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

DO NOT REMOVE

December 24, 1984

Docket No. 50-298

Posted
Amdt. 89
to DPR-46
(See Correction letter
of 6-20-86)

Mr. J. M. Pilant, Technical
Staff Manager
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated March 7, 1984, as supplemented by your submittals dated April 10, 1984, July 19, 1984 and November 15, 1984.

The amendment revises the Technical Specifications to incorporate the Radiological Effluent Technical Specifications (RETS) for Cooper Nuclear Station.

The staff has reviewed the request in your letter dated November 15, 1984 that the effective date for implementation of the majority of the RETS sections be delayed until July 1, 1986 to permit procurement and testing of software which is needed to perform computations as required by the RETS. Accordingly, the enclosed license amendment is effective July 1, 1986 except for Sections 3.21.F and 4.21.F and Tables 3.21.F.1 and 3.21.F.2 and associated notes which are to be effective January 1, 1985.

A copy of our Safety Evaluation related to the RETS is enclosed. This Safety Evaluation also includes the results of our review of your "Offsite Dose Assessment Manual" (ODAM) submitted by letter dated March 7, 1984 in response to our request for an "Offsite Dose Calculation Manual" (ODCM) as a reference document for the RETS. Our review of your ODAM indicates that the ODAM generally uses documented and approved methods that are consistent with the methodology and guidelines in NUREG-0133 with a few minor discrepancies. Discrepancies in the ODAM found in our review will be transmitted shortly for your resolution prior to the July 1, 1986 implementation of those RETS sections which reference the ODAM.

Sincerely,

Ernest D. Sylvester

Ernest D. Sylvester, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures and cc:
See next page

Enclosures:
1. Amendment No. 89 to
   License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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Cooper Nuclear Station

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated March 7, 1984, as supplemented April 10, 1984, July 19, 1984, and November 15, 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 licensee 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-46 is hereby amended to read as follows:

(2) Technical Specification

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

3. This license amendment is effective as of July 1, 1986 except for the following sections of the Radiological Effluent Technical Specifications which are effective January 1, 1985:

   a. Section 3.21.F,

   b. Section 4.21.F, and

   c. Tables 3.21.F.1 and 3.21.F.2 and associated notes.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 24, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

A. On January 1, 1985, revise the Appendix A Technical Specifications by adding the pages noted below:

Insert
iii
216n
216o
216p
216q
216r
216s
216t

B. On July 1, 1986, revise the Appendix A Technical Specifications as follows.

Remove	Insert
i	i
iii	iii
iv	iv
-	v
1	1
2	2
3	3
4	4
5	5
5A	5a
-	5b
48	48
63	63
-	63a
78	78
81	81
87	87
90	90
216n through 216t	216n through 216z
-	216a1 thorugh 216a29
225a	225a
226	226
231	231
-	231a
-	231b

Remove	Insert
-	231c
235	235
-	235a
-	235b
-	235c

C. On July 1, 1986, delete Appendix B in its entirety.

TABLE OF CONTENTS (Cont'd.)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS	Page No.
12 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES	4.12	215 - 215f
A. Main Control Room Ventilation	A	215
B. Reactor Building Closed Cooling Water System	B	215b
C. Service Water System	C	215c
D. Battery Room Vent	D	215c
3.13 RIVER LEVEL	4.13	216
3.14 FIRE DETECTION SYSTEM	4.14	216b
3.15 FIRE SUPPRESSION WATER SYSTEM	4.15	216b
3.16 SPRAY AND/OR SPRINKLER SYSTEM (FIRE PROTECTION)	4.16	216e
3.17 CARBON DIOXIDE SYSTEM	4.17	216f
3.18 FIRE HOSE STATIONS	4.18	216g
3.19 FIRE BARRIER PENETRATION FIRE SEALS	4.19	216h
3.20 YARD FIRE HYDRANT AND HYDRANT HOSE HOUSE	4.20	216i
3.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS	4.21	216n
5.0 MAJOR DESIGN FEATURES		
5.1 Site Features		217
5.2 Reactor		217
5.3 Reactor Vessel		217
5.4 Containment		217
5.5 Fuel Storage		218
5.6 Seismic Design		218
5.7 Barge Traffic		218
6.0 ADMINISTRATIVE CONTROLS		
6.1 Organization		219
6.1.1 Responsibility		219
6.1.2 Offsite		219
6.1.3 Plant Staff - Shift Complement		219
6.1.4 Plant Staff - Qualifications		219a
6.2 Review and Audit		220
6.2.1.A Station Operations Review Committee (SORC)		220
A.1 Membership		220
A.2 Meeting Frequency		220
A.3 Quorum		220
A.4 Responsibilities		220
A.5 Authority		221
A.6 Records		221
A.7 Procedures		222

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS	4.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS
Items A through E to be issued July 1, 1986	Items A through E to be issued July 1, 1986
F. Monitoring Program	F. Monitoring Program
Applicability: At all times.	
Specification:	
1. As a minimum the radiological environmental monitoring program shall be conducted as specified in Table 3.21.F.1. Analytical techniques used shall be such that the detection capabilities in Table 3.21.F.2 are achieved.	1. Radiological environmental samples shall be collected and analyzed as specified in Table 3.21.F.1.
2. In the event the radiological environmental monitoring program is not conducted as specified in Table 3.21.F.1, prepare and submit to the Commission in the Annual Operating Report the reasons for not conducting the program in accordance with Table 3.21.F.1 and the plans for preventing a recurrence.	2. A land use census shall be conducted annually and shall identify the location of the nearest garden that is greater than 500 square feet in area and that yields edible leafy vegetables, the location of the nearest milk animal, and the location of the nearest resident in each of the 16 meteorological sectors within three miles of the Station. The land use census shall be conducted at least once per 12 months.
3. To be issued on July 1, 1986	3. The results of sample analyses performed shall be summarized in the Annual Radiological Environmental Report.
	4. The results of the land use census shall be included in the Annual Radiological Environmental Report.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.F (Cont'd)	4.21 (Cont'd)

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

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

5. To be issued on July 1, 1986.

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

7. The provisions of Definition J are not applicable.

TABLE 3.21.F.1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

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

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

Exposure Pathway and/or Sample	Number of Sample Stations	Sampling and Collection Frequency	Type and Frequency of Analysis
4. Ingestion			
a. Milk (Nearest Producer)	At least one location	At least once per 15 days during Peak Pasture Period[a]; at least once per 31 days at other times.	Gamma isotopic[b] and I-131 analysis of each sample.
b. Milk (Other Producers)	At least 2 locations	At least once per 92 days.	Gamma isotopic[b] and I-131 analysis of each sample.
c. Fish	At least 2 locations	Two times per year (once in the summer and once in the fall). Attempt to include the following: 1. Bottom feeding species 2. Middle-Top feeding species	Gamma isotopic[b] analysis on edible portions.
d. Food Products	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic[b] and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic[b] and I-131 analysis.

NOTES FOR TABLE 3.21.F.1

a. DELETED

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

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

Any radionuclide detected, i.e., having a measured concentration greater than the LLD, whether or not it is one of the 19 nuclides listed above, shall be regarded as present in the sample.

c. Thermoluminescent Dosimeters (TLD) is a single phosphore. Two or more phosphores in one package are considered to be two or more dosimeters.

