DCR 016

June 27, 1984

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
Post Office Box 14042, M.A.C. H-2
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued Amendment No. 69 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 17, 1983, as revised on November 1 and December 16, 1983, and March 22, 1984.

This amendment revises the TSs to require that releases of radioactive materials to unrestricted areas during normal operations be kept as low as is reasonably achievable. The amendment updates those portions of the TSs addressing radioactive waste management and brings them into compliance with current NRC regulations, including 10 CFR 50, Appendix I. This amendment eliminates TS Appendix B, Part I, and adds appropriate limiting conditions for operation, surveillance requirements and environmental monitoring requirements to Appendix A of the TSs. As stated in your letter dated March 22, 1984, these TSs will be implemented on July 1, 1984.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next monthly Federal Register Notice.

Sincerely,

Ronald W. Hernan

Ronald Hernan, Project Manager
Operating Reactors Branch No. 4
Division of Licensing

Division of Licensing

Enclosures:
1. Amendment No. 69
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	AD:OR:DL	OELD
RIngram	RHernan:ef	GRivenbark	GLainas	SA. TREBY
6/21/84	6/22/84	6/22/84	6/21/84	6/25/84

cc w/enclosure(s):

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Inverness, Florida 36250

Mr. Tom Stetka, Resident Inspector
U.S. Nuclear Regulatory Commission
Route #3, Box 717
Crystal River, Florida 32629

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Nuclear Plant Manager
Florida Power Corporation
P. O. Box 219
Crystal River, Florida 32629

Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Mr. R. W. Neiser, Senior
Vice President and General
Counsel
Florida Power Corporation
P. O. Box 14042
St. Petersburg, Florida 33733

Ulray Clark, Administrator
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, Florida 32301

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
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SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al (the licensees) dated January 17, 1983, as supplemented on November 1 and December 16, 1983, and March 22, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment becomes effective on July 1, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

George W Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

1. Replace the following existing pages of the Appendix "A" Technical Specifications with the enclosed pages and add the enclosed new pages (as indicated). The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

I
Ia (new)
IV
IVa (new)
VII
VIIa (new)
VIIIa (new)
XII
XIIa (new)
XIIIa (new)
XIV
XVI
1-6 thru 1-8
1-9 (new)
1-10 (new)
3/4 3-23
3/4 3-23a (new)
3/4 3-24
3/4 3-25
3/4 3-25a (new)
3/4 3-42 thru -54 (new)
3/4 7-48 thru -54 (new)
3/4 11-1 thru -15 (new)
3/4 12-1 thru -12 (new)
B3/4 3-6 (new)
B3/4 7-7 (new)
B3/4 7-8 (new)
B3/4 11-1 thru -4 (new)
B3/4 12-1 (new)
5-1
5-3a (new)
6-6
6-6a (new)
6-10 thru 6-12a
6-14
6-14a thru 6-14c (new)
6-15 thru 6-17
6-17a (new)
6-19
6-20
6-21 (new)

2. Remove Appendix B, Part I, in its entirety from the Technical Specifications. Ensure that Appendix B, Part II, (Environmental Protection Plan (Non-Radiological)) remains in the Appendix B section of the Technical Specifications.

DEFINITIONS

DEFINITIONS

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

SECTION PAGE

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.6 CONTAINMENT SYSTEMS

BASES

BASES

SECTION		PAGE
3/4.8	**ELECTRICAL POWER SYSTEMS**	
3/4.8.1	A. C. SOURCES	B 3/4 8-1
3/4.8.2	ONSITE POWER DISTRIBUTION SYSTEMS	B 3/4 8-1
3/4.9	**REFUELING OPERATIONS**	
3/4.9.1	BORON CONCENTRATION	B 3/4 9-1
3/4.9.2	INSTRUMENTATION	B 3/4 9-1
3/4.9.3	DECAY TIME	B 3/4 9-1
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BASES

BASES

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DESIGN FEATURES

ADMINISTRATIVE CONTROLS

SECTION PAGE

ADMINISTRATIVE CONTROLS

per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or designated components at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL POWER IMBALANCE

1.23 AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core expressed as a percentage of RATED THERMAL POWER minus the THERMAL POWER in the bottom half of the core expressed as a percentage of RATED THERMAL POWER.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.24 The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.25 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.26 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

SOURCE CHECK

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

PROCESS CONTROL PROGRAM (PCP)

1.28 The PROCESS CONTROL PROGRAM is the sampling, tests, analyses, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL is the document which contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

WASTE GAS SYSTEM

1.31 A WASTE GAS SYSTEM is any equipment (e.g., tanks, vessels, piping) capable of 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.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE - PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain 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.

INDEPENDENT VERIFICATION

1.35 INDEPENDENT VERIFICATION is a separate act of confirming or substantiating that an activity or condition has been completed or implemented, in accordance with specified requirements, by an individual not associated with the original determination that the activity or condition was completed or implemented in accordance with specified requirements.

LIQUID RADWASTE TREATMENT SYSTEM

1.36 The LIQUID RADWASTE TREATMENT SYSTEM shall be any available equipment (e.g., filters, evaporators) capable of reducing the quantity of radioactive material, in liquid effluents, prior to discharge.

MEMBER(S) OF THE PUBLIC

1.37 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their normal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

SITE BOUNDARY

1.38 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.39 An UNRESTRICTED AREA shall be any area at or beyond the site boundary, access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or industrial, commercial, institutional, and/or recreational purposes.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 280°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 280°F$
3. HOT STANDBY	< 0.99	0	$\geq 280°F$
4. HOT SHUTDOWN	< 0.99	0	$280°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
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 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool area i. Criticality Monitor	1	*	≤ 15 mr/hr	10^{-1}-10^{4} mr/hr	14
2. PROCESS MONITORS					
a. Reactor Building i. Gaseous Activity-RCS Leakage Detection	1	1,2,3,4	Not Applicable	10^{1}-10^{6} cpm	15
ii. Iodine Activity-RCS Leakage Detection	1	1,2,3,4	Not Applicable	10^{1}-10^{6} cpm	15
b. Control Room i. Iodine Activity-Ventilation System Isolation/Recirculation	1	All Modes	≤ 2 x background	10^{1}-10^{6} cpm	18

* With fuel in the storage pool or building

TABLE 3.3-6 (cont.)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTIO
2. Process Monitors (Continued)					
c. Condenser Vacuum Pump Exhaust Monitor - Gaseous Activity Monitor (RM-A12)	1	1, 2, 3, 4			19
d. Nuclear Services Closed Cooling Water Monitor (RM-L3)	1	All Modes			19
e. Decay Heat Closed Cooling Water Monitors (RM-L5 and RM-L6)	1	All Modes			19

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1

ACTION 18 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, plant operation may continue provided grab samples are collected and analyzed for gross activity* at least once per 24 hours.

* LLD for noble gas principal gamma emitters shall be $5x10^{-5}$ Ci/cc. LLD for liquid samples shall be $1x10^{-6}$ Ci/ml.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area				
i. Criticality Monitor	S	R	M	*
2. PROCESS MONITORS				
a. Reactor Building				
i. Gaseous Activity - RCS Leakage Detection	S	R	M	1,2,3,4
ii. Iodine Activity - RCS Leakage Detection	S	R	M	1,2,3,4
b. Control Room				
i. Iodine Activity - Ventilation System Isolation/ Recirculation	S	R	M	All Modes

* With fuel in the storage pool or building

TABLE 4.3-3 (cont.)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WH SURVEILLAN REQUIRED
2. Process Monitors (continued)				
c. Condenser Vacuum Pump Exhaust Monitor - Gaseous Activity Monitor (RM-A12)	D	R	M	1, 2, 3,
d. Nuclear Services Closed Cooling Water Monitor (RM-L3)	D	R	M	All MODES
e. Decay Heat Closed Cooling Water Monitors (RM-L5 and RM-L6)	D	R	M	All MODES

LIMITING CONDITION FOR OPERATION

3.3.3.2 As a minimum, the incore detectors shall be OPERABLE as specified below.

a. For AXIAL POWER IMBALANCE measurements:

1. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane with one plane at the core mid-plane and one plane in each axial core half.

2. The axial planes in each core half shall be symmetrical about the core mid-plane.

3. The detector strings shall not have radial symmetry.

b. For QUADRANT POWER TILT measurements with the Minimum Incore Detector System:

1. Two sets of 4 detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

2. Detectors in the same plane shall have quarter core radial symmetry.

c. For QUADRANT POWER TILT measurements with the Symmetric Incore Detector System at least 75% of the detectors in each core quadrant shall be OPERABLE.

APPLICABILITY: When the incore detection system is used for surveillance of:

a. The AXIAL POWER IMBALANCE, or

b. The QUADRANT POWER TILT.

ACTION:

With less than the specified minimum incore detector arrangement OPERABLE, do not use incore detector measurements to determine AXIAL POWER IMBALANCE or QUADRANT POWER TILT. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

TABLE 3.3-11
FIRE DETECTION INSTRUMENTS

DETECTOR LOCATION	MINIMUM DETECTORS OPERABLE HEAT/SMOKE
1. Control Complex	
a. Elevation 108'0"	
1. Zone 4 (Plant Battery Room 3B)	NA/1
2. Zone 5 (Plant Battery Room 3A)	NA/1
3. Zone 6 (Battery Charger Room 3B)	NA/1
4. Zone 7 (Battery Charger Room 3A)	NA/1
5. Zone 8 (4160V Switchgear Bus Room 3B)	NA/1
6. Zone 9 (4160V Switchgear Bus Room 3A)	NA/1
7. Zone 10 (Inverter Room 3B)	NA/1
8. Zone 11 (Inverter Room 3A)	NA/1
b. Elevation 120'0"	
1. Zone 5 (Control Rod Drive Equipment Room)	NA/2
2. Zone 7 (480V Switchgear Bus Room 3B)	NA/1
3. Zone 8 (480V Switchgear Bus Room 3A)	NA/1
c. Elevation 134'0"	
1. Zone 3A (Cable Spreading Room) or	NA/5
2. Zone 3B (Cable Spreading Room)	NA/3
d. Elevation 145'0"	
1. Zone 4 (Satellite Instrument Shop and Office)	NA/2
2. Zone 5 (Control Room)	1/6
e. Elevation 164'0"	
1. Zone 3 (HVAC Equipment Room)	NA/5
2. Zone 4 (HVAC Emergency Equipment 3B)	NA/1
3. Zone 5 (HVAC Emergency Equipment 3A)	NA/1
2. Auxiliary Building	
a. Elevation 119'0"	
1. Zone 20 (Emergency Diesel Generator 3B Controls Room)	1/NA
2. Zone 21 (Emergency Diesel Generator 3A Controls Room)	1/NA
3. Zone 27 (Emergency Diesel Generator Room 3B)	5/NA
4. Zone 28 (Emergency Diesel Generator Room 3A)	5/NA

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of specification 3.11.1.1 are not exceeded. The setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown on Table 3.3-12.

ACTION:

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

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-12. Exert best efforts to return the inoperable instrument(s) to OPERABLE status within 30 days. If the affected instrument(s) cannot be returned to OPERABLE status within 30 days, provide information on the reasons for inoperability and lack of timely corrective action in the next Semiannual Radioactive Effluent Release Report.

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

d. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 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 at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE			
a. Auxiliary Building Liquid Radwaste Effluent Line (RM-L2)	1	All MODES	21
b. Secondary Drain Tank Liquid Effluent Line (RM-L7)	1	All MODES	22
2. FLOW RATE MEASUREMENT DEVICES			
a. Auxiliary Building Liquid Radwaste Effluent Line	1	All MODES	23
b. Secondary Drain Tank Liquid Effluent Line	1	All MODES	23

Table 3.3-12 (Continued)
TABLE NOTATION

ACTION 21 With less than the required number of OPERABLE channels, effluent releases via this pathway may continue, provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. An INDEPENDENT VERIFICATION of release rate calculations is performed, and

c. An INDEPENDENT VERIFICATION of discharge valve lineup is performed.

Otherwise, suspend releases of radioactive materials via this pathway.

ACTION 22 With less than the required number of OPERABLE channels, effluent releases via this pathway may continue, provided that grab samples are collected and analyzed for gross radioactivity, at least once per 8 hours, at an LLD of at least 10^{-7} microcuries/ml.

ACTION 23 With less than the required number of OPERABLE channels, effluent releases via this pathway may continue, provided that the flow rate is estimated at least once per 4 hours during actual releases.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Auxiliary Building Liquid Radwaste Effluent Line (RM-L2)	D*	P	R(1)	M
b. Secondary Drain Tank Liquid Effluent Line (RM-L7)	D*	D	R(1)	M
2. FLOW RATE MEASUREMENT DEVICES				
a. Auxiliary Building Liquid Radwaste Effluent Line	D(2)	N.A.	R	N.A.
b. Secondary Drain Tank Liquid Effluent Line	D(2)	N.A.	R	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATION

* During periods of release.

(1) CHANNEL CALIBRATION shall be performed using:

a. One or more standards traceable to the National Bureau of Standards, or

b. Standards obtained from suppliers that participate in measurement assurance activities with the National Bureau of Standards, or

c. Standards related to previous calibrations performed using (a) or (b) above.

(2) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. A CHANNEL CHECK shall be performed at least once per day on any day that continuous, periodic or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with the effluent release isolation alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3-13

ACTION:

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

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-13. Exert best efforts to return the inoperable instrument(s) to OPERABLE status within 30 days. If the affected instruments cannot be returned to OPERABLE status within 30 days, provide information on reasons for inoperability and lack of timely corrective action in the next Effluent and Waste Disposal Semiannual Report.

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

d. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 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 frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Waste Gas Decay Tank Monitor (RM-A11)			
a. Noble Gas Activity Monitor*	1	All MODES	24
b. Effluent System Flow Rate Monitor	1	All MODES	26
2. Reactor Building Purge Exhaust Duct Monitor (RM-A1)			
a. Noble Gas Activity Monitor			
i. Operating Range*	1	All MODES	27
ii. Mid Range#	1	1, 2, 3 & 4	29
iii. High Range#	1	1, 2, 3 & 4	29
b. Iodine Sampler	1	All MODES	25
c. Particulate Sampler	1	All MODES	25
d. Effluent System Flow Rate Monitor	1	All MODES	26
e. Sampler Flow Rate Monitor	1	All MODES	26
3. Auxiliary Building and Fuel Handling Area Exhaust Duct Monitor (RM-A2)			
a. Noble Gas Activity Monitor			
i. Operating Range*	1	All MODES	28
ii. Mid Range#	1	1, 2, 3 & 4	29
iii. High Range#	1	1, 2, 3 & 4	29
b. Iodine Sampler	1	All MODES	25
c. Particulate Sampler	1	All MODES	25
d. Effluent System Flow Rate Monitor	1	All MODES	26
e. Sampler Flow Rate Monitor	1	All MODES	26

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* Provides control room alarm and automatic termination of release.

There is no isolation setpoint or release termination function for this monitor; Alarm setpoints are determined by the appropriate system procedures.

TABLE 3.3-13 (Continued)
TABLE NOTATION

ACTION 24 With less than the required number of OPERABLE channels, the contents of the Waste Gas Decay Tank may be released to the environment, provided that prior to initiating a release:

1. The Auxiliary Building & Fuel Handling Area Exhaust Duct Monitor (RM-A2) is OPERABLE with its setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoint shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL, or

2. a. At least two independent samples of the tank's contents are analyzed in accordance with Table 4.11-2 and

   b. An INDEPENDENT VERIFICATION of release rate calculations is performed, and

   c. An INDEPENDENT VERIFICATION of discharge valve lineup is performed.

