2 District Initiated Changes

A. Shall be submitted to the Commission by inclusion in the Semiannual Radioactive Material Release Report for the period in which the change(s) was made effective and shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

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

6.11.2 District Initiated Changes (Cont'd)

3. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

B. Shall become effective upon review and acceptance by the SORC.

6.12 OFFSITE DOSE ASSESSMENT MANUAL (ODAM)

6.12.1 The ODAM 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.

6.12.2 District Initiated Changes

A. Shall be submitted to the Commission by inclusion in the Semi-annual Radioactive Material Release Report pursuant to Specification 6.7.1.D within 90 days of the date the change(s) was made effective and shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODAM to be changed with each page numbered and provided with a signed approval and date box, together with appropriate analyses of evaluations justifying the change(s).

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

3. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

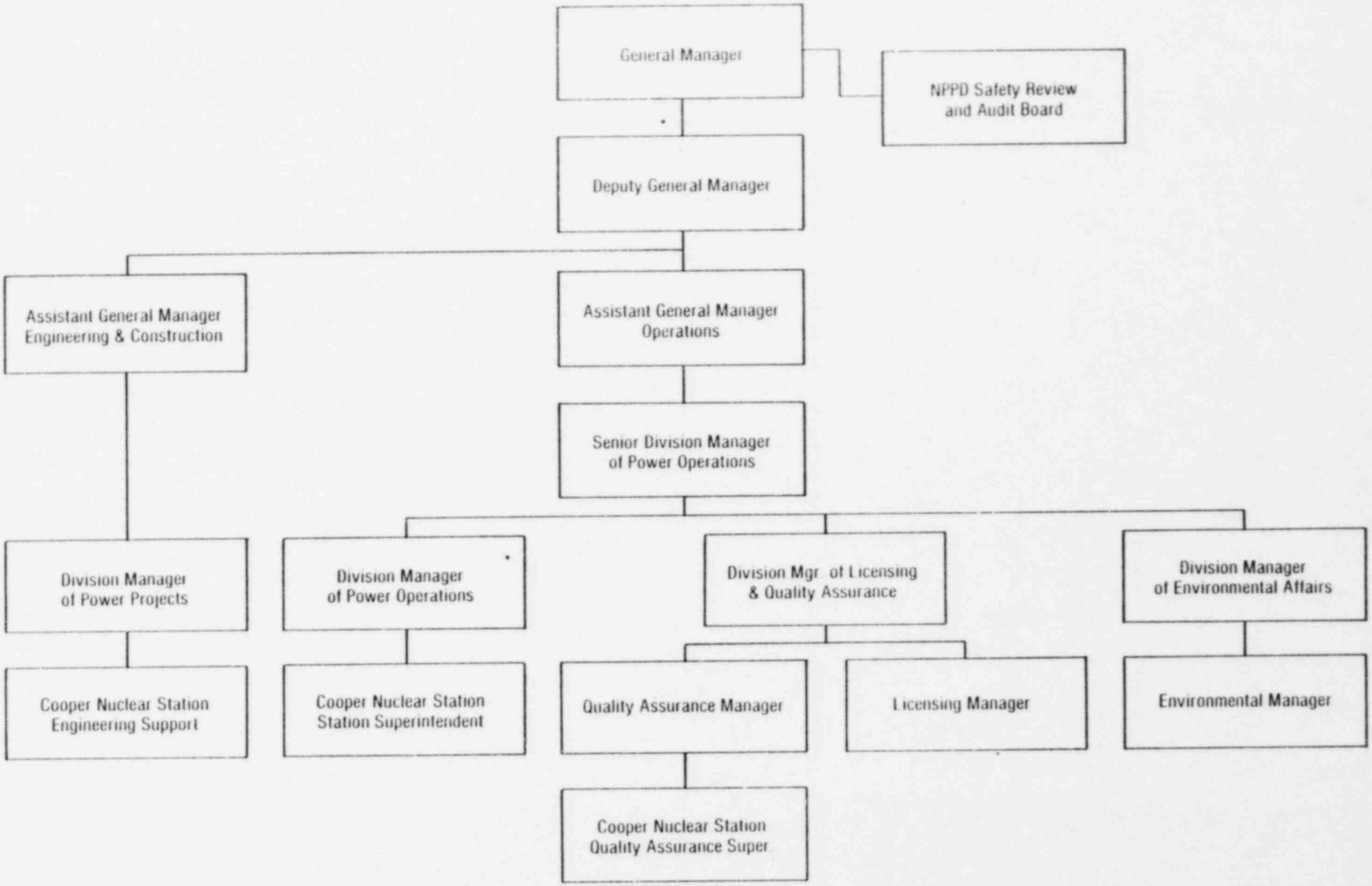
B. Shall become effective upon review and acceptance by the SORC.

6.13 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS, AND SOLID)

6.13.1 The radioactive waste treatment systems (liquid, gaseous, and solid) are those systems described in the facility Safety Analysis Report and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCO's set forth in Specifications 3.21.B, 3.21.C, 3.21.D, and 3.21.E. The NRC is notified of major changes to these systems under the provisions of 10 CFR Part 50.59 and Part 50.71 (USAR revisions).

Nebraska Public Power District

MANAGEMENT ORGANIZATION CHART



* Responsible for the Fire Protection Program

Figure 6.1.1
NPPD Management
Organization Chart

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

APPENDIX B

TO

OPERATING LICENSE NO. DPR-46

FOR THE

COOPER NUCLEAR STATION

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

(All 84 pages of these Appendix B Technical Specifications have been deleted in their entirety by the generation of Radiological Environmental Technical Specifications (RETS) in Appendix A.)