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

TABLE 3.21.F.2

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

	Lower Limit of Detection (LLD)[a]					
Anaysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}				
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	9[b]	7×10^{-2}		1	60	
^{134}Cs	15	5×10^{-2}	130	15	60	150
^{137}Cs	18	6×10^{-2}	150	18	80	180
^{140}Ba	60			60		
^{140}La	15			15		

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

NOTES FOR TABLE 3.21.F.2

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

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

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

Where

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

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

E is the counting efficiency (as counts per transformation)

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

2.22 is the number of transformation per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

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

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

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

b. LLD for drinking water.

EFFECTIVE JULY 1, 1986

TABLE OF CONTENTS

			Page No.
1.0	DEFINITIONS		1 - 5a
	SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS	
1.1	FUEL CLADDING INTEGRITY	2.1	6 - 22
1.2	REACTOR COOLANT SYSTEM INTEGRITY	2.2	23 - 26
	LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS	
3.1	REACTOR PROTECTION SYSTEM	4.1	27 - 46
3.2	PROTECTIVE INSTRUMENTATION	4.2	47 - 92
	A. Primary Containment Isolation Functions		47
	B. Core and Containment Cooling Systems Initiation and Control (CS, LPCI, HPCI, RCIC, ADS)		47
	C. Control Rod Block Actuation		47
	D. Radiation Monitoring Systems - Isolation and Initiation Functions		48
	1. Steam Jet Air Ejector Off-Gas System		48
	2. Reactor Building Isolation and Standby Gas Treatment Initiation		48
	3. Liquid Radwaste Discharge Isolation		48
	4. Main Control Room Ventilation		48
	5. Mechanical Vacuum Pump Isolation		49
	E. Drywell Leak Detection		49
	F. Primary Containment Surveillance Information Readouts		49
	G. Recirculation Pump Trip		49
3.3	REACTIVITY CONTROL	4.3	93 - 106
	A. Reactivity Limitations	A	93
	B. Control Rods	B	94
	C. Scram Insertion Times	C	97
	D. Reactivity Anomalies	D	98
	E. Restrictions	E	98
	F. Recirculation Pumps	F	98
	G. Scram Discharge Volume	G	98a
3.4	STANDBY LIQUID CONTROL SYSTEM	4.4	107 - 113
	A. Normal Operation	A	107
	B. Operation with Inoperable Components	B	108
	C. Sodium Pentaborate Solution	C	108

Amendment Nos. 13, 32, 83,89
Collection letter of 6-20-86

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS	Page No.
3.12 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES	4.12	215 - 215f
A. Main Control Room Ventilation	A	215
B. Reactor Building Closed Cooling Water System	B	215b
C. Service Water System	C	215c
D. Battery Room Vent	D	215c
3.13 RIVER LEVEL	4.13	216
3.14 FIRE DETECTION SYSTEM	4.14	216b
3.15 FIRE SUPPRESSION WATER SYSTEM	4.15	216b
3.16 SPRAY AND/OR SPRINKLER SYSTEM (FIRE PROTECTION)	4.16	216e
3.17 CARBON DIOXIDE AND HALON SYSTEMS	4.17	216f
3.18 FIRE HOSE STATIONS	4.18	216g
3.19 FIRE BARRIER PENETRATION FIRE SEALS	4.19	216h
3.20 YARD FIRE HYDRANT AND HYDRANT HOSE HOUSE	4.20	216i
3.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS	4.21	216n
A. Instrumentation		216n
B. Liquid Effluents		216x
C. Gaseous Effluents		216a4
D. Effluent Dose Liquid/Gaseous		216a11
E. Solid Radioactive Waste		216a12
F. Monitoring Program		216a13
G. Interlaboratory Comparison Program		216a20
3.22 SPECIAL TESTS/EXCEPTIONS	4.22	216b1
A. Shutdown Margin Demonstration		216b1
B. Training Startup		216b2
C. Physics Tests		216b3
D. Startup Test Program		216b3
5.0 MAJOR DESIGN FEATURES		
5.1 Site Features		217
5.2 Reactor		217
5.3 Reactor Vessel		217
5.4 Containment		217
5.5 Fuel Storage		218
5.6 Seismic Design		218
5.7 Barge Traffic		218
6.0 ADMINISTRATIVE CONTROLS		
6.1 Organization		219
6.1.1 Responsibility		219
6.1.2 Offsite		219
6.1.3 Plant Staff - Shift Complement		219
6.1.4 Plant Staff - Qualifications		219a

TABLE OF CONTENTS (Cont'd)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS	Page No.
6.2 Review and Audit		220
6.2.1.A Station Operations Review Committee (SORC)		220
A.1 Membership		220
A.2 Meeting Frequency		220
A.3 Quorum		220
A.4 Responsibilities		220
A.5 Authority		221
A.6 Records		221
A.7 Procedures		222
6.2.1.B NPPD Safety Review and Audit Board (SRAB)		222
B.1 Membership		223
B.2 Meeting Frequency		223
B.3 Quorum		223
B.4 Review		223
B.5 Authority		224
B.6 Records		225
B.7 Audits		225
6.3 Procedures and Programs		226
6.3.1 Introduction		226
6.3.2 Procedures		226
6.3.3 Maintenance and Test Procedures		226
6.3.4 Radiation Control Procedures		226
.A High Radiation Areas		226a
6.3.5 Temporary Changes		226a
6.3.6 Exercise of Procedures		226a
6.3.7 Programs		226a
.A Systems Integrity Monitoring Program		226a
.B Iodine Monitoring Program		226a
.C Environmental Qualification Program		226a
.D Post-Accident Sampling System		227
6.4 Record Retention		228
6.4.1 5 year retention		228
6.4.2 Life retention		228
6.4.3 2 year retention		229
6.5 Station Reporting Requirements		230
6.5.1 Routine Reports		230
.A Introduction		230
.B Startup Report		230
.C Annual Reports		230
.D Monthly Operating Report		231
.E Annual Radiological Environmental Report		231
.F Semiannual Radioactive Material Release Report		232

Collection letter of 6-20-86

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS	Page No.
6.5.2 Reportable Events		234
6.5.3 Unique Reporting Requirements		235
6.6 Process Control Program		235b
6.7 Offsite Dose Assessment Manual (ODAM)		235b
6.8 Major Changes to Radioactive Waste Treatment Systems		235c

Amendment No. 89
Collection letter of 6-20-86

1.0 DEFINITIONS

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

A. Thermal Parameters

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

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

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

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

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

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

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

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

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

E.A Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose if inhaled by an adult as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose equivalent I-131 concentration is calculated by: equiv. I-131 = (I-131) + 0.0096 (I-132) + 0.18 (I-133) + 0.0025 (I-134) + 0.037 (I-135).

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

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

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

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

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

I. Instrumentation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, and in at least COLD SHUTDOWN within the following 30 hours. This specification is not applicable in the cold condition or the refueling mode.

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

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

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

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

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

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

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

L.B Offsite Dose Assessment Manual (ODAM) - An OFFSITE DOSE ASSESSMENT MANUAL (ODAM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints, and describes the Environmental Radiation Monitoring Program.