Otherwise, suspend releases of radioactive effluents via this pathway.

ACTION 25 With the number of OPERABLE channels less than required, effluent releases via the affected pathway may continue, provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 26 With the number of OPERABLE channels less than required, effluent releases via this pathway may continue, provided flow rate is estimated at least once per 4 hours.

ACTION 27 With the number of OPERABLE channels less than required, immediately suspend PURGING of radioactive effluents via this pathway.

TABLE 3.3-13 (Continued)
TABLE NOTATION

ACTION 28 With the number of OPERABLE channels less than required, releases via this pathway may continue, provided grab samples are collected at least once per 12 hours and analyzed within 24 hours, and either the requirements of ACTION 24 Part 2 are met or Radiation Monitor RM-A11 is OPERABLE prior to releasing the contents of the Waste Gas Decay Tanks.

ACTION 29 With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

NOTE: Action Statement 3.3.3.9.a ~~is~~ not applicable

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS DECAY TANK MONITOR (RM-A11)					
a. Noble Gas Activity Monitor	P	P	R(1)	M	All MODES
b. Effluent System Flow Rate Monitor	P	N.A.	R	M	All MODES
2. REACTOR BUILDING PURGE EXHAUST DUCT MONITOR (RM-A1)					
a. Noble Gas Activity Monitor					
i. Operating Range	D#	P	R(1)	M	1, 2, 3, 4
ii. Mid Range	W	M	R(1)	M	1, 2, 3, 4
iii. High Range	W	M	R(1)	M	1, 2, 3, 4
b. Iodine Sampler	W#	N.A.	N.A.	N.A.	All MODES
c. Particulate Sampler	W#	N.A.	N.A.	N.A.	All MODES
d. Effluent System Flow Rate Monitor	D#	N.A.	R	M	All MODES
e. Sampler Flow Rate Monitor	D#	N.A.	R	M	All MODES
3. AUXILIARY BUILDING & FUEL HANDLING AREA EXHAUST DUCT MONITOR (RM-A2)					
a. Noble Gas Activity Monitor					
i. Operating Range	D#	P	R(1)	M	All MODES
ii. Mid Range	W	M	R(1)	M	1, 2, 3, 4
iii. High Range	W	M	R(1)	M	1, 2, 3, 4
b. Iodine Sampler	W	N.A.	N.A.	N.A.	All MODES
c. Particulate Sampler	W	N.A.	N.A.	N.A.	All MODES
d. Effluent System Flow Rate Monitor	D	N.A.	R	M	All MODES
e. Sampler Flow Rate Monitor	D	N.A.	R	M	All MODES

TABLE 4.3-9 (Continued)
TABLE NOTATION

During periods of Reactor Building Purge.

(1) CHANNEL CALIBRATION shall be performed using:

a. One or more standards traceable to the National Bureau of Standards, or

b. Standards obtained from suppliers that participate in measurement assurance activities with the National Bureau of Standards, or

c. Standards related to previous calibrations using (a) or (b) above.

WASTE GAS DECAY TANK - EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The Waste Gas Decay Tanks shall have one hydrogen and one oxygen monitoring channel OPERABLE.

APPLICABILITY: All MODES.

ACTION: With the number of OPERABLE channels less than required above, operation of this system may continue for up to 14 days, provided grab samples are collected and analyzed:

(a) at least once per 4 hours during degassing operations

(b) at least once per 24 hours during other operations.

SURVEILLANCE REQUIREMENTS

4.3.3.10 The Waste Gas Decay Tank explosive gas monitoring instrumentation shall be demonstrated operable by performing the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-10.

TABLE 4.3-10

WASTE GAS SYSTEM EXPLOSIVE GAS MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Hydrogen Monitors	D	Q*	M
2. Oxygen Monitors	D	Q*	M

* The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

Hydrogen Monitors

a. 1 volume percent hydrogen, balance nitrogen.
b. 4 volume percent hydrogen, balance nitrogen.

Oxygen Monitors

c. 1 volume percent oxygen, balance nitrogen.
d. 4 volume percent oxygen, balance nitrogen.

PLANT SYSTEMS

3/4 7.12 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.12 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers) in fire zone boundaries protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour either, establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.12 Each of the above required fire barrier penetrations shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetration to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration(s).

3/4.7.13 RADIOACTIVE WASTE SYSTEMS

WASTE GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.13.1 The quantity of radioactivity contained in each Waste Gas Decay Tank shall be limited to less than or equal to 39000 curies (considered as Xe 133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactivity in any Waste Gas Decay Tank exceeding the above limit, immediately suspend all additions of radioactive material to that tank, and within 48 hours reduce the tank contents to within its limit.

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

SURVEILLANCE REQUIREMENTS

4.7.13.1 The quantity of radioactive material contained in each Waste Gas Decay Tank shall be determined to be within the limit at least once per 7 days whenever radioactive materials are being added to the tank, and at least once per 24 hours during primary coolant system degassing operations.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13.2 The LIQUID RADWASTE TREATMENT SYSTEM shall be used, as required, to reduce radioactive materials in liquid wastes prior to their discharge, when projected monthly doses due to liquid effluents discharged to UNRESTRICTED AREAS (see Figure 5.1-3) would exceed the following values:

a. 0.06 mrem whole body;

b. 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. When radioactive liquid waste, in excess of the above limits, is discharged without prior treatment, prepare and submit to the Commission within 30 days, a Special Report pursuant to Specification 6.9.2, which includes the following information:

1. Identification of inoperable equipment and the reasons for inoperability.
2. Actions taken to restore the inoperable equipment to OPERABLE status.
3. Actions taken to prevent recurrence.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

LIQUID RADWASTE TREATMENT SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS

4.7.13.2 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

WASTE GAS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13.3 The WASTE GAS SYSTEM shall be used, as required, to reduce the radioactivity of materials in gaseous waste prior to discharge, when projected monthly air doses due to releases of gaseous effluents from the site to areas at or beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed:

1) 0.2 mrad gamma;
2) 0.4 mrad beta; and

The VENTILATION EXHAUST TREATMENT SYSTEM shall be used, as required, to reduce the quantity of radioactive materials in gaseous waste prior to discharge, when projected monthly air doses due to release of gaseous effluents from the site to areas at or beyond the SITE BOUNDARY would exceed:

1) 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. When the WASTE GAS SYSTEM and/or VENTILATION EXHAUST TREATMENT SYSTEM are not used and gaseous waste in excess of the above limits is discharged without prior treatment, prepare and submit to the Commission, within 30 days a Special Report, pursuant to Specification 6.9.2, which includes:

   1) Identification of the inoperable equipment and the reason(s) for inoperability.

   2) Actions taken to restore the inoperable equipment to OPERABLE status.

   3) Actions taken to prevent recurrence.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.13.3 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

WASTE SOLIDIFICATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13.4 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.13.4 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and boric acid solutions).

WASTE GAS DECAY TANK - EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.7.13.5 The concentration of oxygen in any Waste Gas Decay Tank shall be limited to less than or equal to 2% by volume whenever the concentration of hydrogen in that Waste Gas Decay Tank is greater than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

Whenever the concentration of hydrogen in any Waste Gas Decay Tank is greater than or equal to 4% by volume, and:

a. The concentration of oxygen in that Waste Gas Decay Tank is greater than 2% by volume, but less than 4% by volume, without delay begin to reduce the oxygen concentration to within its limit.

b. The concentration of oxygen in that Waste Gas Decay Tank is greater than or equal to 4% by volume, immediately suspend additions of waste gas to that Waste Gas Decay Tank and without delay begin to reduce the oxygen concentration to within its limit.

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

SURVEILLANCE REQUIREMENTS

4.7.13.5 The concentrations of hydrogen and oxygen in the Waste Gas Decay Tank shall be continuously monitored with the hydrogen and oxygen monitors required OPERABLE by Specification 3.3.3.10 or by sampling in accordance with Specification 3.3.3.10 if the hydrogen and/or oxygen monitors are inoperable.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released to UNRESTRICTED AREAS (see Figure 5.1-3) shall be less than or equal to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be less than or equal to 2×10^{-4} microcuries/ml. total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive materials released to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration of radioactive materials being released to UNRESTRICTED AREAS to within the above limits. If the concentration of radioactive materials being released in excess of the above limit~~s~~ is related to a plant operating characteristic, appropriate corrective measures (e.g., power reduction, plant shutdown) shall be taken to restore the concentration of radioactive materials being released to UNRESTRICTED AREAS to within the above limits.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed in accordance with the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methods in the OFFSITE DOSE CALCULATION MANUAL (ODCM) to assure the concentrations of radioactive material released from the site are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

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)[a]
A. Batch Waste Release Tanks[d] 1. Evaporator Condensate Storage Tanks (2) 2. Laundry & Shower Sump Tanks (2)	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e] 1. Secondary Drain Tank	Continuous[c]	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous[c]	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous[c]	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. 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):

$$LLD = \frac{4.66 s_b}{(E)\ (V)\ (2.22 \times 10^6)\ (Y)\ (e^{-\lambda \Delta t})}$$

Where:

LLD is the lower limit of detection as defined above (as microcurie 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 disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

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).

Typical values of E, V, Y, and Δt shall be used in the calculation.

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* The LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. 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.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously 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.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

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

f. The principal gamma emitters for which the LLD specification applies exclusively are 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 shall be reported as "less than" the nuclide's LLD, and shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS - DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited as follows:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and less than or equal to 5 mrem to any organ.

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITIY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission, within 30 days, a Special Report pursuant to Specification 6.9.2, which includes:

   1. Identification of the cause for exceeding the limit(s);

   2. Corrective action taken to reduce the release of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year so that the dose or dose commitment to a MEMBER OF THE PUBLIC from this source is less than or equal to 3 mrem total body and less than or equal to 10 mrem to any organ during the calendar year.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 DOSE CALCULATIONS. Cumulative dose contributions from liquid effluents shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) at least once per 31 days.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS - DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate at or beyond the SITE BOUNDARY, due to radioactive materials released in gaseous effluents (see Figure 5.1-3), shall be limited as follows:

a. Noble gases: less than or equal to 500 mrem/year total body and less than or equal to 3000 mrem/year to the skin.

b. Iodine-131, Tritium, and radioactive particulates with half-lives of greater than 8 days: less than or equal to 1500 mrem/year to any organ.

APPLICABILITY: At all times

ACTION:

a. With dose rate(s) exceeding the above limits, without delay decrease the dose rate to within the above limits(s). If the dose rate at or beyond the SITE BOUNDARY due to radioactive materials in gaseous effluents in excess of the above limits is related to a plant operating characteristic, appropriate corrective measures (e.g., power reduction, plant shutdown) shall be taken to decrease the dose rate to within the above limits.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the OFFSITE DOSE CALCULATION MANUAL (ODCM).

4.11.2.1.2 The dose rate due to radioactive materials specified above, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) by obtaining representative samples and performing analyses in accordance with Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml$)[a]
A. Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[f]	1×10^{-4}
B. Reactor Building Purge Exhaust Duct Monitor (RM-A1)	P Each Purge[c] Grab Sample	P Each Purge	Principal Gamma Emitters[b,f]	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building and Fuel Handling Area Exhaust Duct Monitor (RM-A2)	M[c] Grab Sample	M	Principal Gamma Emitters[b,f]	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as Listed in A, B, C above	Continuous[e]	W[d] Charcoal Sample	I-131	1×10^{-12}
	Continuous[e]	W[d] Particulate Sample	Principal Gamma Emitters[f] (I-131, Others)	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta & Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. 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):

$$LLD = \frac{4.66 s_b}{(E)\ (V)\ (2.22 \times 10^6)\ (Y)\ (e^{-\lambda \Delta t})}$$

Where:

LLD is the lower limit of detection as defined above (as microcurie 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 disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

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).

Typical values of E, V, Y, and Δt shall be used in the calculation.

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed between 2 and 6 hours following shutdown, startup or a change in power level exceeding 15% RATED THERMAL POWER within one hour.

c. Tritium grab samples shall be taken between 12 and 24 hours after flooding the refueling canal and at least once per 7 days thereafter while the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling and analyses shall be performed at least once per 24 hours for at least 7 days following each shutdown, startup or change in power level exceeding 15% of RATED THERMAL POWER within one hour, unless the Iodine Monitoring Channels in Radiation Monitors RM-A1 and RM-A2 show that the Radioiodine concentration in the Auxiliary Building and Fuel Handling Area or the Reactor Building Purge Exhaust Ducts will lead to a release which is less than 10% of the 10 CFR 20, Appendix B, Table II, Column I limits, at or beyond the SITE BOUNDARY.

e. 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 the Specifications 3.11.2.1, 3.11.2.2 and 3.1.2.3.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-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 shall be reported as "less than" the nuclide's LLD and shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose at or beyond the SITE BOUNDARY (see Figure 5.1-3), due to radioactive noble gases released in gaseous effluents shall be limited to:

a. During any calendar quarter: less than or equal to 5 mrad gamma and less than or equal to 10 mrad beta radiation, and

b. During any calendar year: less than or equal to 10 mrad gamma and less than or equal to 20 mrad beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission, within 30 days, a Special Report, pursuant to Specification 6.9.2, which includes:

   1) Identification of the cause for exceeding the limit(s).

   2) Corrective action taken to reduce the release of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year so that the average dose during the calendar year is less than or equal to 10 mrad gamma and 20 mrad beta radiation.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 DOSE CALCULATIONS: Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, TRITIUM, AND RADIOACTIVE PARTICULATES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Tritium, and radioactive particulates with half-lives greater than 8 days in gaseous effluents released from the site to areas at or beyond the SITE BOUNDARY (See Figure 5.1-3) shall be limited as follows:

a. During any calendar quarter: less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Tritium, and radioactive particulates with greater than 8 day half-lives, in gaseous effluents, exceeding any of the above limits, prepare and submit to the Commission, within 30 days, a Special Report, pursuant to Specification 6.9.2, which includes:

   1) Identification of the cause for exceeding the limits(s);

   2) Corrective action to reduce those releases during the remainder of the current calendar quarter and the remainder of the current calander year so that the average dose to any organ is less than or equal to 15 mrem.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

DOSE - IODINE-131, TRITIUM, AND RADIOACTIVE PARTICULATES (Continued)

SURVEILLANCE REQUIREMENTS

4.11.2.3 DOSE CALCULATIONS: Cumulative dose calculations for the current calendar quarter and current calendar year shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) at least once per 31 days.

RADIOACTIVE EFFLUENTS

3/4.11.3 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.3 The calendar year dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrems).