M. Operable - Operability - Operating

1. Operable - Operability - A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified functions(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources (except as specified in Section 1.0.J and 3.9), cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its functions(s) are also capable of performing their related support functions(s).

2. Operating - Operating means a system, subsystem, train, component, or device is performing its intended function in its required manner.

N. Deleted

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

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

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment, and which are not required to be open during accident conditions, are closed.
2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

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

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

Q. Rated Power - Rated power refers to operation at a reactor power of 2381 megawatts thermal. This is also termed 100% power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 104.4% of rated power (105% of rated steam flow), which corresponds to 2486 megawatts thermal.

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

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

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

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

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

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

2. The standby gas treatment system is operable.

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

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

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

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

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

X. Spiral Reload - Pertains to the spiral reloading of the core with fuel, at least 50% of which has previously accumulated a minimum exposure of 1000 MWD/T.

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

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

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

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

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

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

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

Z.B Offsite - Offsite means outside of the exclusion area as defined in 10CFR Part 100.3. The exclusion area boundary around Cooper Station is defined in Figure 1.1 and may also be referred to as the Site Boundary.

Z.C Member of the Public - A Member of the Public is a person who is not occupationally associated with NPPD and who does not normally frequent the Cooper Station. The category does not include contractors, contractor employees, vendors, or persons who enter the site to make deliveries, to service equipment, or work on the site.

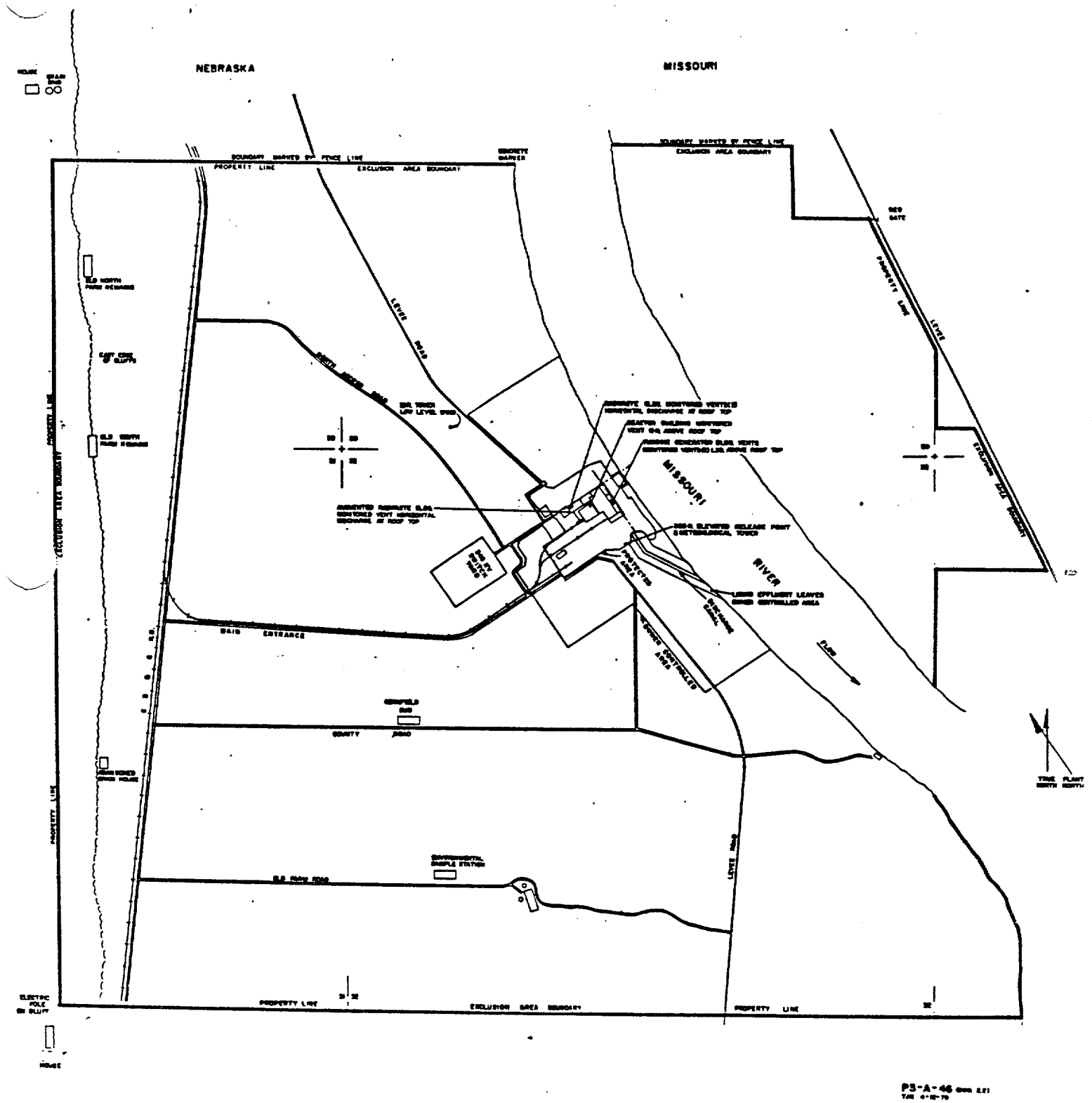

Figure 1.1

Exclusion Area Boundary
For
Gaseous and Liquid Effluents

Amendment No. 89
Correction letter of 6-20-86 56

3.1 BASES (Cont.d)

ence paragraph VII.5.7 FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRM's and APRM's provide adequate coverage in the startup and intermediate range.

The requirement to have the scram functions indicated in Table 3.1.1 operable in the Refuel mode assures that shifting to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

Turbine stop valve scram occurs at 10% of valve closure. Below 30% of the rated turbine first stage pressure, the scram signal due to the turbine stop valve closure may be bypassed because the flux and pressure scrams are adequate to protect the reactor. The actual scram bypass setpoint, however, is implemented at ≤25% of rated turbine first stage pressure (or 179 psig) to compensate for possible turbine trips during bypass valve testing. During bypass valve testing, the first stage pressure is reduced due to flow through the bypass valves while reactor power is unchanged.

Turbine control valves fast closure initiates a scram based on pressure switches sensing Electro-Hydraulic Control (EHC) system oil pressure. The switches are located on the Control Valve Emergency Trip oil header, and detects the loss of oil to hold the valves open.

This scram signal is also bypassed when turbine first stage pressure is less than 179 psig.

The requirements that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the Startup and Refuel modes assures that

4.1 BASES (Cont.d)

full scale flow signal will be sent to half of the APRM's resulting in a rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less that 0.4%/month; i.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

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

COOPER NUCLEAR STATION
TABLE 3.2.D
RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

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

Amendment Nos. 80, 89
Correction letter 90 10-20-86

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

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

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

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

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

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

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

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

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

4. Minimum number of channels operable shall be one during mechanical vacuum pump operation.

COOPER NUCLEAR STATION
TABLE 4.2.D
MINIMUM TEST AND CALIBRATION FREQUENCIES FOR RADIATION MONITORING SYSTEMS

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

Amendment Nos. 4, 89
Collection Date 06-20-86

1. Initially once every month until exposure (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 (after NRC approval). The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of CNS.

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

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

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

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

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

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

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

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

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

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

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

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip setting of 1.0 ci/sec (prior to 30 min. delay) provides an improved capability to detect fuel pin cladding failures to allow prevention of serious degradation of fuel pin cladding integrity which might result from plant operation with a misoriented or misloaded fuel assembly. This limit is more restrictive than 0.39 ci/sec noble gas release rate at the air ejectors (after 30 min. delay) which was used as the source term for an accident analysis of the augmented off-gas system. Using the .39 ci/sec source term, the maximum off-site total body dose would be less than the .5 rem limit.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

Two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. The trip is actuated by one hi-hi or two downscale indications.

Trip settings of <100 mr/hr for the monitors in the ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3. Liquid Radwaste Discharge Isolation

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

4. Main Control Room Ventilation

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

5. Mechanical Vacuum Pump

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

E. Drywell Leak Detection

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

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval and

2. more than one channel should not be bypassed for testing at any one time.

The bases for the radiation monitors are contained in the section denoted Environmental/Radiological Effluents.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS

A. Instrumentation

1. Liquid Effluent Monitoring

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

Specification:

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

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

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

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

e. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.5.2 are not applicable.

4.21 ENVIRONMENTAL/RADIOLOGICAL EFFLUENTS

A. Instrumentation

1. Liquid Effluent Monitoring

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

b. Radioactive liquid effluent monitor alarm and trip setpoints shall be determined in the manner described in the ODAM.

TABLE 3.21.A.1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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

Amendment No. 89
Collection letter of 6-30-86

NOTES FOR TABLE 3.21.A.1

* During releases via this pathway.

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

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

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

TABLE 4.21.A.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

Amendment No. 89
Revision Date 6-30-86

NOTES FOR TABLE 4.21.A.1

* During releases via this pathway.

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

Technical Specifications

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

2. Circuit failure.

3. Instrument indicates a downscale failure.

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

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

2. Circuit failure.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) The CHANNEL CALIBRATION shall be performed according to established station calibration procedures.

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

FREQUENCY NOTATION:

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.A (Cont'd)	4.21.A (Cont'd)
2. Gaseous Effluent Monitoring	2. Gaseous Effluent Monitoring
Applicability: As shown in Table 3.21.A.2.	a. The setpoints shall be determined in accordance with the method described in the ODAM.
Specification:	b. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.21.A.2.
a. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.21.A.2 shall be OPERABLE with their alarm setpoints set to ensure that the limits of Specification 3.21.C.1 are not exceeded.	
b. With a radioactive gaseous effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of 3.21.C.1 are met, reset without delay to comply with Specification 3.21.A.2.a, declare the channel inoperable; immediately suspend release; or change the setpoint so it is acceptably conservative.	
c. With less than the minimum required number of radioactive gaseous effluent monitoring instrumentation channels operable, take the ACTION shown in Table 3.21.A.2.	
d. If the minimum number of instrument channels are not returned to OPERABLE status within 31 days, in lieu of any other report, explain in the next Semiannual Radioactive Effluent Report why the instrument was not repaired in a timely manner.	
e. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.5.2 are not applicable.	

Amendment No. 89
Correction letter of 6-20-86

Amendment No. 89
Collection until 6-30-86

TABLE 3.21.A.2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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

NOTES FOR TABLE 3.21.A.2

Channels shall be operable and in service as indicated except that outages are permitted for the purpose of required tests, checks, calibrations, or for maintenance.

* During releases via this pathway.

** During Augmented Offgas Treatment System Operation.

*** During operation of the Steam Jet Air Ejector.

**** Elevated Release Point (ERP) Monitoring System, Radwaste Building Ventilation Monitoring System, and Turbine Building Ventilation Monitoring System.

Grand Gulf Nuclear Station
Technical Specifications

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

1. The offgas delay system is not bypassed; and

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

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

ACTION 26 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 24 hours.

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

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

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

Amendment No. 89
Correction letter 86 6-20-86

TABLE 4.21.A.2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Steam Jet Air Ejector				
a. Noble Gas Activity Monitor	D***	M	R(3)	Q(2) R(1)
b. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
2. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor (2% Monitor)	D**	NA	Q(4)	M
3. Reactor Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor (KAMAN)	D*	M	R(3)	Q(5)
b. Iodine Sampler Cartridge	W*	NA	NA	NA
c. Particulate Sampler Filter	W*	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
e. Sampler Flow Rate Measuring Device	D*	NA	R	Q
f. Isolation Monitor (GE)	D*	Q	R(3)	R(1)
4. (****)				
a. Noble Gas Activity Monitor (KAMAN)	D*	M	R(3)	Q(5)
b. Iodine Sampler	W*	NA	NA	NA
c. Particulate Sampler	W*	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	D*	NA	R	Q
e. Sampler Flow Rate Monitor	D*	NA	R	Q

NOTES FOR TABLE 4.21.A.2

* During releases via this pathway.
** During augmented offgas treatment system operation.
*** During operation of the Steam Jet Air Ejector.
**** Elevated Release Point (ERP) Monitoring System, Radwaste Ventilation Monitoring System, and Turbine Building Ventilation Monitoring System.

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

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

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

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

(3) The CHANNEL CALIBRATION shall be established in accordance with established station calibration procedures.

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

(5) Same as (2) except Parts 3 and 4 are deleted.

FREQUENCY NOTATION:

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

Amendment No. 89
Correction letter of 6-20-86

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21 (Cont'd)

B. Liquid Effluents

Applicability: At all times.

Specification:

1. Concentration

a. The concentration of radioactive material in water Offsite (Figure 1.1) due to radioactive liquid effluent shall not exceed the concentration specified in 10 CFR Part 20.106 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} μCi/ml total activity.

b. With the concentration of radioactive material released Offsite exceeding the limit, attend to the cause without delay and restore the concentration within the above limit.

c. The provisions of Specification 6.5.2 do not apply.

4.21 (Cont'd)

B. Liquid Effluents

1. Concentration

a. Radioactive liquid wastes shall be sampled and analyzed according to Table 4.21.B.1.

b. The analytical results shall be used with methods in the ODAM to verify that the average concentration beyond the site boundary does not exceed Specification 3.21.B.1.a, when Sr-89, Sr-90 and Fe-55 concentrations are averaged over no more than 3 months and other radionuclide concentrations are averaged over no more than 31 days.

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

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

Amendment No. 89
~~Collection~~ Letter of 6-20-86

Prairie Island Nuclear Station
Technical Specifications

NOTES FOR TABLE 4.21.B.1

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

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

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

Where:

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

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

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

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

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

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

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

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

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

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

(9) Analysis may be performed after release.