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made, which include direct radiation contributions from the reactor, to determine whether the above limits of Specification 3.11.3 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

3/4.11.3 TOTAL DOSE (Continued)

SURVEILLANCE REQUIREMENTS

4.11.3 DOSE CALCULATIONS - Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity, resulting from plant effluents, in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission, within 30 days of obtaining analytical results from the affected sampling period, a Special Report pursuant to Specification 6.9.2, which identifies the cause(s) for exceeding the limit(s) and defines corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 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 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is greater than or equal to the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

ACTION (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify the cause of the unavailability of samples and identify locations for obtaining replacement samples in the next Annual Radiological Environmental Operating Report. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

e. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure(s) in the OFFSITE DOSE CALCULATION MANUAL (ODCM) and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations	Sampling/ Collection Frequency	Type/Frequency of Analysis
1. AIRBORNE Radioiodine and particulates	One sample each: C07, C18, C40, C41, C46, and Control Location C47	Continuous sampler/ Weekly collection	Radioiodine canister: a) I-131 analysis weekly Particulate sampler: a) Gross β at $\geq$ 24 hours/ following weekly filter change. b) Composite gamma spectral analysis (by location)/ quarterly. (Gamma Spectral Analysis shall also be performed on individual samples if gross beta activity of any sample is greater than 1.0 p Ci/m^3 and which is also greater than ten times the control sample activity.)
2. DIRECT RADIATION	1) Site Boundary: C60, C61, C62, C63, C64, C65, C66, C67, C68, C69, C41, C70, C27, C71, C72, C73 2) Five Miles: C18, C03, C04, C74, C75, C76, C08, C77, C09, C78, C14G, C01, C79 3) Control Location: C47	Continuous placement/Quarterly collection	Gamma exposure rate/quarterly

TABLE 3.12-1 (Continued)

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations	Sampling/ Collection Frequency	Type/Frequency of Analysis
3. WATERBORNE Seawater	One sample each: C14H, C14G Control Location C13	Grab sample/Monthly	Gamma spectral analysis/monthly Tritium analysis on each sample or on a quarterly composite of monthly samples
Ground water	One sample: C40 (Control Location)	Grab sample/ semiannual	Gamma spectral and Tritium analysis/each sample
Drinking water	One sample each: C07, C10, C18 (All Control Locations)	Grab sample/quarterly	Gamma spectral and Tritium analysis/each sample
Shoreline sediment	One sample each: C14H, C14M, C14G Control Location C09	Semiannual sample	Gamma spectral analysis/each sample
4. INGESTION Fish & Invertebrates	One sample each: C29, Control Location C30	Quarterly: Oysters and carnivorous fish	Gamma spectral analysis on edible portions/each sample

TABLE 3.12-1 (Continued)

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations	Sampling/ Collection Frequency	Type/Frequency of Analysis
Food Products	One sample each: C48a*, C48b*, Control Location C47	Monthly (when available): Sample comprised of three (3) types of broad leaf vegetation from each location	Gamma spectral and I-131 analysis/each sample
	One sample: C19	Annual during harvest: Citrus	Gamma spectral analysis/ each sample
	One sample: C04	Annual during harvest: Watermelon	Gamma spectral analysis/ each sample

* Stations C48a and C48b are located at or beyond the 4400 ft. site boundary for gaseous effluents in the two sectors which yield the highest historical annual average D/Q values.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)
H-3	20,000[a]				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95[b]	400				
I-131	2[c]	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140[b]	200			300	

(a) For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

(b) An equilibrium mixture of the parent and daughter isotope which contains the reporting value of the parent isotope.

(c) For drinking water samples only.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD) a,d

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
gross beta		0.01				
^{3}H	2000[b]					
^{54}Mn	15		130			
^{59}Fe	30		260			
^{58}Co	15		130			
^{60}Co	15		130			
^{65}Zn	30		260			
^{95}Zr-Nb	15[c]					
^{131}I	1[f]	0.07[g]		1	60	
^{134}Cs	15	0.05[e]	130	15	60	150
^{137}Cs	18	0.06[e]	150	18	80	180
^{140}Ba-La	15[c]			15[c]		

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. 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):

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

Where:

LLD is the 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 disintegration),

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between sample collection (or end of the sample collection perod) and time of counting (for environmental samples, not plant effluent samples).

Typical values of E, V, Y and Δt should be used in the calculation.

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* The LLD is defined as an a priori (before the fact) limit representing the capability of the measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions. Occasionally, background fluctuations, unavoidable small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLD's unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

TABLE 4.12-1 (Continued)

TABLE NOTATION

b. LLD for drinking water. If no drinking water pathway exits, a value of 3000 pCi/l may be used.

c. The specified LLD is for an equilibrium mixture of parent and daughter nuclides which contain 15 pci/l of the parent nuclide.

d. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

e. Cs-134, and Cs-137 LLD's apply only to the quarterly composite gamma spectral analysis, not to analyses of single particulate filters.

f. LLD for drinking water. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

g. LLD for I-131 applies to a single weekly filter.

LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.1.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the land based meteorological sectors within a distance of five miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated by Specification 4.11.2.3, identify the new location in the next Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.c.

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) which is at least 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1.1, this location shall be added to the radiological environmental monitoring program within 30 days. The new sampling location shall replace the present sampling location, which has the lower calculated dose or dose commitment (via the same exposure pathway), after June 30 following this land use census. Identification of the new location and revisions of the appropriate figures from the OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be submitted with the next Semiannual Radioactive Effluent Release Report.

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

d. The reporting requirements of Specification 6.9.1.9.b are not applicable.

* Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LAND USE CENSUS (Continued)

SURVEILLANCE REQUIREMENTS

4.12.1.2 The land use census shall be conducted at least once per 12 months during the growing season by a door-to-door survey, aerial survey, or by consulting local agriculture authorities, using that information which will provide adequate results.

RADIOLOGICAL ENVIRONMENTAL MONITORING

INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. A summary of the results obtained from this program shall be included in the Annual Radiological Environmental Operating Report.

APPLICABILITY: At all times.

ACTION:

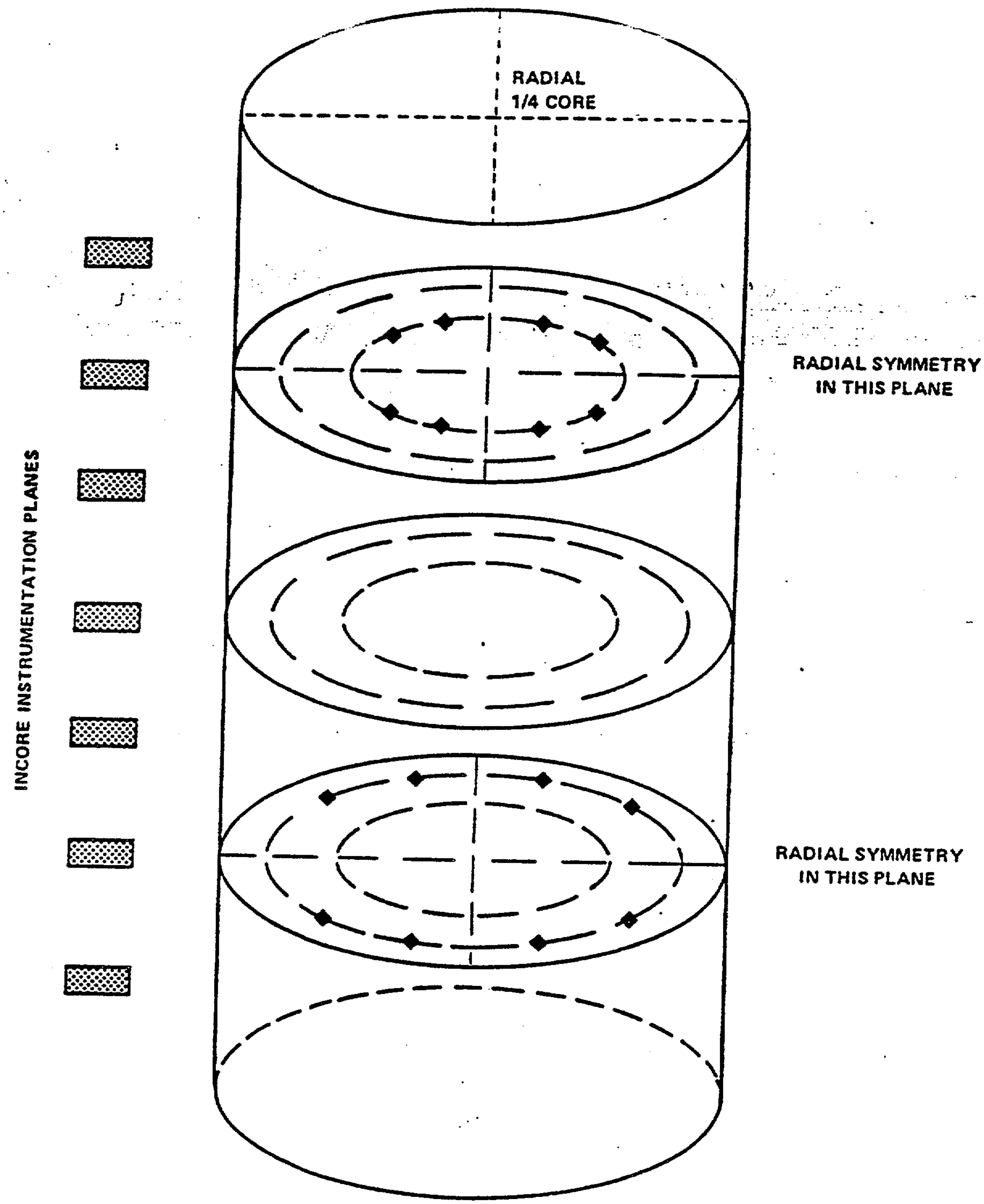
a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

c. The reporting requirements of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1.3 No surveillance requirements other than those required by the Interlaboratory Comparison Program.



Bases Figure 3-2 Incore Instrumentation Specification
Acceptable Minimum QUADRANT POWER TILT Arrangement

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the OFFSITE DOSE CALCULATION MANUAL (ODCM) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. 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.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

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 alarm/trip setpoints for these instruments are calculated in accordance with the procedures in the OFFSITE DOSE CALCULATION MANUAL (ODCM) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. 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.

3/4.3.3.10 WASTE GAS DECAY TANK - EXPLOSIVE GAS MONITORING INSTRUMENTATION

The OPERABILITY of the Waste Gas Decay Tank explosive gas monitoring instrumentation or the sampling and analysis program required by this specification provides for the monitoring (and controlling) of potentially explosive gas mixtures in the Waste Gas Decay Tanks.

3/4.7.13.1 WASTE GAS DECAY TANKS

Restricting the quantity of radioactivity contained in each waste gas decay tank provides assurance that in the event of a simultaneous uncontrolled release of all of the tanks' contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with FSAR accident analyses.

3/4.7.13.2 LIQUID WASTE TREATMENT

The requirement that these systems be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable" (ALARA). This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.7.13.3 WASTE GAS SYSTEM

The requirement that 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 reasonable achievable" (ALARA). This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were 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.

3/4.7.13.4 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36 and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.7.13.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the Waste Gas Decay Tanks is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits 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.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained 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.)

3/4.11.1.2 DOSE

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

3.4.11.2 GASEOUS EFFLUENTS

3.4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC, either within or outside the SITE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)(1)). For a MEMBER OF THE PUBLIC who may at time be within the SITE BOUNDARY, the occupancy of the MEMBER OF THE PUBLIC will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

3/4.11.2.2 DOSE, NOBLE GASES

This Specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable" (ALARA). The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the OFFSITE DOSE CALCULATION MANUAL (ODCM) for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of

3/4.11.2.2 DOSE, NOBLE GASES (Continued)

Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The OFFSITE DOSE CALCULATION MANUAL (ODCM) equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE, IODINE-131, TRITIUM, AND RADIOACTIVE PARTICULATES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable" (ALARA). The OFFSITE DOSE CALCULATION MANUAL (ODCM) calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The OFFSITE DOSE CALCULATION MANUAL (ODCM) methods for calculating the dose due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131, Tritium, and radioactive particulates with half-life less than eight days are dependent on the existing radionuclide pathways to man, in areas at and beyond the SITE BOUNDARY. The pathways which were examined in the development of these calculations were: 1) Individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leaf vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.3 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1 and 3.11.2. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

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

The LLD's required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at or beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census. Adequate information gained from door-to-door or aerial surveys or through consultation with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. 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.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an 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 the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which allows identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 130 feet.

b. Nominal inside height = 157 feet.

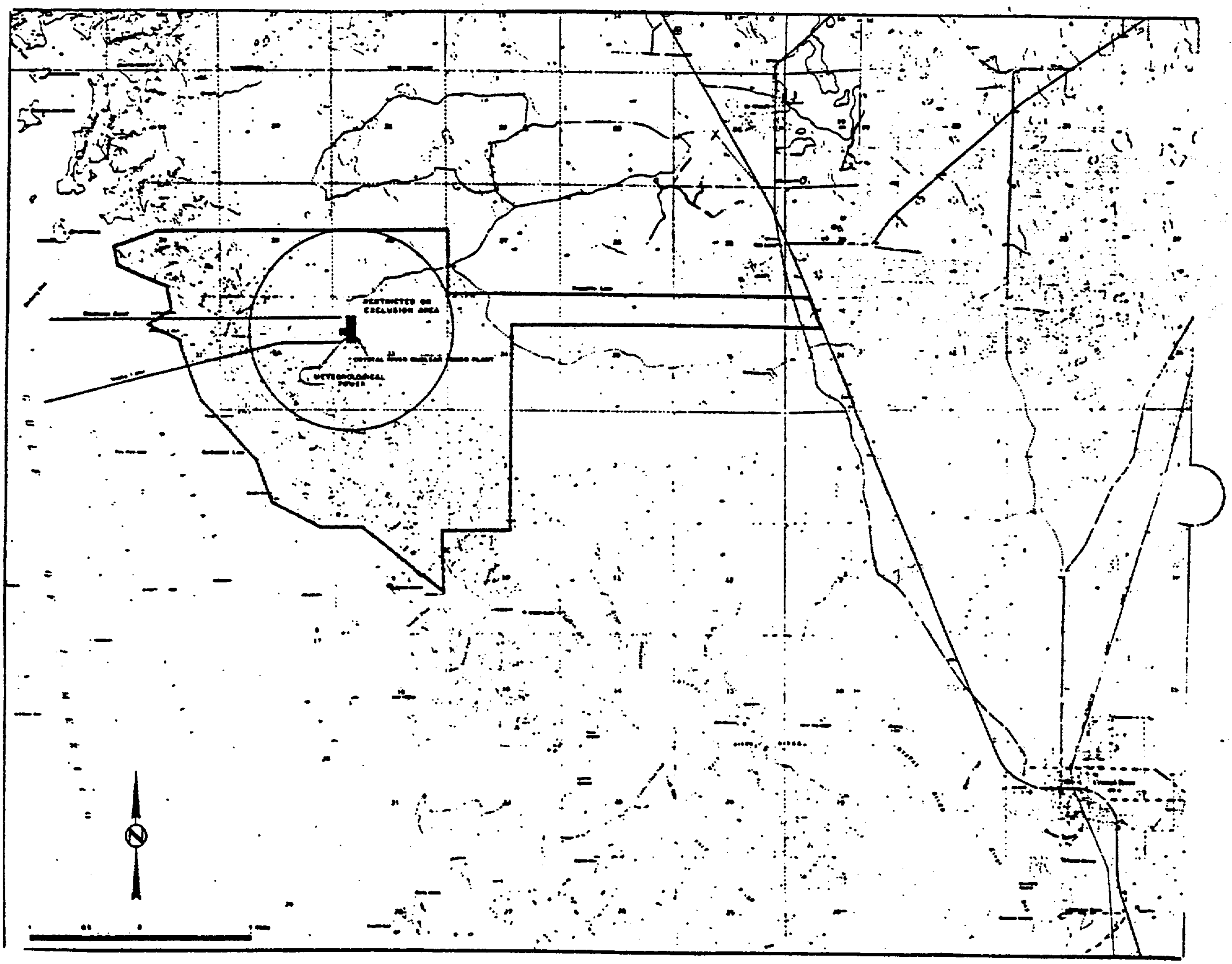
c. Minimum thickness of concrete walls = 3.5 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor pad = 12.5 feet.