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

FREQUENCY NOTATION:

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.B (Cont'd)	4.21.B (Cont'd)
2. Liquid Dose	2. Liquid Dose
a. The dose to a Member of the Public due to radioactive material in liquid effluents offsite (see Figure 1.1) shall not exceed 1.5 mrem to the total body or 5 mrem to any body organ during any calendar quarter and not more than 3 mrem to the total body or 10 mrem to any body organ during any calendar year.	a. Dose Assessment - An assessment of compliance with Specification 3.21.B.2.a shall be made in accordance with the Offsite Dose Assessment Manual (ODAM) at least once per 31 days.
b. In the event Specification 3.21.B.2.a is exceeded, prepare and submit to the Commission within 31 days after the end of the quarter in which the limit was exceeded, pursuant to Specification 6.5.3, a Special Report in lieu of any other report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken.	b. In any quarter in which radioactive liquid releases are made and the radwaste system is not operated, a projection of the prospect of compliance with Specification 3.21.B.2.a shall be made in accordance with the ODAM.
c. Appropriate parts of the system shall be used to reduce the concentration of radioactive materials in liquid wastes prior to their discharge when the pre-release analysis indicates a radioactivity concentration, excluding tritium and noble gases, in excess of 0.01 μCi/ml.	
d. With radioactive liquid waste being discharged without treatment in excess of the limit in Specification 3.21.B.2.c, prepare and submit to the Commission within 31 days after the end of the quarter in which the limit was exceeded, pursuant to Specification 6.5.3, a Special Report in lieu of any other report which includes the following information: 1) Indentification of equipment or subsystems not OPERABLE and the reason for nonoperability. 2) Action(s) taken to restore the nonoperable equipment to OPERABLE status. 3) Summary description of action(s) taken to prevent a recurrence.	
e. The provisions of Definition J are not applicable.	

Amendment No. 89
Correction letter of 6-20-86

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.B (Cont'd)	4.21.B (Cont'd)
3. Temporary, Outside Storage Tanks of Radioactive Liquid	3. Temporary, Outside Storage Tanks of Radioactive Liquid
a. In the event temporary, unprotected tanks are used outside to store radioactive liquid, the contents of each tank shall not exceed 10 curies, excluding H-3 and dissolved noble gas.	a. When radioactive liquid is being added to a temporary, unprotected outside storage tank, the liquid shall be sampled and analyzed for radioactivity at least once per 7 days.
b. If the quantity of radioactive material in a temporary, unprotected storage tank outside exceeds 10 curies, excluding H-3 and dissolved noble gas, immediately suspend addition of radioactive material and begin measures to reduce the content to 10 curies or less without delay and describe the events leading to the condition in the next Semiannual Radioactive Materials Release Report.	
c. The provisions of Definition J are not applicable.	

Hooper Nuclear Station
Technical Specifications

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
C. Gaseous Effluents	C. Gaseous Effluents
Applicability: At all times.	Cooper Nuclear Station Technical Specifications
Specification:	
1. Concentration	1. Concentration
a. The dose rate Offsite due to radioactive noble gases shall not exceed 500 mrem/yr to the total body or 3000 mrem/yr to skin.	a. The release rate of radioactive noble gas shall be monitored according to Specification 3.21.A.2.
b. The dose rate Offsite due to H-3, I-131, I-133, and radioactive material in particulate form having half-lives of 8 days or more in gaseous effluent shall not exceed 1500 mrem/yr to any body organ when the dose rate due to H-3, Sr-89, Sr-90, and alpha emitting radionuclides is averaged over no more than 3 months and the dose rate due to other radionuclides is averaged over no more than 31 days.	b. A radioactive noble gas effluent monitor shall be set to cause automatic alarm when the monitor alarm setpoint, determined as specified in the ODAM, is exceeded.
c. In the event a limit in Specification 3.21.C.1.a or b is exceeded, decrease the release rate to comply with the limit.	c. An assessment of compliance with Specification 3.21.C.1.b shall be made in accordance with the ODAM at least once every 31 days.
d. The provisions of Definition J are not applicable.	
2. Noble Gases Dose	2. Noble Gases Dose
a. The air dose Offsite (see Figure 1.1) due to noble gases released in gaseous effluents shall not exceed 5 mrad from gamma radiation and 10 mrad from beta radiation during any calendar quarter. The air dose Offsite due to noble gases released in gaseous effluents shall not exceed 10 mrad from gamma radiation and 20 mrad from beta radiation during any calendar year.	a. Dose Assessment - An assessment of compliance with Specification 3.21.C.2.a shall be made in accordance with the ODAM at least once every 31 days.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C (Cont'd)
b. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding Specification 3.21.C.2.a, prepare and submit to the Commission within 31 days after the end of the quarter in which the limit was exceeded, pursuant to Specification 6.5.3 a Special Report in lieu of any other report which identifies the cause(s) and defines the corrective actions taken.	Cooper Nuclear Station Technical Specifications
c. The provisions of Definition J are not applicable.	
3. Iodine and Particulate	3. Iodine and Particulate
a. The dose to a Member of the Public due to I-131, I-133, and radioactive material in particulate form having a half-life greater than 8 days in gaseous effluents Offsite (see Figure 1.1) shall not exceed 7.5 mrem to any organ during any calendar quarter and 15 mrem to any organ during any calendar year.	a. Radioactive gaseous effluent other than noble gases shall be sampled and analyzed as specified in Table 4.21.C.1.
b. In the event Specification 3.21.C.3.a is exceeded, prepare and submit a Special Report to the NRC within 31 days after the end of the quarter in which the specification was exceeded, pursuant to Specification 6.5.3.B and in lieu of any other report, which identifies the cause(s) and describes the corrective action taken.	b. Dose Assessment - An assessment of compliance with Specification 3.21.C.3.a shall be performed in accordance with the ODAM at least once every 31 days.
c. The provisions of Definition J are not applicable.	

Amendment No. 89
Collection letter of 6-20-86

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C (Cont'd)
4. Gaseous Releases	4. Gaseous Releases
a. Every reasonable effort shall be made to operate at least one train of charcoal adsorbers in the Offgas Treatment System whenever the main condenser air ejector is in operation except during startup or shutdown with reactor power less than 10% of rated or when system cannot function due to low offgas flow.	a. In any month in which radioactive material in gaseous effluent is being released without treatment, a projection of the prospect of compliance with Specification 3.21.C.4.b shall be made at least once every 31 days in accordance with the ODAM.
b. The Exhaust Ventilation Treatment System (EVTS) shall be operated to treat radioactive materials in effluent air when the projected dose to a Member of the Public due to the activity in air effluent via the EVTS would exceed 0.3 mrem to any body organ.	b. Operation of the Offgas Treatment System charcoal adsorbers shall be verified by using the gaseous effluent monitoring program in Specification 3.21.A.2.
c. In the event radioactive gas from the main condenser air ejector is discharged in effluent air for more than 7 days without treatment by charcoal adsorbers or in the event air is discharged via an exhaust ventilation treatment system for more than 31 days without treatment and the limit of Specification 3.21.C.4.b is exceeded, prepare and submit a Special Report to the NRC, pursuant to Specification 6.5.3 and in lieu of any other report, which identifies the inoperable equipment and describes the corrective action taken.	
d. The provisions of Definition J are not applicable.	