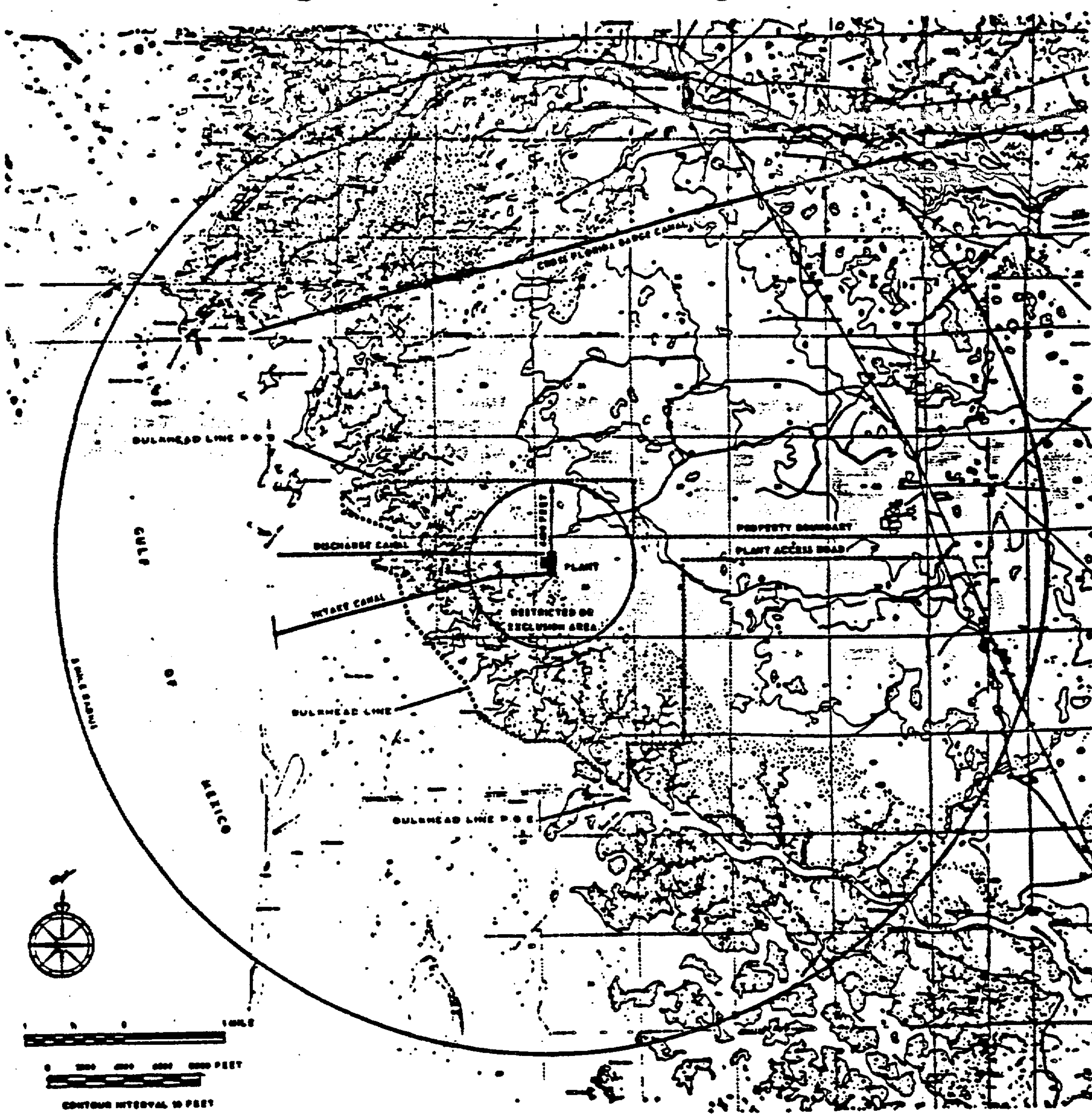
f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 2×10^6 cubic feet.



EXCLUSION AREA

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

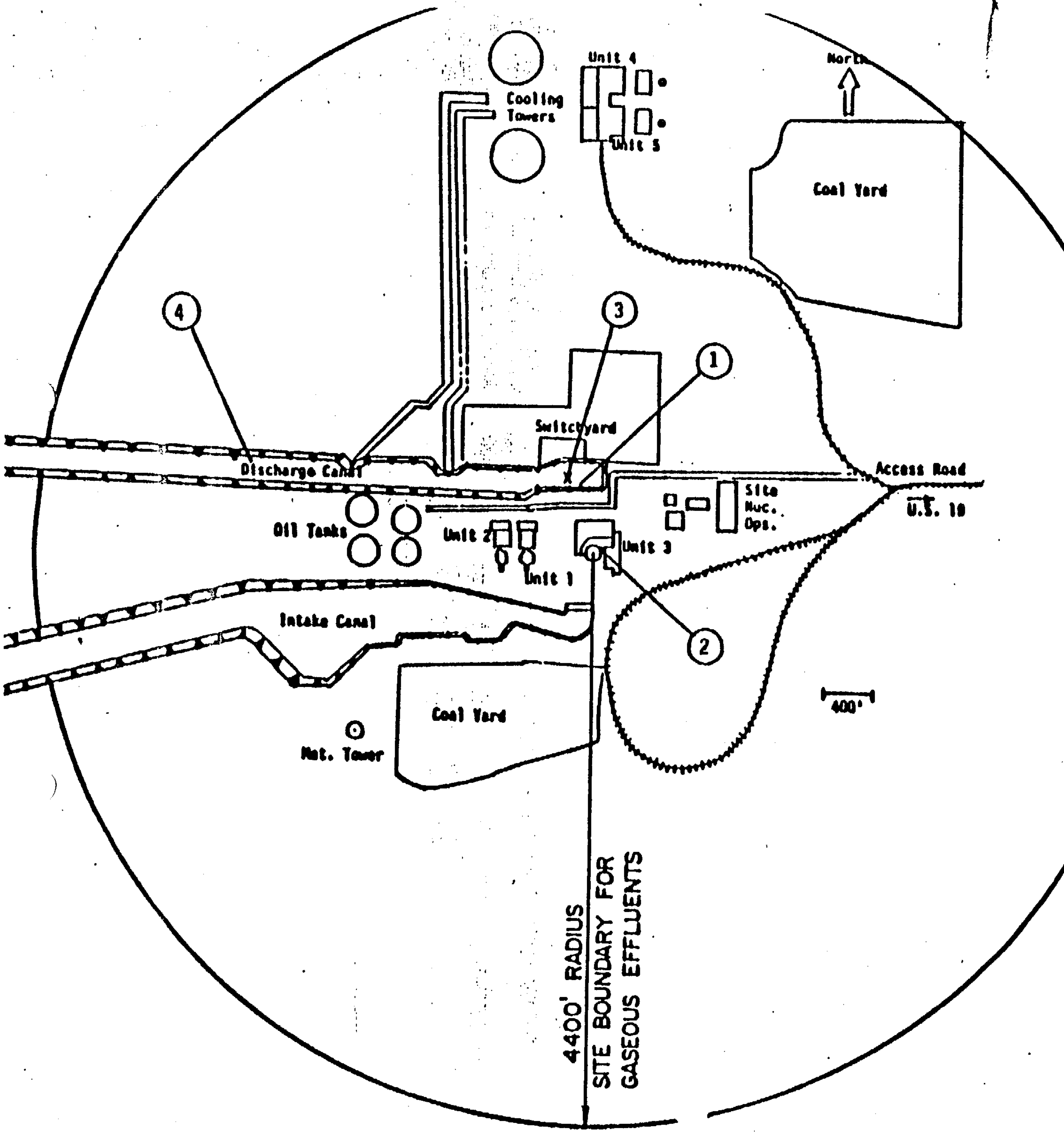


FIGURE 5.1-3

SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

1) DISCHARGE POINT FOR LIQUID EFFLUENTS

2) DISCHARGE POINT FOR GASEOUS EFFLUENTS

3) SITE BOUNDARY FOR LIQUID EFFLUENTS

4) UNRESTRICTED AREA FOR LIQUID EFFLUENTS WITHIN SITE BOUNDARY

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 55 psig and a temperature of 281°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The rector core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2253 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.30 weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 61 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

6.3 FACILITY STAFF QUALIFICATIONS

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

6.4 TRAINING

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

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Plant Training Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Nuclear Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of the:

Chairman:	Technical Services Superintendent
Member:	Operations Superintendent
Member:	Maintenance Superintendent
Member:	Nuclear Technical Services Superintendent (Security)
Member:	QA/QC Compliance Manager
Member:	Chem/Rad Protection Manager
Member:	Technical Support Engineer
Member:	Performance Engineering Supervisor
Member:	At Large (Designated by Chairman)
Member:	At Large (Designated by Chairman)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chairman or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsible for:

a. Review of 1) all procedures and changes thereto as required by Specification 6.8.2, 2) any other proposed procedures or changes thereto as determined by the Nuclear Plant Manager to affect nuclear safety.

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

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety, and changes to radwaste systems which could significantly alter their ability to meet Appendix I.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President, Nuclear Operations and to the Chairman of the Nuclear General Review Committee.

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

RESPONSIBILITIES (Continued)

k. Review of every unplanned on-site release of radioactive material to the environs, including the review and forwarding of reports covering evaluation, recommendations and disposition of the corrective action taken to prevent recurrence to the Plant Manager and Nuclear General Review Committee.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

CONSULTANTS

6.5.2.5 Consultants shall be utilized as determined by the NGRC Chairman to provide expert advice to the NGRC.

MEETING FREQUENCY

6.5.2.6 The NGRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.7 A quorum of NGRC shall consist of the Chairman or his designated alternate and five additional NGRC members, including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.8 The NGRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

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

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

REVIEW (Continued)

g. Events requiring 24-hour written notification to the Commission.

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

i. Reports and meetings minutes of the Plant Review Committee.

j. Changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the NGRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

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

g. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

h. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

i. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

AUDITS (Continued)

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. The facility fire protection program and implementing procedures at least once per 24 months.

h. Any other area of facility operation considered appropriate by the NGRC or the Senior Vice President-Engineering and Construction.

AUTHORITY

6.5.2.10 The NGRC shall report to and advise the Senior Vice President- Engineering and Construction on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of NGRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NGRC meeting shall be prepared, approved and forwarded to the Senior Vice President-Engineering and Construction within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President-Engineering and Construction within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be forwarded to the Senior Vice President-Engineering and Construction and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the PRC and submitted to the NGRC and the Vice President, Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the NGRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NGRC and the Vice President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Systems Integrity Program implementation.

h. Iodine Monitoring Program implementation.

i. PROCESS CONTROL PROGRAM implementation.

j. OFF-SITE DOSE CALCULATION MANUAL implementation.

k. Quality Assurance Program for effluent and environmental monitoring.

6.8 PROCEDURES (Continued)

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation as follows:

a. The Emergency Plan, Security Plan, Fire Protection Plan and implementing procedures, Administrative Instructions and those test procedures associated with plant modifications shall be reviewed and approved by the PRC and the Nuclear Plant Manager prior to implementation.

b. For all other procedures, the review cycle shall consist of: an intradepartmental review by a Qualified Reviewer, and interdisciplinary review by Qualified Reviewer(s) in interfacing departments, as specified in Administrative Procedures, and approval by the responsible Superintendent or Manager, as specified by Administrative Procedures. The PRC shall then review the 10 CFR 50.59 evaluation within 14 days of approval.

c. The training and qualification of Qualified Reviewers shall be governed by Administrative Procedures, with final certification by the Nuclear Plant Manager. Recertification will be required on a periodic basis and upon transfer between departments. As a minimum, all Qualified Reviewers shall meet the requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, or 4.6, or the equivalent.

d. Each procedure and administrative policy of 6.8.1 shall be reviewed on a periodic basis as set forth in Administrative Procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

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

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

c. The change is documented and subsequently reviewed and approved within 14 days of implementation, in accordance with the requirements of Specification 6.8.2.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing will be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup 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 requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 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 the resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL AND SEMIANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation 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 assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. A list of the reactor vessel material surveillance capsules installed in the reactor at the end of the report period and a summary of any withdrawals or insertions of capsules during the report period. In supplying this information, the ownership of each capsule shall be indicated and the irradiation location in the vessel of each capsule which was inserted during the report period shall be identified.

c. A routine Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, unachievable LLDs, and an analysis of trends of the results of the radiological environmental studies and previous Annual Radiological Environmental Operating Reports and an assessment of any observed impacts of the plant operation on the environment. If harmful effects or

[1] This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

ANNUAL AND SEMIANNUAL REPORTS (Continued)

evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples 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 results shall be submitted as soon as possible in a supplementary report.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from the reactor; the unavailability of milk or fresh leafy vegetable samples required by Table 3.12-1; the results of land use censuses required by Specification 3.12.1.2; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.1.3.

d. A semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

The Semiannual Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report which is submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of

ANNUAL AND SEMIANNUAL REPORTS (Continued)

an hour-by-hour listing of wind speed, wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to individuals due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the hypothetical worst case individual from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year.

Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1.

The radioactive effluent release report shall include the following information for each type of solid waste shipped offsite during the report period:

1) container volume,

2) total curie quantity (specify whether determined by measurement or estimate),

3) principal radionuclides (specify whether determined by measurement or estimate),

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* In lieu of submittal, these meteorological data may be retained in an on-site file and provided to the NRC upon request.

ANNUAL AND SEMIANNUAL REPORTS (Continued)

4) type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

5) type of container (e.g., LSA, Type A, Type B, Large Quantity), and

6) solidification agent (e.g., cement).

The Semiannual Radioactive Effluent Release Report shall include a list and description of unplanned releases to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Report shall include a description of any changes to the PROCESS CONTROL PROGRAM (PCP), the Radioactive Waste Treatment Systems, the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new Environmental Radiological Monitoring Program dose calculation location changes identified by the land use census made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, submitted no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety-system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. 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 startup rate of greater than 5.2 DPM or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. 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.

f. 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.

g. 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.

h. 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.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than that 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.

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

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

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

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

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above, designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specification 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meterorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Inoperable Fire Detection Monitoring Instrumentation, Specification 3.3.3.7.

f. Inoperable Fire Suppression System, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, and 3.7.11.4.

g. Dose due to radioactive materials in liquid effluents in excess of specified limits, Specification 3.11.1.2.

h. Dose due to noble gas in gaseous effluents in excess of specified limits, Specification 3.11.2.2.

i. Total calculated dose due to release of radioactive effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b (required by Specification 3.11.3).

j. Dose due to Iodine-131, Tritium, and radioactive particulates with greater than 8 day half-lives, in gaseous effluents in excess of specified limits, Specification 3.11.2.3.

k. Failure to process liquid radwaste, in excess of limits, prior to release, Specification 3.7.13.2.

l. Failure to process gaseous radwaste, in excess of limits, prior to release, Specification 3.7.13.3.

m. Measured levels of radioactivity in environmental sampling medium in excess of the reporting levels of Table 3.12-2, when averaged over any quarterly sampling period, Specification 3.12.1.1.

n. Unavailability of milk or fresh leafy vegetable samples, Specification 3.12.1.

o. Inoperable Mid or High Range Noble Gas Efficient Monitoring Instrumentation, Specification 3.3.3.8.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PRC and NGRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of analytical results required by the Operational Radiological Environmental Monitoring Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

6.12 HIGH RADIATION AREA (Continued)

b. A High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1a above, and in addition locked doors shall be provided to prevent unauthorized entry into such area. The keys shall be maintained under the administrative control of the Health Physics Supervisor with one key assigned to the administrative control of Shift Supervisor on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 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 1E 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-72 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable 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.

6.14 PROCESS CONTROL PROGRAM (PCP)

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

6.14.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made.

2. Shall become effective upon review and acceptance by the Plant Review Committee.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the OFFSITE DOSE CALCULATION MANUAL (ODCM):

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective.

2. Shall become effective upon review and acceptance by the Plant Review Committee.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

6.16.1 Licensee initiated major changes* to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Plant Review Committee or be included as part of the annual FSAR update.

2. May be implemented upon review and acceptance by the Plant Review Committee.

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* A major change to a radioactive waste system shall be any change which would alter the ability of the plant or system to meet the requirements of 10 CFR 50, Appendix I.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Florida Power Corporation (the licensee) has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The Florida Power Corporation filed this information with the Commission by letter dated January 17, 1983, as supplemented by letters dated November 1, and December 16, 1983, and March 22, 1984, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-72 for Crystal River Unit 3. The proposed Technical Specifications update those portions of the Technical Specifications addressing radioactive waste management and make them consistent with the current NRC staff positions as expressed in NUREG-0472. The supplemental revisions to the January 17, 1983, change request were, for the most part, requested by the staff to make the new Technical Specifications more consistent with staff positions, with other nuclear power plants in the State of Florida and with certain items implemented by NUREG-0737. These revised Technical Specifications will reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 REGULATIONS

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors", provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of Part 20.106, "Radioactivity in Effluents to Unrestricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the

periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases for radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

10 CFR Part 50, Appendix I, Section IV, provides guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0472 provides radiological effluent technical specifications for pressurized water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment sytems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent technical specifications may be acceptable if the staff determines that the alternatives are in compliance with the regulations and with the intent of the regulatory guidance.

The standard radiological effluent technical specifications can be grouped under the following categories:

(1) Instrumentation
(2) Radioactive effluents
(3) Radiological environmental monitoring
(4) Design features
(5) Administrative controls.

Each of the specifications under the first three categories is comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions for operation or surveillance requirements.

Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

3.0 EVALUATION

A Technical Evaluation Report (EGG-PHYS-6171) was prepared for us by EG&G Idaho, Inc. (EG&G) as part of our technical assistance contract program. Their report provides their Technical Evaluation of the compliance of the licensee's submittal with NRC provided criteria. The staff has reviewed this TER and agrees with the evaluation with the following exceptions. Because of the location of the plant, the staff considers that the licensee's commitment to extra airborne radioiodine samples provides a sufficient supplement to their proposed ingestion samples to meet the intent of NUREG-0472. In relation to the PCP, the licensee is presently operating under a Process Control Program (PCP) that is available for review by the NRC at any time. This is acceptable under NRC guidelines. A copy of the TER, minus the detailed Appendix, is enclosed.

3.1 SAFETY CONCLUSIONS

The proposed radiological effluent technical specifications for Crystal River Unit No. 3 have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 (Crystal River 3 is a pressurized water reactor) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes will not remove or relax any existing requirement related to the probability or consequences of accidents previously considered or needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area. We have determined that the amendment involves no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 27, 1984

Attachment: TER No. EGG-PHYS-6171 dated May 1983

Principal Contributors: C. Willis, W. Meinke.

Table 1. Relation Between Provisions of the Regulations and the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors and Boiling Water Reactors

● Indicate the specifications that are needed to assure compliance with the identified provision of the regulations.

	Standard Radiological Effluent Technical Specifications																												
	Instrumentation		Radioactive Effluents																Rad. Envir. Monitoring			Design Features	Administrative Control						
			Liquid				Gaseous																						
							PWR/BWR				PWR		BWR																
Provisions of Title 10 Code of Federal Regulations	Rad. Liquid Effl. Monitoring	Rad. Gas. Effl. Monitoring	Effluent Concentration	Dose	Liquid Radwaste Treatment	Liquid Holdup Tanks	Dose Rate	Dose Noble Gases	Dose I-131, Trit., and Part.	Explosive Gas Mixture	Gaseous Radwaste Treatment	Gas Storage Tanks	Gaseous Radwaste Treatment	Ventilation Exhaust Treatment	Main Condenser	Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Site Boundaries*	Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
§ 50.36a Technical specifications on effluents from nuclear power reactors																													
Remain within limits of § 20.106			●			●	●					●				●												●	
Establish and follow procedures to control effluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				●	
Maintain and use radioactive waste system equipment				●	●			●	●		●		●	●														●	
Submit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●	●
§§ 20.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●							●	
Part 50 Appendix A - General Design Criteria																													
Criterion 60 - Control of releases of radioactive materials to the environment	●	●			●	●				●	●	●	●	●	●	●	●							●			●	●	
Criterion 61 - Fuel storage and handling and radioactivity control	●	●																											
Criterion 63 - Monitoring fuel and waste storage	●	●																											
Criterion 64 - Monitoring radioactivity releases	●	●													●				●									●	
Part 50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●		
Part 50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																													
Maintain releases within design objectives				●	●			●	●		●			●														●	
Establish surveillance & monitoring program to provide data on:																													
(1) quantities of rad. matls. in effluents	●	●																											
(2) radiation & rad. matls. in the environment																			●						●	●			
(3) changes in use of unrestricted areas																				●									
Exert best efforts to keep releases "ALARA"				●	●			●	●		●		●	●															●
Submit report if calculated doses exceed the design objective				●				●	●		●			●											●				
Demonstrate conform. to des. obj. by calc. proced.				●				●	●		●			●														●	
Part 100															●														

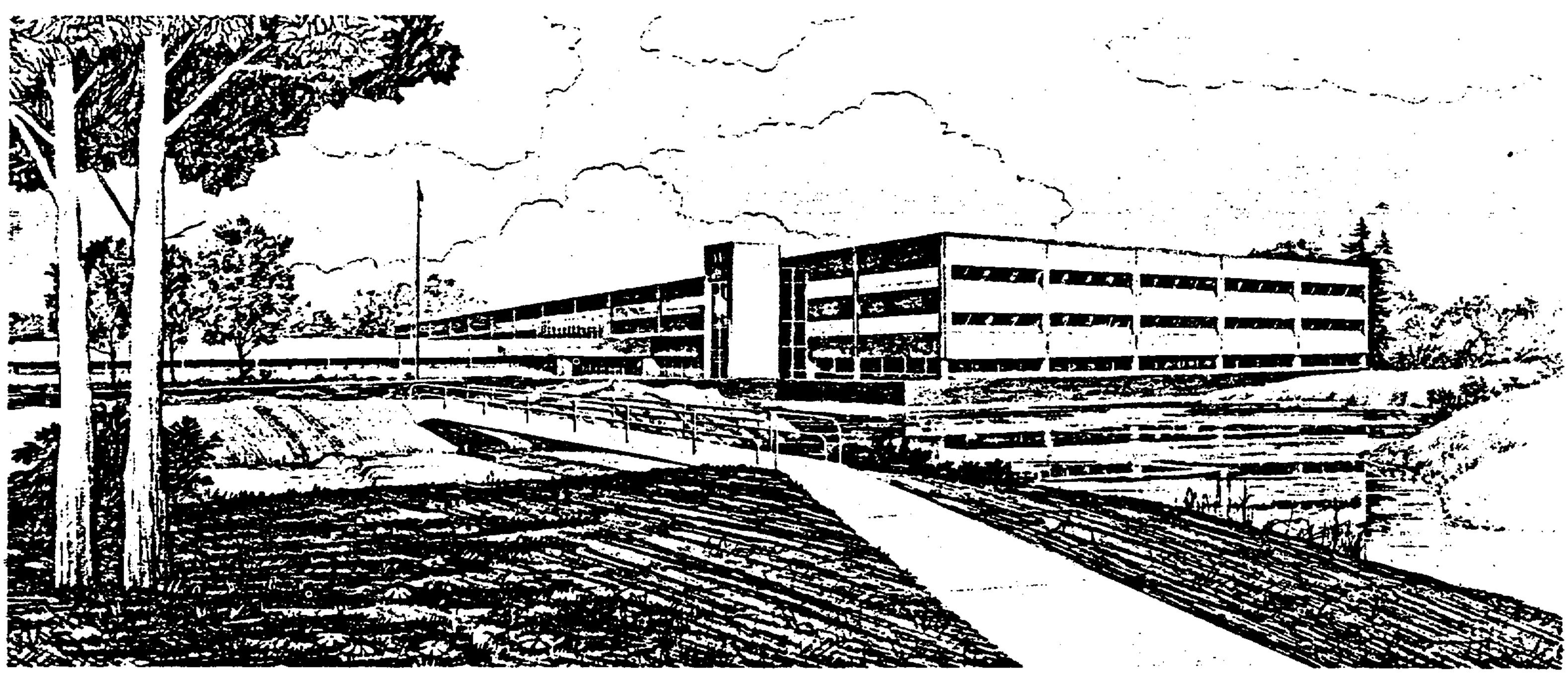
*Note: Needed to fully implement other specifications.

EGG-PHYS-6171
May 1983

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) IMPLEMENTATION - CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

William Serrano
Stephen W. Duce
John W. Mandler
Ferrol B. Simpson
Douglas W. Akers
Thomas E. Young

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570

EG&G Idaho

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EG&G Idaho, Inc.
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FOREWORD

This Technical Evaluation Report was prepared by EG&G Idaho, Inc. under a contract with the U. S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Systems Integration) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

ABSTRACT

A review of the Radiological Effluent Technical Specifications (RETS) of the Crystal River Unit No. 3 Nuclear Generating Plant was performed. The principal review guidelines used were NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," and Draft 7' of NUREG-0472, Revision 3, "Radiological Effluent Technical Specifications for PWR's." Draft submittals were discussed with the Licensee until all items requiring changes to the Technical Specifications were resolved. The Licensee then submitted final proposed RETS to the NRC which were evaluated and found to be in compliance with the requirements of the NRC review guidelines with the exception of one item in the environmental monitoring program. The proposed Offsite Dose Calculation Manual was reviewed.

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APPENDIX

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1.0 INTRODUCTION

1.1 Purpose of the Technical Evaluation

The purpose of this Technical Evaluation Report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Crystal River Unit No. 3 Nuclear Generating Plant with regard to Radiological Effluent Technical Specifications (RETS), the proposed Offsite Dose Calculation Manual (ODCM), and the Process Control Program (PCP).

The evaluation used criteria proposed by the Nuclear Regulatory Commission (NRC) staff in the model Technical Specifications for pressurized water reactors (PWR's), NUREG-0472,[1] and subsequent revisions. This effort is directed toward the NRC objective of implementing RETS which comply with the regulatory requirements, primarily those of 10 CFR Part 50, Appendix I.[2] Other regulations pertinent to the control of effluent releases are also included within the scope of compliance.

1.2 Generic Issue Background

Since 1970, 10 CFR Part 50, Section 50.36a,[3] "Technical Specifications on Effluents from Nuclear Power Reactors," has required licensees to provide Technical Specifications which ensure that radioactive releases will be kept as low as is reasonably achievable (ALARA). In 1975 numerical guidance for the ALARA requirement was issued in 10 CFR Part 50, Appendix I. The licensees of all operating reactors were required[4] to submit, no later than June 4, 1976, their proposed ALARA Technical Specifications and information for evaluation in accordance with 10 CFR Part 50, Appendix I.

However, in February 1976 the NRC staff recommended that proposals to modify Technical Specifications be deferred until the NRC completed the model RETS. The initial NRC position on the model RETS was established in May 1978 when the NRC's Regulatory Requirements Review Committee approved the first model RETS (NUREG-0472 for PWR's and NUREG-0473 for boiling water reactors [BWR's]).

The model RETS deal with radioactive waste management systems and environmental monitoring. Although the model RETS address the 10 CFR Part 50, Appendix I requirements, subsequent revisions include provisions for addressing issues not covered in Appendix I. These provisions are stipulated in the following regulations:

- 10 CFR Part 20,[5] "Standards for Protection Against Radiation," Sections 20.105(c), 20.106(g), and 20.405(c) which require that nuclear power plants and other licensees comply with 40 CFR Part 190,[6] "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

- 10 CFR Part 50, Appendix A,[7] "General Design Criteria for Nuclear Power Plants," which contains Criterion 60--Control of releases of radioactive materials to the environment; Criterion 63--Monitoring fuel and waste storage; and Criterion 64-Monitoring radioactive releases.

- 10 CFR Part 50, Appendix B,[8] which establishes the quality assurance required for nuclear power plants.

Copies of the model RETS were sent to licensees in July 1978 with a request to submit proposed site-specific RETS on a staggered schedule over a six-month period. Licensees responded with requests for clarifications and extensions.

The Atomic Industrial Forum (AIF) formed a task force to comment on the model RETS. NRC staff members first met with the AIF task force on June 17, 1978. The model RETS were subsequently revised (Revision 1) to reflect comments from the AIF and others. A principal change was the transfer of much of the material concerning dose calculations from the model RETS to a separate document, the ODCM.

Revision 1 of the model RETS was sent to licensees on November 15 and 16, 1978 with guidance (NUREG-0133)[9] for preparation of the RETS and the

ODCM and a new schedule for responses, again staggered over a six-month period.

Four regional seminars on the RETS were conducted by the NRC staff during November and December 1978. Subsequently, a preliminary copy of Revision 2 of the model RETS and additional guidance on the ODCM and a PCP were issued in February 1979 to each utility at individual meetings. NUREG-0472, Revision 2[1] and NUREG-0473, Revision 2[10] were published in July 1979 and updated in January 1980 and February 1980. In response to the NRC's request, operating reactor licensees subsequently submitted initial proposals on plant RETS and the ODCM. Review leading to ultimate implementation of these documents was initiated by the NRC in September 1981 using subcontracted independent teams as reviewers.

As the RETS reviews progressed, feedback from the licensees led the NRC to modify some of the provisions in the February 1, 1980 version of Revision 2 to clarify specific concerns of the licensees and thus expedite the reviews. Starting in April 1982, the NRC distributed revised versions of RETS in draft form to the licensees during the site visits. The new guidance on these changes was presented in an AIF meeting on May 19, 1982.[11] Some interim changes regarding the Radiological Environmental Monitoring Section were issued in August 1982.[12] With the incorporation of these changes, the NRC issued a draft Revision 3 of NUREG-0472[13] in September 1982 to serve as new guidance for the review teams.

1.3 Plant-Specific Background

In conformance with the 1975 directive[4] Florida Power Corporation (FPC), the Licensee of the Crystal River Unit No. 3 Nuclear Generating Plant, had the NUS Corporation submit information for an "Appendix I Analysis for Crystal River Nuclear Unit," dated May 28, 1976.[14] This information was evaluated by the Nuclear Regulatory Commission Division of Site Safety and Environmental Analysis and found to meet the requirements of Appendix I.[15] The Licensee did not propose new RETS at this time.

The RETS were addressed in the next submittal by the Licensee[16] to the NRC dated February 13, 1979. The submittal followed the format of NUREG-0472 for PWR's. EG&G Idaho Inc. (EG&G), selected as an independent task review team, initiated a review and evaluation of this submittal. The submittal was compared with the model RETS and assessed for complaince with the requirements of 10 CFR Part 50, Appendix I, and 10 CFR Part 50, Appendix A.

Review comments and questions dated November 23, 1981[17] were mailed to the NRC and the Licensee prior to arranging a site visit with the Licensee. The Licensee chose to prepare another submittal based on the questions included in the review and postponed the site visit. The Licensee's revised RETS (Revision 1) were received July 15, 1982 by EG&G. This submittal was reviewed for compliance with Revision 3 of the model RETS. Review comments and questions on the Revision 1 submittal (September 14, 1982[18]) were mailed to the NRC and the Licensee prior to a site visit at the Crystal River Unit No. 3 Generating Plant. The site visit was arranged for the purpose of resolving questions identified in the review of the Revision 1 submittal.