/ Amendment No. 89
Correction letter of 6-20-86

TABLE 4.21.C.1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

	Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)(1)
A.	Elevated Release Point (ERP), Reactor Bldg Vent, Augmented Radwaste Bldg Vent, Turbine Bldg Vent (Gaseous)	M(3) Grab Sample	M(3)	Principal Gamma Emitters(6)	1×10^{-4}(2)
		Q(9) Grab Sample	Q	H-3	1×10^{-6}
B.	All Release Types as Listed in A Above	Continuous(5)	W(4) Charcoal Sample	I131 I-133	1×10^{-12} 1×10^{-10}
		Continuous(5)	W(4) Particulate Sample	Principal Gamma Emitters(6) (I-131, Others)	1×10^{-11}(2)
		Continuous(5)	Q Composite Particulate Sample(7)	Sr89, Sr-90 Gross Alpha	1×10^{-11} 1×10^{-11}
		Continuous(5)	Noble Gas Monitor	Gross Noble Gases Beta and Gamma(8)	1×10^{-6}

Amendment No. 89
Collection letter 86 6-20-84

NOTES FOR TABLE 4.21.C.1

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

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

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

Where:

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

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

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

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

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

Amendment No. 89
Correction letter of 6-30-86

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

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

(6) The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances cause LLD's higher than required for more than 31 days, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.

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

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

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

FREQUENCY NOTATION:

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

Amendment No. 89
Correction letter of 6-20-86

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C (Cont'd)
5. Hydrogen Concentration	5. Hydrogen Concentration
a. The concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners shall be limited to ≤ 2% by volume.	a. The concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners shall be determined by continuously monitoring the waste gases in the main condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3.21.A.2.
b. With the concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners exceeding the limit, restore the concentration to within the limit within 48 hours.	
c. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.5.2 are not applicable.	
6. Air Ejector	6. Air Ejector
a. The gross radioactivity (beta and/or gamma) rate of noble gases measured at the main condenser air ejector shall be limited to ≤ 1 Ci/sec at the air ejector.	a. The gross radioactivity (beta and/or gamma) rate of noble gases from the main condenser air ejector shall be determined at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the main condenser air ejector:
b. With the gross radioactivity (beta and/or gamma) rate of noble gases at the main condenser air ejector exceeding Specification 3.21.C.6.a, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21.C (Cont'd)	4.21.C.6 (Cont'd)
7. Containment a. Whenever the primary containment is vented/purged, it shall be vented/purged through the Standby Gas Treatment System. With this specification not satisfied, suspend all venting/purging of the containment. This specification does not apply to Normal Ventilation, or during start-up while performing primary containment inerting in accordance with Specification 3.7.A.5.b following a shutdown of greater than 24 hours. b. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.5.2 are not applicable.	1) At least once per 31 days during normal operation. 2) Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant. b. The radioactivity rate of noble gases at or near the outlet of the main condenser air ejector shall be monitored in accordance with Table 3.21.A.2.
D. <u>Effluent Dose Liquid/Gaseous</u> <u>Applicability</u>: At all times. <u>Specification</u>: 1. The dose or dose commitment to a (actual) member of the public due to radiation and radioactive releases from Cooper Station shall not exceed 75 mrem to his thyroid or 25 mrem to his total body or any other body organ during a calendar year. In the event the calculated dose from radioactive material in liquid or gaseous effluents exceeds two times the limit of Specification 3.21.B.2.a, 3.21.C.2.a, or 3.21.C.3.a, prepare and submit a Special Report, in lieu of any other report, to the Commission pursuant to Specification 6.5.3 within 31 days which 1) defines actions to be taken to reduce releases and prevent recurrence and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a member of the public due to radiation and radioactive releases from Cooper Station during the calendar year through the period covered by the calculation was less than limits stated in this Specification. If the estimated dose exceeds the limits stated herein, and if the condition resulting in doses exceeding these limits has not already been corrected, submission of the Special Report shall be deemed a timely request for a variance in accord with provisions of 40 CFR Part 190, provided	D. <u>Effluent Dose Liquid/Gaseous</u> 1. Dose Calculations - The cumulative dose to a Member of the Public contributed by radioactive material in gaseous and liquid effluents shall be calculated at least once per year in accordance with the ODAM in order to verify compliance with Specification 3.21.D.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21.D (Cont'd)

information specified in 40 CFR Part 190.11(b) is included. In that event, a variance is granted until NRC Staff action on the item is complete.

2. The provisions of Definition J are not applicable.

E. Solid Radioactive Waste

Applicability: During solid radwaste processing.

Specification:

1. The appropriate equipment of the solid radwaste system shall be operated to process radioactive waste containing liquid and liquid destined for disposal subject to 10 CFR Part 61 to a form that meets applicable requirements of 10 CFR Part 61.56 before the waste is shipped from the site.

2. Suspend delivery to a carrier for transport of any container of waste subject to Specification 3.21.E.1 which does not comply with 10 CFR Part 61.56.

4.21 (Cont'd)

Cooper Nuclear Station
Technical Specifications

E. Solid Radioactive Waste

1. Operating parameters and limits for the solidification of radioactive waste were established during preparational testing of the system. Radioactive waste solidification shall be performed in accordance with established parameters and limits. In addition, every 10th batch of dewatered waste will be sampled prior to solidification and analyzed for pH.

2. Each drum of solidified or dewatered radioactive waste will be inspected, prior to capping, to insure that there is no free standing liquid on top of the solid waste.

3. The Semiannual Radioactive Material Release Report in Specification 6.5.1.F shall include the following information for radioactive solid waste shipped offsite during the report period:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
3. The provisions of Definition J are not applicable.	a. Container burial volume, b. Total curie quantity (determined by measurement or estimate), c. Principal gamma radionuclides (determined by measurement or estimate), d. Type of waste, e. Solidification agent.
F. Monitoring Program	F. Monitoring Program
Applicability: At all times.	
Specification:	
1. As a minimum the radiological environmental monitoring program shall be conducted as specified in Table 3.21.F.1. Analytical techniques used shall be such that the detection capabilities in Table 3.21.F.2 are achieved.	1. Radiological environmental samples shall be collected and analyzed as specified in Table 3.21.F.1.
2. In the event the radiological environmental monitoring program is not conducted as specified in Table 3.21.F.1, prepare and submit to the Commission in the Annual Operating Report the reasons for not conducting the program in accordance with Table 3.21.F.1 and the plans for preventing a recurrence.	2. A land use census shall be conducted annually and shall identify the location of the nearest garden that is greater than 500 square feet in area and that yields edible leafy vegetables, the location of the nearest milk animal, and the location of the nearest resident in each of the 16 meteorological sectors within three miles of the Station. The land use census shall be conducted at least once per 12 months.
3. When the radioactivity in a sampled environmental medium, averaged over a calendar quarter, exceeds an appropriate value stated in Table 6.5-2, prepare and submit to the Commission within 31 days from the end of the affected calendar quarter a Special Report in accordance with 6.5.3 which includes an evaluation of any release conditions, environmental factors or other conditions which caused the value(s) of Table 6.5-2 to be exceeded. If the radioactivity in environmental sample(s) is not attributable to release from the	3. The results of sample analyses performed shall be summarized in the Annual Radiological Environmental Report. 4. The results of the land use census shall be included in the Annual Radiological Environmental Report.