During the site visit (October 27-28, 1982), the Licensee presented a revised RETS (Revision 2) which, together with technical discussions, resolved most of the shortcomings of the Crystal River RETS (e.g., missing information and other deviations from the requirements) identified in the review of the Revision 1 submittal.

On November 11, 1982 another RETS submittal (Revision 3) was received which reflected the agreements made at the October 27-28, 1982 meeting. This document was compared with Revision 3 of NUREG-0472[13] to ensure that all items met the intent of the model RETS requirements proposed by the NRC. A telephone conference call was held on December 13, 1982[19] between NUS and EG&G personnel (NUS is preparing the FPC submittal). This telecon clarified the few unresolved items identified in the review of the Revision 3 submittal. A telecon was then held between NRC and EG&G personnel on December 13, 1982[20] for concurrence on deviations identified in the submittal. It was agreed that the Licensee's Revision 3 submittal was acceptable. Based on the acceptance of these reviews the FPC submitted final proposed RETS to the NRC.

A copy of FPC's final proposed Technical Specifications[21] was received by the EG&G review team on January 28, 1983. The Licensee's final proposed

RETS submittal was reviewed against NUREG-0472[13] and it was concluded that one open item remained in the environmental monitoring program. All other items regarded as deviations from the intent of the NUREG-0472 requirements were resolved, allowing the EG&G review team to complete a TER for submittal to the NRC.

The Licensee's proposed ODCM[22] was received by the EG&G review team on May 3, 1983. The ODCM submitted was reviewed against NUREG-0133 and Regulatory Guide 1.109 and it was concluded that the ODCM contains methods consistent with the criteria of NUREG-0133. Discrepancies in the ODCM are identified in the cover letter for this document.

The Licensee has not submitted a PCP. The Licensee has committed to having a PCP in the Technical Specifications. Therefore, a PCP will be submitted to the NRC for review and approval.

2.0 REVIEW CRITERIA

Review criteria for the RETS were provided by the NRC in three documents:

1. NUREG-0472, RETS for PWR's
2. NUREG-0473, RETS for BWR's
3. NUREG-0133, Preparation of RETS for Nuclear Power Plants.

Twelve essential criteria are given for the RETS and ODCM:

1. All significant releases of radioactivity shall be controlled and monitored.

2. Offsite concentrations of radioactivity shall not exceed the 10 CFR Part 20, Appendix B, Table II limits.[23]

3. Offsite radiation doses shall be ALARA.

4. Equipment shall be maintained and used to keep offsite doses ALARA.

5. Radwaste tank inventories shall be limited so that failures would not cause offsite doses exceeding 10 CFR Part 20 limits.

6. Hydrogen and/or Oxygen concentrations in the waste gas system shall be controlled to prevent explosive mixtures.

7. Wastes shall be processed to shipping and burial ground criteria under a documented program, subject to quality assurance verification.

8. An environmental monitoring program, including a land use census, shall be implemented.

9. The radwaste management program shall be subject to regular audits and reviews.

10. Procedures for control of liquid ~~and~~ gaseous effluents shall be maintained and followed.

11. Periodic and special reports on environmental monitoring and on releases shall be submitted.

12. Offsite dose calculations shall be performed using documented and approved methods consistent with NRC methodology.

In addition to NUREG-0472 and NUREG-0473, and their subsequent revisions, the NRC staff issued guidelines,[24,25] clarifications,[26,27] and branch positions[28,29,30] establishing a policy that requires the licensees of operating reactors to meet the intent, if not the letter, of the model RETS requirements. The NRC branch positions issued since the RETS implementation review began have clarified the model RETS for operating reactors.

Review criteria for the ODCM is based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual"[31]; NUREG-0133[9]; and Regulatory Guide 1.109[32]. The format for the ODCM is left to the Licensee and may be simplified by tables and grid printouts.

Review criteria for the Process Control Program is based on guidance provided by the NRC staff[33].

3.0 TECHNICAL EVALUATION

3.1 General Description of Radiological Effluent System

This section briefly describes the liquid and gaseous radwaste effluent treatment systems, release paths, and control systems installed at Crystal River Unit No. 3 Nuclear Generating Plant, a PWR.

3.1.1 Radioactive Liquid Effluents

Miscellaneous wastes from (a) radioactive laboratory drains, (b) building and equipment drains and sumps, (c) regeneration solution for deborating demineralizers, (d) demineralizer backwash, and (e) radioactive laundry and shower drains are processed through the miscellaneous liquid process system. The miscellaneous processing system consists of the miscellaneous waste storage tank, cation demineralizer, miscellaneous waste evaporator, evaporator condensate demineralizer, and evaporator condensate storage tanks. The evaporator condensate demineralizers and storage tanks are common to both the primary coolant process system and the miscellaneous liquid process system. The contents of the evaporator condensate tanks may be transferred to the reactor coolant bleed tanks for feed to the primary system or to the nuclear service seawater system for release to the discharge canal.

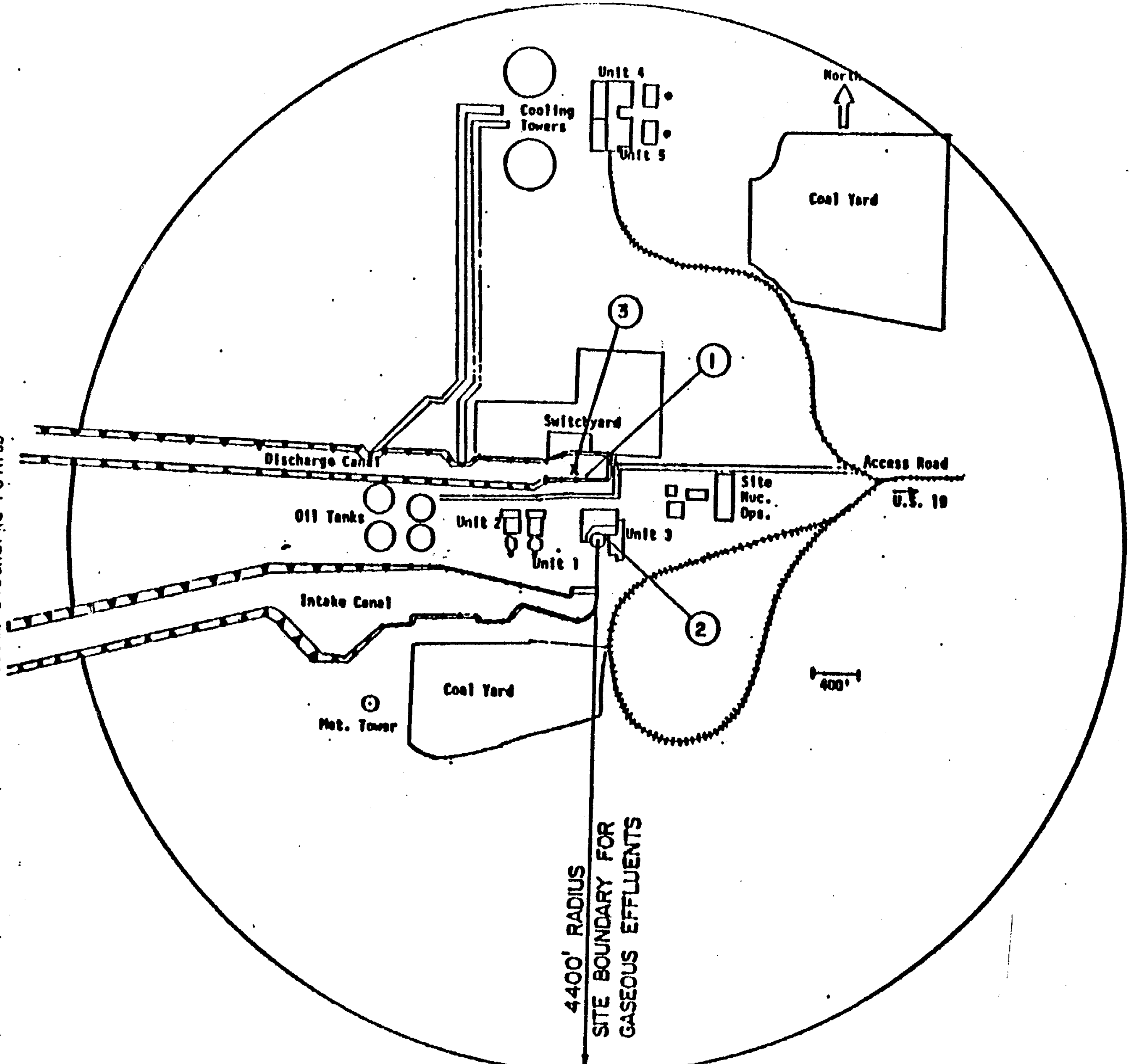
The secondary drain tank receives liquid from the turbine building drains and sump, any leakage from the component cooling water (CCW) and service water (SW) systems and the steam generator blowdown. These liquids are not treated prior to collection in the secondary drain tank as the liquid radioactivity concentration is normally very low.

Cooling water is withdrawn from and returned to the Gulf of Mexico. The CCW and SW systems are both closed loop systems, i.e., have no direct discharge pathway.

The secondary drain tank discharges to the nuclear service seawater system. This discharge path is monitored by RM-L7, which will terminate the discharge upon reaching the alarm/trip setpoint. The auxiliary building radwaste line receives liquid wastes from the (a) laundry and shower tanks (A and B), and (b) radwaste evaporator condensate storage tanks (A and B). The discharge is monitored by RM-L2, which will terminate the discharge upon reaching the alarm/trip setpoint, and goes to the nuclear service seawater discharge system. The intake and discharge canals constructed for the plant are considered to be navigable waters and the State of Florida requires public access to navigable waters. The radioactive liquid effluents are released to the discharge canal as shown in Figure 1. Figure 1 identifies the site boundary for liquid effluents as being in the discharge canal; however, it is not clear if it is a site boundary or an unrestricted area within the site boundary. A schematic diagram of the liquid radwaste and the miscellaneous liquid process systems are shown in Figures 2 and 3, respectively.

3.1.2 Radioactive Gaseous Effluents

The waste gas vent header system is essentially split into two sections: one section within the reactor building and one section within the auxiliary building. Condensing water vapor and liquids entering the section of the vent header system within containment drain to the reactor coolant drain tank, while those entering the vent header system within the auxiliary building drain to the miscellaneous waste storage tank. The vent line from the reactor coolant drain tank discharges to the miscellaneous waste storage tank. The gases from the miscellaneous waste storage tank and the three reactor coolant bleed tanks are joined and discharged to the suction of the waste gas compressors via an intermediate waste gas surge tank. The compressed gas portion of the waste gas system starts at the waste gas compressors and includes the three waste gas decay tanks (WGDT's).



1.) DISCHARGE POINT FOR LIQUID EFFLUENT

2.) DISCHARGE POINT FOR GASEOUS EFFLUENTS

3.) SITE BOUNDARY FOR LIQUID EFFLUENTS

Figure 1. Liquid and Gaseous Discharge Points

Figure 2. Liquid Radwaste System

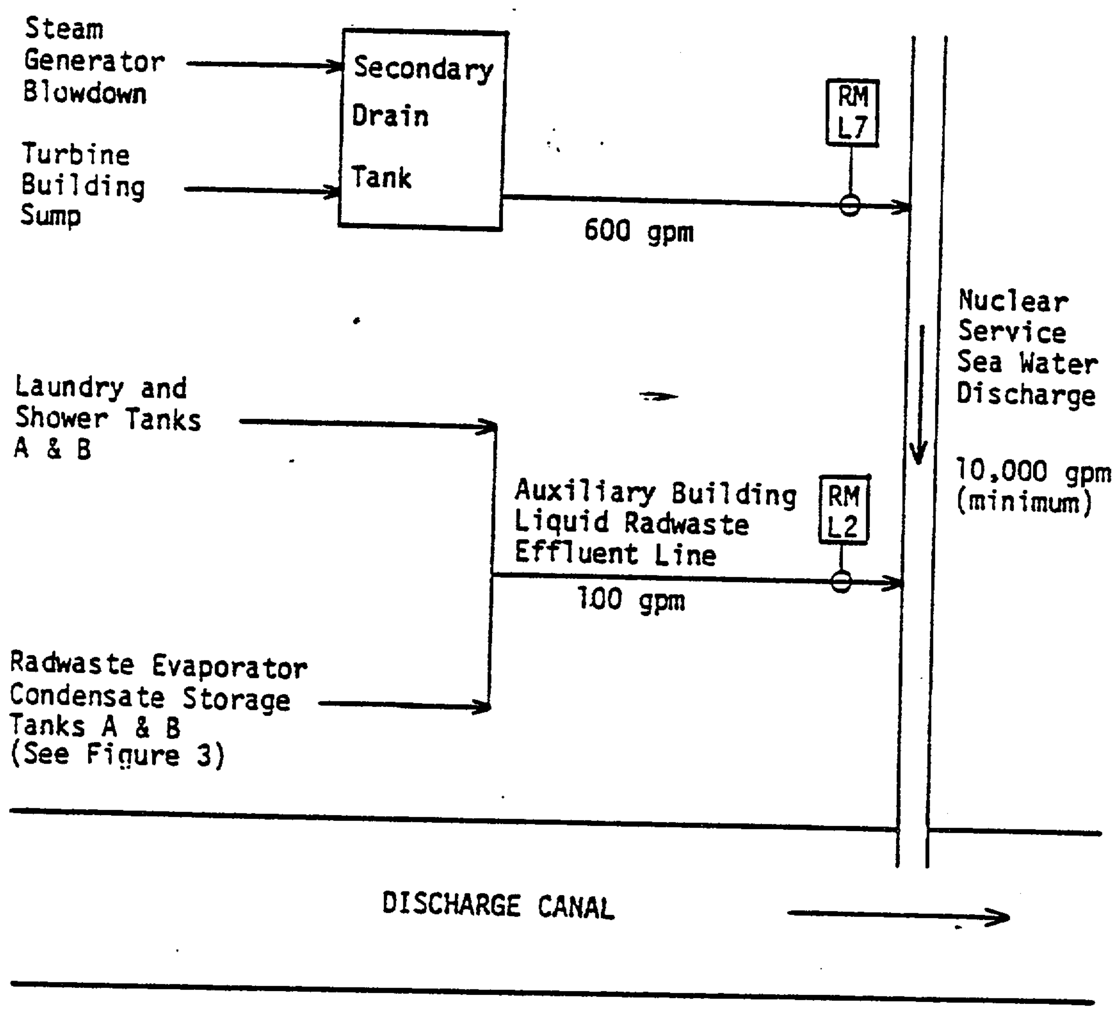


Figure 3. Miscellaneous Liquid Process System

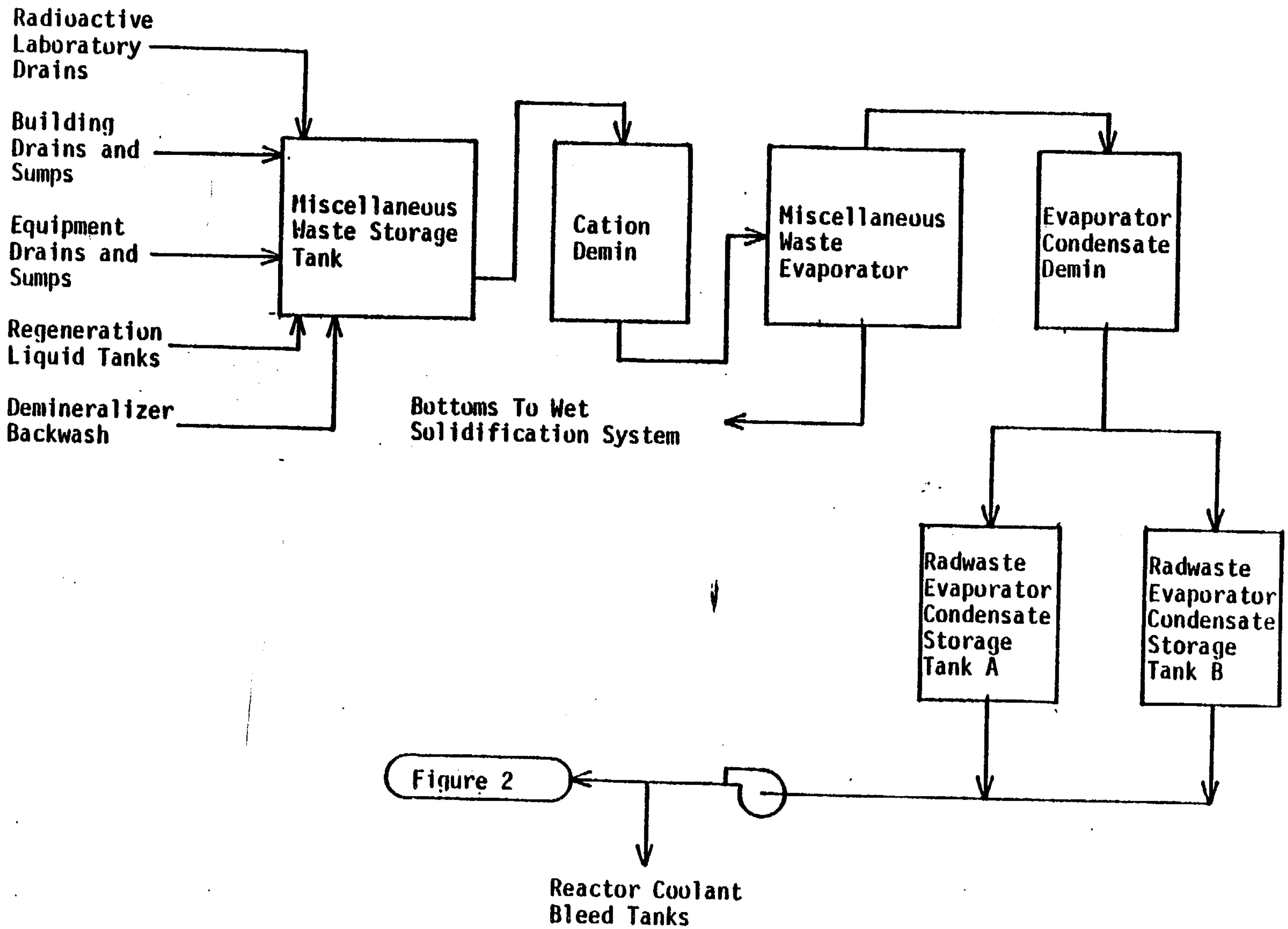
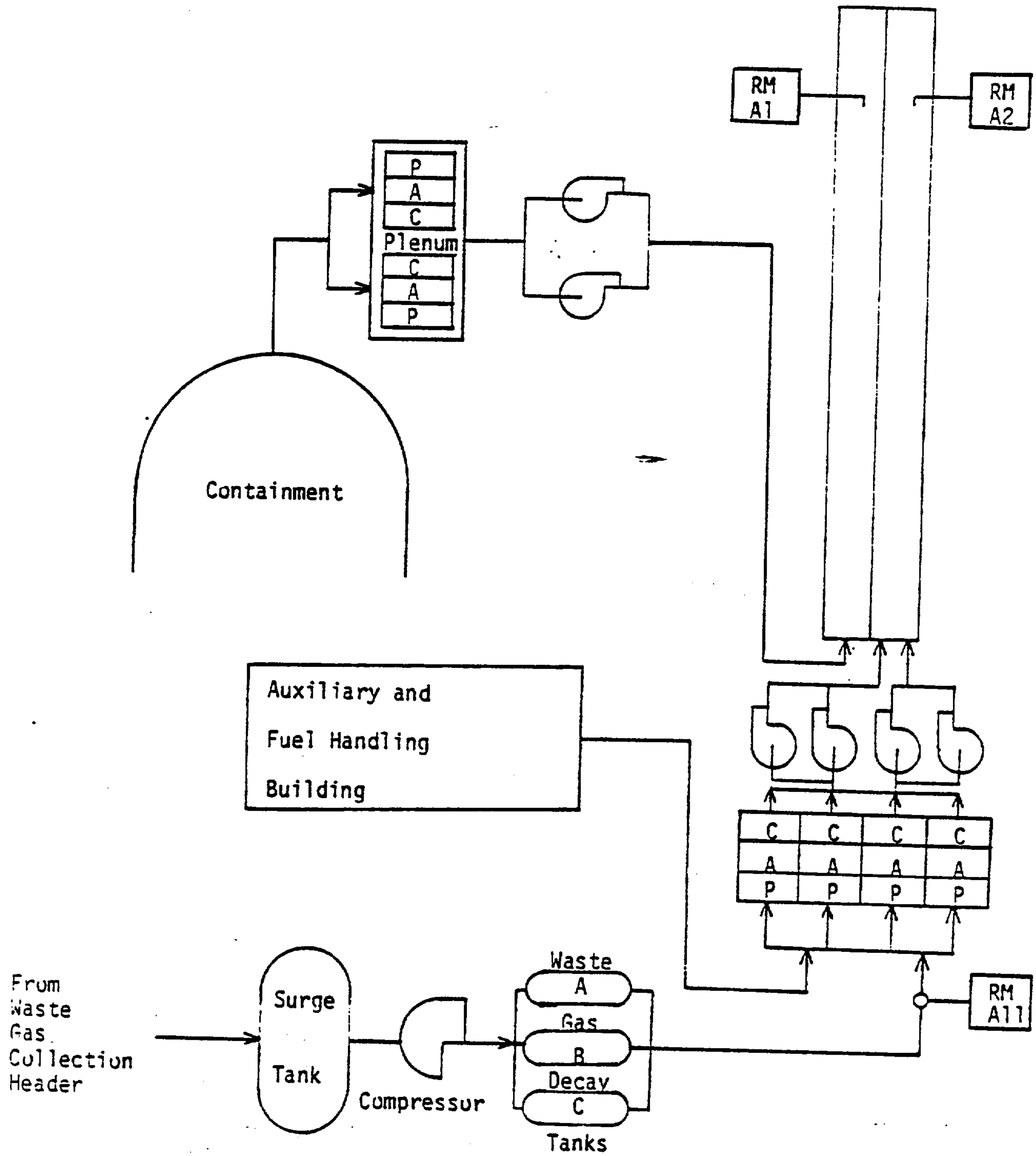


Figure 4. Gaseous Radwaste/Effluent Treatment System



Gaseous discharge from the WGDT's go to the auxiliary and fuel handling building ventilation. This discharge ventilation air is processed through a ventilation exhaust treatment system (which is a train of prefilters, HEPA filters, and charcoal adsorbers) prior to release out the plant vent. The containment atmosphere is processed through a ventilation exhaust treatment system (also a train of prefilters, HEPA filters, and charcoal adsorbers) prior to release out the containment vent. Gaseous discharge from the WGDT's is monitored by RM-A11 which will terminate the discharge upon reaching the alarm/trip setpoint. The facility vent is a single cylinder which is divided by a septum creating two separate discharge pathways (i.e., plant vent and containment vent). The plant vent is continuously monitored by RM-A2, which will also terminate the discharge of a WGDT upon reaching the alarm/trip setpoint of the noble gas monitor. The containment vent is continuously monitored by RM-A1 when ventilation air is being discharged. This monitor will terminate the discharge when the alarm/trip setpoint is reached. Figure 1 shows the gaseous discharge point. Figure 4 shows a schematic of the gaseous radwaste/effluent treatment system.

3.2 Radiological Effluent Technical Specifications

The following sub-sections describe the primary objectives of each section of the model RETS and a summary of the commitments of the Licensee's RETS. A cross reference between the numbering in the model RETS and the Licensee's RETS is contained in Table 1. The chronological sequence of the RETS review was described in the Plant-Specific Background, Section 1.3 of this report.

3.2.1 Effluent Instrumentation

The objective of the model RETS with regard to effluent instrumentation is to ensure that all significant liquid and gaseous radioactive effluents are monitored. The model RETS specify that all effluent monitors be operable with periodic surveillance and that alarm/trip setpoints be determined in order to ensure that offsite radioactive effluent concentrations do not exceed maximum permissible concentrations (MPC's) listed in 10 CFR Part 20.

The Licensee has provided radiation monitors for potential liquid or gaseous effluent lines. In addition, automatic isolation is provided for major effluent lines such as the liquid radwaste, the WGDT effluent, and the containment purge.

There are two radioactive liquid effluent release points at Crystal River Unit No. 3: the auxiliary building liquid radwaste effluent line and the secondary drain tank liquid effluent line. Both of these systems are monitored with adequate instrument surveillance being performed. The CCW and the SW systems are both closed loop systems. Potential leakage from these systems would be discharged through the secondary drain tank system.

All gaseous effluent releases at Crystal River Unit No. 3 are discharged through either the auxiliary building and fuel handling area vent or the reactor building purge exhaust vent. Both of these systems are monitored with adequate instrument surveillance being performed. Both will isolate the release pathway on alarm of the noble gas monitor. In addition, a noble gas monitor on the WDGT discharge line will isolate this release on alarm. This instrument also has adequate surveillance requirements.

The Licensee has stated that the concentration of radioactive material will be monitored "at all times," or "during releases" for batch releases. The setpoints at each release point are established to prevent exceeding the release concentrations for liquid releases or corresponding dose rates for gaseous releases of 10 CFR Part 20 in unrestricted areas. The setpoints for the liquid and gaseous effluent instrumentation will be determined according to the Offsite Dose Calculation Manual (ODCM).

The Licensee's RETS submittal on liquid and gaseous effluent monitoring instrumentation has satisfied the provisions and meets the intent of NUREG-0472.

3.2.2 Concentration and Dose Rates of Effluents

3.2.2.1 Liquid Effluent Concentration

The Licensee's RETS include a commitment to maintain the concentration of radioactive liquid effluents released from the site to the unrestricted areas to within 10 CFR Part 20 limits, and if the concentration of liquid effluents to the unrestricted area exceeds these limits, it will be restored without delay to a value equal to or less than the MPC values specified in 10 CFR Part 20. Both batch and continuous releases are sampled and analyzed periodically in accordance with a sampling and analysis program.

Therefore, the Licensee's RETS submittal on liquid effluent concentrations meets the intent of NUREG-0472.

3.2.2.2 Gaseous Effluent Dose Rate

The Licensee's RETS include a commitment to maintain the offsite gaseous dose rate from the site to areas at and beyond the site boundary to within 10 CFR Part 20 limits, and if the concentration of gaseous effluents exceeds these limits or the equivalent dose rate values, it will be restored without delay to a value equal to or less than these limits.

The radioactive gaseous waste sampling and analysis program provides adequate sampling and analysis of the discharges.

Therefore, the Licensee's RETS submittal on gaseous effluent dose rates meets the intent of NUREG-0472.

3.2.3 Offsite Doses from Effluents

The objectives of the model RETS with regard to offsite doses from effluents are to ensure that offsite doses are kept ALARA, are in compliance with dose specifications of NUREG-0472 and are in accordance with 10 CFR Part 50, Appendix I and 40 CFR Part 190.

The Licensee's RETS include commitments (a) to meet the quarterly and yearly dose criteria for liquid effluents and to use the ODCM methodology for determining the cumulative dose to individuals, (b) to maintain the air doses for noble gases in unrestricted areas to those specified in 10 CFR Part 50,

Appendix I, Section II.B., (c) to maintain the dose level to an individual from release of Iodine-131, tritium, and particulates with half-lives greater than eight days to meet the design objectives of 10 CFR Part 50, Appendix I, Section II.C, and (d) to limit the annual dose to the maximally exposed member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources to within the requirements of 40 CFR Part 190.

Therefore, the Licensee's RETS submittal on offsite doses from radioactive effluents meets the intent of NUREG-0472.

3.2.4 Effluent Treatment

The objectives of the model RETS with regard to effluent treatment are to ensure that the radioactive waste treatment systems are used to keep releases ALARA and to satisfy the provisions for Technical Specifications governing the maintenance and use of radwaste treatment equipment.

The Licensee's RETS include a commitment to use the liquid and gaseous radwaste treatment systems when the projected monthly doses exceed 25 percent of the annual dose design objectives and to use the ventilation exhaust treatment system if the projected monthly dose exceeds the limits prescribed in the model RETS. The projections are to be made at lease once per 31 days. The Licensee's RETS include a commitment to prepare a special report if radwaste treatment is required before release and the radwaste treatment equipment is inoperable.

Therefore, the Licensee's RETS submittal on effluent treatment meets the intent of NUREG-0472.

3.2.5 Tank Inventory Limits

The objective of the model RETS with regard to a curie limit on liquid-containing tanks is to ensure that in the event of a tank rupture, the concentrations in the nearest potable water supply and the nearest surface water supply in an unrestricted area would not exceed the limits of 10 CFR Part 20, Appendix B,

Table II. The objective of the model RETS with regard to a curie limit on gas-containing tanks is to ensure that in the event of an uncontrolled release of the tank's contents the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

The Licensee's RETS does not include a specification on curie limits for outside tanks containing liquids. The Crystal River site is adjacent to the Gulf of Mexico and any water flows are to the Gulf; consequently, any leakage would not effect drinking or surface water supplies. Thus, a technical specification is not required.

The Licensee's RETS state that each WGDT is limited to less than or equal to 39,000 curies (Xe-133 equivalent) which is less than the value corresponding to 0.5 rem at the exclusion area boundary. Surveillance on the WGDT's will be performed once per 24 hours during degassing periods and weekly at other times.

This surveillance is acceptable since the tank is sampled at the frequency required by the model RETS during degassing which is the time that has the greatest potential for exceeding the dose limit.

Therefore, the Licensee's RETS submittal on tank inventory limits meets the intent of NUREG-0472.

3.2.6 Explosive Gas Mixtures

The objective of the model RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in the waste gas system.

The Licensee's RETS include a commitment to maintain a safe concentration of oxygen in the WGDT's when the hydrogen concentration is equal to or greater than four percent. Flamability curves[34] show that if H_2 is less than or equal to four percent, oxygen can be at any concentration and a flamable mixture will not result. Both hydrogen and oxygen are monitored and the system is being modified to sample exclusively from the in-service WGDT.

Although the Licensee does not have the number of channels specified in the model RETS, the number of channels and the modification to sample only the in-service WGDT were accepted on an interim basis at the site meeting[25].

3.2.7 Solid Radwaste System

The objective of the model RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before it is shipped from the plant to the burial site to satisfy the requirements of 10 CFR Part 20, Section 20.301 and 10 CFR Part 71.[35]

The Licensee has committed to use the methods prescribed in a Process Control Program (PCP) to ensure that the requirements of 10 CFR Part 20 and 10 CFR Part 71 are met prior to shipment of radwaste from the site. The plant will use the Chem Nuclear Services waste solidification system which is currently being evaluated by the NRC.

Therefore, the Licensee's RETS submittal on solid radioactive waste meets the intent of NUREG-0472.

3.2.8 Radiological Environmental Monitoring Program

The objectives of the model RETS with regard to radiological environmental monitoring are to ensure that (a) an adequate full-area coverage environmental monitoring program exists, (b) there is an appropriate land use census, and (c) an acceptable interlaboratory comparison program exists. The monitoring program implements Section IV.B.2 of Appendix I to 10 CFRPart 50, the land use census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50, and the requirement for participation in an approved interlaboratory comparison program is provided to ensure that independent checks are performed as part of the quality assurance program for environmental monitoring to demonstrate that valid results are obtained for Section IV.B.2 of Appendix I to 10 CFR Part 50.