Amendment No. 89
Correction letter of 6-20-86

3.21.F (Cont'd)

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

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

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

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

7. The provisions of Definition J are not applicable.

4.21 (Cont'd)

Cooper Nuclear Station
Technical Specifications

Amendment No. 89
Correction letter of 6-20-86

Amendment No. 89
Collection letter 86-20-86

TABLE 3.21.F.1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

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

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

Exposure Pathway and/or Sample	Number of Sample Stations	Sampling and Collection Frequency	Type and Frequency of Analysis
4. Ingestion			
a. Milk (Nearest Producer)	At least one location	At least once per 15 days during Peak Pasture Period[d]; at least once per 31 days at other times.	Gamma isotopic[b] and I-131 analysis of each sample.
b. Milk (Other Producers)	At least 2 locations	At least once per 92 days.	Gamma isotopic[b] and I-131 analysis of each sample.
c. Fish	At least 2 locations	Two times per year (once in the summer and once in the fall). Attempt to include the following: 1. Bottom feeding species 2. Middle-Top feeding species	Gamma isotopic[b] analysis on edible portions.
d. Food Products	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic[b] and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic[b] and I-131 analysis.

Amendment No. 89
Corrected letter of 6-20-86

Pioneer Nuclear Station
Technical Specifications

NOTES FOR TABLE 3.21.F.1

a. DELETED

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

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

Any radionuclide detected, i.e., having a measured concentration greater than the LLD, whether or not it is one of the 19 nuclides listed above, shall be regarded as present in the sample.

c. Thermoluminescent Dosimeters (TLD) is a single phosphore. Two or more phosphores in one package are considered to be two or more dosimeters.

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

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

	Lower Limit of Detection (LLD)[a]					
Anaysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}				
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	9[b]	7×10^{-2}		1	60	
^{134}Cs	15	5×10^{-2}	130	15	60	150
^{137}Cs	18	6×10^{-2}	150	18	80	180
^{140}Ba	60			60		
^{140}La	15			15		

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

Amendment No. 89
Collection letter of 6-20-86

Cooper Nuclear Station
Technical Specifications

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

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

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

Where

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

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

E is the counting efficiency (as counts per transformation)

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

2.22 is the number of transformation per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

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

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

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

b. LLD for drinking water.

Amendment No. 89
Correction letter of 6-20-86

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.21 (Cont'd)	4.21 (Cont'd)
G. Interlaboratory Comparison Program	G. Interlaboratory Comparison Program
Applicability: Applicable at all times to Radiological Environmental Monitoring Program.	
Specification:	
1. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the NRC.	1. A brief summary of results obtained as part of the Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Report, pursuant to Specification 6.5.1.E.
2. With analyses not being performed as required in Specification 3.21.G.1, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Report.	
3. The provisions of Definition J are not applicable.	

Amendment No. 89
Correction letter of 6-20-80

3.21.A & 4.21.A INSTRUMENTATION

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

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

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

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

In the event no flow rate measurement device is operable on a gaseous stream, alternative 24-hour estimates are adequate since the system design is constant flow and loss of flow is alarmed in the control room.

3.21.B & 4.21.B LIQUID EFFLUENTS

3.21.B.1 & 4.21.B.1 Concentration

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

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

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

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

3.21.B.1 & 4.21.B.1 Concentration (Cont'd)

Compliance with 10CFR Part 20.106 implies that the concentration limit represented by 10CFR Part 20, Appendix B, Table 2 will be met within a suitable and reasonable averaging time for assessing compliance. That averaging time is dependent upon the resolving time of the measurements or estimates which are used to evaluate compliance. Assessment of compliance is done by sampling and analysis according to Specification 4.21.B.1.2, by estimating or measuring the maximum release flow and the minimum dilution flow coincident during the period of release represented by the sample, and by computing the concentration as a fraction of the limit in the unrestricted area periodically on the basis of these data.

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

Specifications 3.21.B.2, 3.21.C.2 and 3.21.C.3 implement the requirements of 10 CFR Part 50.36a and of 10 CFR Part 50, Appendix I, Section IV. These specifications state limiting conditions for operation (LCO) to keep levels of radioactive materials in LWR effluents as low as is reasonably achievable. Compliance with these specifications will also keep average releases of radioactive material in effluents at small percentages of the limits specified in 10 CFR Part 20.106. Surveillance Requirements provide for the measurement of releases and calculation of doses to verify compliance with the Specifications. Action statements in these Specifications implement the requirements of 10 CFR Part 50.36(c)(2) and 10 CFR Part 50, Appendix I, Section IV.A in the event an LCO is not met. Annual dose limitations stated in Specifications 3.21.B.2, 3.21.C.2, and 3.21.C.3 are not strict limits as used elsewhere in the Technical Specifications (are not an immediate safety concern) but do obligate NPPD to take the applicable reporting action required in Specifications 3.21.B.2.b, 3.21.C.2.b, or 3.21.C.3.b.

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

> Section 50.36a(b) of 10 CFR Part 50 provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications which take into account the need for operating flexibility and at the same time ensure that the licensee will exert his best efforts to keep levels of radioactive materials in effluents as low as practicable. The Appendix I that we adopt provides more specific guidance to licensees in this respect.

Aemndent No. 89
Correction letter of 6-20-86

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

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

A. The Rule

Section IV of Appendix I specifies action levels for the licensee. If, for any individual light water cooled nuclear power reactor, the quantity of radioactive material actually released in effluents to unrestricted areas during any calendar quarter is such as to cause radiation exposure, calculated on the same basis as the design objective exposure, which would exceed one-half the annual design objective exposure, the licensee shall make an investigation to identify the causes of these high release rates, define and initiate a program of action to correct the situation, and report these actions to the Commission within 30 days of the end of the calendar quarter.

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

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

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

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

With the implementation of Specification 3.21.B.2 and 4.21.B.2 there is reasonable assurance that Station operation will not cause a radionuclide concentration in public drinking water taken from the River that exceeds the standard for anthropogenic radioactivity in community drinking water.

Amendment No. 89
Correction letter of 6-20-86

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

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

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

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

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

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

thyroid	1.95 manrem
total body	0.56
total	2.5 manrem

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

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

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

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

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

Unit volume operating costs are about:

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

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

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

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

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

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

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

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

(Cont'd)

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

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

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

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

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

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

[3] NRC Commissioners, p. 101.

[4] NRC Staff, op. cit.

[5] NRC Commissioners, op. cit.

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

3.21.C & 4.21.C GASEOUS EFFLUENTS

3.21.C.1 & 4.21.C.1 Concentration

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

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

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

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

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

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

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

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

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

The OPERABILITY of the gaseous radwaste treatment system and the ventilation exhaust treatment systems ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems are specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

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

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

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

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

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

This specification provides reasonable assurance that releases of Iodine from drywell purging during power operations and 24 hours after shutdown will not be excessively large, particularly due to Iodine spiking. The exemptions to using the SBGT system are intended to minimize the time the SBGT system is on line while coolant temperature is greater than 200°F, hence to decrease the probability of damage to the SBGT filters that could occur from overpressurization due to a LOCA and the main purge and vent valves open.

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

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

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

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

3.21.E & 4.21.E SOLID RADIOACTIVE WASTE

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

3.21.F & 4.21.F MONITORING PROGRAM

The radiological environmental monitoring program, including the land use census, is conducted to satisfy the requirements of 10 CFR Part 50, Appendix I, Section IV.B.2 and 3. The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The environmental monitoring program described in Table 3.21.F.1 is the minimum program which will be maintained. The Offsite Dose Assessment Manual (ODAM) describes in detail the actual monitoring program which is performed to ensure compliance with the specified minimum program. Control of the radiological environmental monitoring program, including the ODAM, rests with the Environmental Affairs Division of Operations and not the Cooper Nuclear Station organization.