The environmental monitoring program obtains milk samples from a control location as there are no other milk samples available within the radii specified in the model RETS. If milk samples become available, they will be identified in the land use census and included in the environmental monitoring program. Since milk sampling is not performed, emphasis should be placed on vegetation sampling in the two highest D/Q sectors. There are no foods grown on land that is irrigated by water in which liquid plant wastes have been discharged.

The Licensee performs a gamma spectral I-131 analysis on green leafy vegetable samples taken semi-annually during harvest. The collection frequency should be increased to monthly when available and the sample should be any broad leaf vegetation at two different offsite locations of highest predicted annual average ground level D/Q.

The Licensee's RETS on a radiological environmental monitoring program have followed the model RETS and the Branch Technical Position on the subject issued November 1979,[29] as applicable to the site, and have provided an adequate number of sample locations for pathways identified except for the broad leafy vegetation. The Licensee's method of sample analysis and maintenance of the monitoring program satisfies the requirements of Appendix I, 10 CFR Part 50. The Licensee's RETS contain a land use census specification which requires the appropriate annual information for a PWR. The RETS also state that the Licensee will participate in an NRC approved interlaboratory comparison program.

Thus, the Licensee's RETS submittal for a radiological environmental program meets the intent of NUREG-0472 except for one item.

3.2.9 Audits and Reviews

The objective of the model RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted.

The Licensee's administrative structure identifies the Plant Review Committee (PRC) and the Nuclear General Review Committee (NGRC) as the two groups comparable to the Unit Review Group (URG) and the Company Nuclear Review and Audit Group (CNRAG), respectively.

The PRC is responsible for reviewing every unplanned release of radioactive material and any changes to the ODCM and PCP, as required by the model RETS.

The NGRC is responsible for reviewing the radiological environmental program and results thereof, the ODCM and implementing procedures, the PCP and implementing procedures, and the performance of activities required by the quality assurance (QA) program. These reviews are performed at the frequency required by the model RETS. These reviews were determined to be acceptable substitutes for the audit requirements described in the model RETS.

The PRC and NGRC encompass the total responsibility for reviews and audits specified in NUREG-0472.

3.2.10 Procedures and Records

The objective of the model RETS with regard to procedures is to ensure that written procedures be established, implemented and maintained for the PCP, the ODCM and the QA program for effluent and environmental monitoring. The objective of the model RETS with regard to records is to ensure that the documented records pertaining to the radiological environmental monitoring program are retained.

The Licensee's RETS include a commitment to establish, implement, and maintain written procedures for the PCP, ODCM, and QA programs. The Licensee's RETS state that the records of the radiological environmental monitoring program will be retained for the duration of the facility operating license.

Therefore, the Licensee's RETS submittal on procedures and records meets the intent of NUREG-0472.

3.2.11 Reports

The objective of the model RETS with regard to reporting requirements is to ensure that appropriate annual and semi-annual periodic reports and special reports are submitted to the NRC.

The Licensee's RETS include commitments to submit the following reports:

1. Annual Radiological Environmental Operating Report This report includes summaries, interpretations and analysis of trends of the results of the radiological environmental surveillance program. The report also includes the results of the land use census and results of the participation in the interlaboratory comparison program. The report will be submitted prior to March 1 of each year.

2. Semiannual Radioactive Effluent Release Report This report contains a summary of the quantities of radioactive liquid and gaseous effluents and is submitted within 60 days after January 1 and July 1 of each year. The report also includes a summary of solid waste shipped offsite, an assessment of offsite doses, doses to individuals due to their activities inside the site boundary, doses to the hypothetical worst case individual (including direct radiation), the prescribed meterological data, and a list of unplanned releases. A listing of new locations required by the land use census as well as any changes to ODCM, PCP and the radioactive waste treatment system is included.

3. Special Reports The Licensee's RETS include a commitment to file a special report within 30 days under the following conditions:

- Exceeding the liquid effluent dose limits according to Specification 3.11.1.2.

- Exceeding the gaseous effluent dose limits according to Specifications 3.11.2.2 and 3.11.2.3

- Exceeding the total dose limits according to Specification 3.11.3.

- Exceeding the reporting levels for the radioactivity measured in environmental sampling program Specification 3.12.1.1.

- When radioactive liquid or gaseous effluents require treatment before discharge and the waste treatment equipment is inoperable as specified in 3.7.13.2 and 3.17.13.3.

Therefore, the Licensee's RETS submittal on reports meets the intent of NUREG-0472.

3.2.12 Other Administrative Controls

An objective of the model RETS in the administrative controls section is to ensure that any changes to the PCP and ODCM and major changes to the radioactive waste treatment systems are reported to the NRC. Such changes shall be reviewed and accepted by the URG before implementation.

The Licensee's RETS state that the aforementioned changes will be reported to the NRC after review and acceptance by the PRC.

Therefore, the Licensee's RETS submittal for these administrative controls meets the intent of NUREG-0472.

3.3 OFFSITE DOSE CALCULATION MANUAL

As specified in NUREG-0472, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and maintain the operability of the effluent system. As a minimum, the ODCM should provide equations and methodology for the following topics:

- alarm and trip setpoints for effluent instrumentation
- liquid effluent concentration in unrestricted areas
- gaseous effluent dose rate or concentrations at or beyond the site boundary

- liquid and gaseous effluent dose contributions
- total dose compliance, including direct shine
- liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams, consistent with the systems being used at the station, defining the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. A description and the location of samples in support of the environmental monitoring program are also needed in the ODCM.

3.3.1 Evaluation

The Licensee's ODCM satisfies the equation in the addendum of NUREG-0133 to determine the alarm and trip setpoints for the liquid effluent monitors. This assures that the alarm and trip actions will occur prior to exceeding the 10 CFR Part 20, Appendix B, Table II values at the discharge point to the unrestricted area.

The alarm and trip setpoints for the gaseous effluent monitors are calculated to assure that alarm and trip actions will occur prior to exceeding the limits set in 10 CFR Part 20 for annual dose rate to unrestricted areas. The Licensee uses equations similar to those contained in NUREG-0133 with the dose rate values identified in NUREG-0472.

The Licensee's ODCM contains the methods and calculational relationships that are used to compare the radioactivity concentrations at the point of release to the 10 CFR Part 20 limits prior to the release and after the release.

The Licensee's ODCM states that noble gas discharges are assured to be within the NUREG-0472 dose rate limits by correctly determining the setpoints for the noble gas monitors. Therefore, additional sampling and analysis are not required. The dose rate due to the release of I-131, tritium, and particulates with half-lives greater than eight days is assured to be within the NUREG-0472 limit of 1500 mrem per year to a child via the inhalation pathway by calculating the dose rate due to the actual release using the highest calculated annual average dispersion parameter X/Q used for estimating dose to an individual.

The Licensee's ODCM demonstrates compliance with 10 CFR Part 50, Appendix I by calculating the monthly dose commitments for liquid and gaseous effluents at least once per 31 days. The calculated cumulative values are compared to the quarterly and annual limits to demonstrate compliance. The doses due to liquid releases are calculated using the adult ingestion pathway since drinking water is not affected by plant discharges. The doses due to gaseous releases are calculated using the inhalation, ingestion, and ground plane pathways.

The Licensee's ODCM contains the method used to project the monthly doses due to anticipated liquid and gaseous releases. The dose projection is made at least once every 31 days. If the projected values exceed 25 percent of the annual dose prorated monthly the radwaste treatment system must be operated.

The Licensee's ODCM contains a description and map of the sample locations for the environmental monitoring program. In addition, the ODCM contains block diagram descriptions of the flow paths and treatment systems for the liquid and gaseous releases.

The Licensee's ODCM for Crystal River Unit No. 3 is in compliance with the NRC requirements and uses methods consistent with the methodology and guidance prescribed in NUREG-0133.

3.4 PROCESS CONTROL PROGRAM

NUREG-0472 specifies that the Licensee develop a PCP to ensure that the processing and packaging of solid radioactive wastes will be accomplished in compliance with 10 CFR Part 20, 10 CFR Part 71, and other federal and state regulations or requirements governing the offsite disposal of the low-level radioactive waste.

The PCP is not intended to contain a set of detailed procedures; rather, it is the source of basic criteria for the detailed procedures to be developed by the Licensee. The criteria used for the PCP are to address only today's requirements. The uncertainty about PCP requirements results from the recent promulgation of 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." The NRC staff's technical positions are presently being developed by the Division of Waste Management[33].

3.4.1 Evaluation

The Licensee did not submit a PCP for review. A PCP will be submitted to the NRC for review and approval.

4.0 CONCLUSIONS

The Licensee's proposed RETS and ODCM were reviewed and evaluated and the following conclusions were reached:

- The Licensee's proposed RETS for the Crystal River Unit No. 3 Nuclear Generating Plant, submitted January 17, 1983, meets the intent of the NRC staff's "Standard Radiological Effluent Technical Specifications," NUREG-0472 except for the item in the radiological environmental monitoring program.

- The Licensee's ODCM, submitted May 1983 uses documented and approved methods that are applicable to Crystal River Unit No. 3 and are consistent with the criteria of NUREG-0133. Discrepancies are identified in the cover letter for this report.

- The Licensee did not submit a PCP for review. A PCP will be submitted to the NRC for review and approval.

A correspondence between (a) NUREG-0472, (b) the Licensee's current RETS, and (c) the Licensee's proposed RETS is shown in Table 1. A more detailed explanation of how each Specification in the Licensee's RETS meets the intent of NUREG-0472 is contained in Appendix A.

TABLE 1. CORRESPONDENCE OF PROVISIONS OF NUREG-0472, THE LICENSEE'S CURRENT TECHNICAL SPECIFICATIONS AND THE LICENSEE'S PROPOSAL FOR CRYSTAL RIVER UNIT NO. 3

RETS Requirement	NUREG-0472[1]	Current Technical Specification[2] Appendix B (Section)	Licensee Proposal (Section)
Effluent Instrumentation	3.3.3.10 3.3.3.11	2.4.1 2.4.2	3.3.3.8 3.3.3.9
Concentrations	3.11.1.1 3.11.2.1	2.4.1 2.4.2	3.11.1.1 3.11.2.1
Offsite Doses	3.11.1.2 3.11.2.2 3.11.2.3 3.11.4	2.4 2.4	3.11.1.2 3.11.2.2 3.11.2.3 3.11.3
Effluent Treatment	3.11.1.3 3.11.2.4		3.7.13.2 3.7.13.3
Tank Inventory Limits	3.11.1.4 3.11.2.6	 2.4.2	--- 3.7.13.1
Explosive Gas Mixtures	3.11.2.5		3.7.13.5
Solid Radwaste	3.11.3	2.4.3	3.7.13.4
Environmental Monitoring	3.12.1	3.2	3.12.1.1
Land Use Census	3.12.2		3.12.1.2
Interlaboratory Comparisons	3.12.3		3.12.1.3
Audits and Reviews	6.5.1 6.5.2	5.3	6.5.1 6.5.2
Procedures and Records	6.8,6.10	5.5	6.8.1
Reports	6.9	5.6.1 5.6.1	6.9.1.5.c 6.9.1.5.d
Other Administrative Controls	6.13,6.14,6.15		6.14,6.15,6.16

1 Section number sequence is according to NUREG-0472, Rev. 3, Draft 7'.

2 Being Revised or Deleted

5. REFERENCES

1. United States Nuclear Regulatory Commission, Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors, NUREG-0472, Revision 2, July 1979.

2. United States Office of The Federal Register, Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

3. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 50, Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 50, Appendix I, Section V.B., "Effective Dates."

5. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation."

6. United States Office of the Federal Register, Title 40, Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

7. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Critera for Nuclear Power Plants."

8. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

9. United States Nuclear Regulatory Commission, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, NUREG-0133, October 1978.

10. United States Nuclear Regulatory Commission, Standard Radiological Effluent Technical Specifications for Boiling Water Reactors, NUREG-0473, Revision 2, July 1979.

11. C. A. Willis and F. J. Congel, "Summary of Draft Contractor Guidance of RETS," AIF Environmental Subcommittee Meeting, Washington, D.C., May 19, 1982.

12. F. J. Congel, memo to RAB Staff (NRC), Interim Changes in the Model RETS, August 9, 1982.

13. United States Nuclear Regulatory Commission, Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors, NUREG-0472, Revision 3, Draft 7', September 1982.

14. NUS Corporation, letter of transmittal, Appendix I Analysis for Crystal River Nuclear Unit, NUS-1721 (Revision 1), May 28, 1976.

15. R. Vollmer, letter of transmittal, Supplement No. 2 to the Safety Evaluation Report for Crystal River Unit No. 3 With Respect to Appendix I to 10 CFR Part 50, August 30, 1976.

16. R. M. Bright, letter of transmittal, Technical Specification Change Request No. 36, February 13, 1979.

17. F. B. Simpson, letter of transmittal, Transmittal of Questions for Crystal River RETS Review, SIM-28-81, November 23, 1981.

18. W. Serrano, letter of transmittal, Transmittal of Questions for Crystal River RETS Review, Serr-11-82, September 14, 1982.

19. E. W. Ford (NUS) and S. W. Duce (EG&G), telephone conference, December 13, 1982.

20. C. A. Willis (NRC), S. W. Duce (EG&G), and W. Serrano (EG&G), telephone conference, December 13, 1982.

21. G. R. Westafer, letter of transmittal, Technical Specification Change Request No. 36, January 17, 1983.

22. G. R. Westafer, letter of transmittal, Offsite Dose Calculation Manual, May 1983.

23. United States Office of The Federal Register, Title 10, Code of Federal Regulations, Part 20, Appendix B, "Concentrations in Air and Water Above Natural Background."

24. C. A. Willis, letter to F. B. Simpson (summarizing changes to RETS requirements following meeting with Atomic Industrial Forum), November 20, 1981.

25. W. E. Kreger (NRC), memo to R. J. Mattson (NRC), Plans for Dealing with The Explosive Gas Issue in Implementing The Radiological Effluent Technical Specifications (RETS), December 14, 1981.

26. C. A. Willis and F. J. Congel, "Status of NRC Radiological Effluent Technical Specification Activities," Atomic Industrial Forum Conference on HEPA and Nuclear Regulations, October 4-7, 1981.

27. C. A. Willis, memo to P. C. Wagner, Plans for Implementing Radiological Effluent Technical Specifications for Operating Reactors, November 4, 1981.

28. W. P. Gammill (NRC), memo to P. C. Wagner (NRC), Current Position on Radiological Effluent Technical Specifications (RETS) including Explosive Gas Controls, October 7, 1981.

29. United States Nuclear Regulatory Commission, Radiological Assessment Branch Technical Position, An Acceptable Radiological Environmental Monitoring Program, November 1979.

30. United States Nuclear Regulatory Commission, Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190), NUREG-0543, February 1980.

31. United States Nuclear Regulatory Commission, Branch Technical Position, General Contents of the Offsite Dose Calculation Manual, Revision 1, February 8, 1979.

32. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, " Revision 1, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, October 1977.

33. C. A. Willis, letter of transmittal, Guidance on the Review of the Process Control Programs (PCP), January 3, 1983.

34. C. A. Willis (NRC) and J. S. Boegli (NRC), memo to W. P. Gammill (NRC), Tech Spec on Control of Explosive Gas Mixtures in the Waste Gas System at TMI-1, July 22, 1981.

35. United States Office of the Federal Register, Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Material For Transport and Transportation of Radioactive Material Under Certain Conditions."