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

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

3.21.G & 4.21.G INTERLABORATORY COMPARISON PROGRAM

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

6.2 (cont'd)

a. Verification of compliance with internal rules, procedures (for example: normal, off-normal, emergency, operating, maintenance, surveillance, test, and radiation control procedures) and applicable license conditions at least once per 24 months.

Cooper Nuclear Station
Technical Specifications

b. The training, qualification, and performance of the operating staff at least once per 24 months.

c. The Emergency Plan and implementing procedures at least once per 12 months.

d. The Security Plan and implementing procedures at least once per 12 months.

e. The facility fire protection and its implementing procedures at least once per 24 months.

f. A fire protection and loss prevention inspection will be performed utilizing either qualified off-site licensee personnel or an outside fire protection consultant at least once per 12 months.

g. An inspection and audit by an outside qualified fire protection consultant shall be performed at least once per 36 months.

h. The Radiological Environmental Monitoring Program and the Offsite Dose Assessment Manual with their implementing procedures at least once every 24 months.

6.3 PROCEDURES AND PROGRAMS

6.3.1 Introduction

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

Robert Huff or Sandra: Technical Specifications

6.3.2 Procedures

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

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

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

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

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

E. Implementing procedures for the fire protection program.

F. Administrative procedures for shift overtime.

G. Implementing procedures for the Offsite Dose Assessment Manual.

6.3.3 Maintenance and Test Procedures

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

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

B. Routine testing of standby and redundant equipment.

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

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

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

6.3.4 Radiation Control Procedures

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

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

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

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

4. Documentation of all challenges to relief valves or safety valves.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

E. Annual Radiological Environmental Report

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

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

   a. A summary of doses to a Member of the Public Offsite due to Cooper Station aqueous and airborne radioactive effluents, calculated in accordance with methods compatible with the ODAM.

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

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

c. Summarized and tabulated results in the format of Table 6.5-1 of analyses of samples required by the radiological environmental monitoring program, and taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

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

F. Semiannual Radioactive Material Release Report

1. A report of radioactive materials released from the Station during the preceding six months shall be submitted to the NRC within 60 days after January 1 and July 1 of each year*.

2. A Semiannual Radioactive Material Release Report shall include the following:

a. A summary by calendar quarter of the quantities of radioactive liquid and gaseous effluents released from the Station. The data should be reported in the format recommended in Regulatory Guide 1.21, Appendix B, Tables 1 and 2.

b. A summary of radioactive solid waste shipped from the Station, including information named in Specification 4.21.E.3.

c. A summary of meteorlogical data collected during the year shall be included in the Semiannual Report submitted within 60 days after January 1 of each year.

d. A list and brief description of each unplanned release of gaseous or liquid radioactive effluent that causes a limit in Specification 3.21.B.1.a, 3.21.B.2.a, 3.21.C.1.a, 3.21.C.2.a, or 3.21.C.3.a to be exceeded.

e. Calculated offsite dose to humans resulting from the release of effluents and their subsequent dispersion in the atmosphere reported in accordance with Regulatory Guide 1.21.

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

TABLE 6.5-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

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

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

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

Table Notes:

(1) Nominal Lower Limit of Detection (LLD) as defined in Definition K.A.

(2) Mean and Range based upon detectable measurements only. Fraction of detectable measurements at specified locations indicated in brackets [].

Amendment No. 89

Collection letter 06-20-86

6.5.2 Reportable Events

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50. The NRC shall be notified and a report submitted pursuant to the requirements of Section 50.73. Each Reportable Event shall be reviewed by SORC and the results of this review shall be submitted to SRAB and the Assistant General Manager - Nuclear.

Amendment No. 89

Correction letter of 6-30-86

XXXXXXX

6.5.3 Unique Reporting Requirements

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

A. None

B. Special Reports

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

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

1. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 6.5-2 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 6.5-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 6.5-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.21.B.2.a and 3.21.C.3.a. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

TABLE 6.5-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

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

(a) For drinking water samples. This is the 40 CFR 141 value.
(b) Concentration of parent or daughter.
(c) For samples of water not used as a source of drinking water.

Amendment No. 89

Collection letter of 6-30-86

6.6 Process Control Program (PCP)

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

6.6.2 District Initiated Changes

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

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

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

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

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

6.7 Offsite Dose Assessment Manual (ODAM)

6.7.1 The ODAM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. The ODAM also describes the Environmental Radiation Monitoring Program.

6.7.2 District Initiated Changes

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

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

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

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

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

6.8 Major Changes To Radioactive Waste Treatment Systems (Liquid, Gaseous, and Solid)

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

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

APPENDIX B

TO

OPERATING LICENSE NO. DPR-46

FOR THE

COOPER NUCLEAR STATION

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

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

Amendment No. 89
Correction letter of 6-20-86

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, Nebraska Public Power District 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. Nebraska Public Power District filed this information with the Commission by letter dated March 7, 1984 and supplemented by letters dated April 10, 1984 and July 19, 1984, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-46 for Cooper Nuclear Station. The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0473. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 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-0473 provides radiological effluent technical specifications for boiling 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 of 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

The enclosed report (EGG-PBS-6612) was prepared for us by EG&G Idaho, Inc., as part of our technical assistance contract program*. Their report provides their technical evaluation of the compliance of the Licensee's submittals up to and including the April 10, 1984 submittal with NRC provided criteria. The staff has reviewed the TER and agrees with the contractor's evaluation of the radiological effluent technical specifications and the conclusions expressed in the TER. That is, the staff finds that the licensee's proposed RETS submitted March 7, 1984 and supplemented by the submittal dated April 10, 1984, meets the guidelines of NUREG-0473, "Standard Radiological Effluent Technical Specification for Boiling Water Reactors"

*Although the Report contains a legend that this is "an informal report intended for use as a preliminary working document," the report in fact represents EG&G's final evaluation of this amendment request for NRC. These statements will be clarified in reports issued in the near future.

Table 1. ation Between Provisions of the Regulations and the Standard .iological Effluent Technical Specifications for Pressurized .. er Reactors and Boiling Water Reactors

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

Provisions of Title 10 Code of Federal Regulations	Standard Radiological Effluent Technical Specifications: Instrumentation		Radioactive Effluents: Liquid				Radioactive Effluents: Gaseous: PWR/BWR				Radioactive Effluents: Gaseous: PWR		Radioactive Effluents: Gaseous: BWR					Rad. Envir. Monitoring			Design Features	Administrative Control						
	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.

TABLE 1

and is, therefore, acceptable. The review of the licensee's additional RETS submittal dated July 19, 1984 was performed solely by the staff. The staff finds that the proposed RETS revision of July 19, 1984 is also consistent with the guidelines of NUREG-0473 and is, therefore, acceptable.

The staff also agrees with the conclusion expressed in the TER that the licensee's Offsite Dose Assessment Manual (ODAM), dated January 1984, is generally consistent with the guidelines in NUREG-0133. The discrepancies in the ODAM found by the contractor will be addressed in a separate action. Because the ODAM is a reference to the RETS which is expected to undergo periodic revision, issuance of the RETS is not dependent on prior resolution of the discrepancies noted in the TER. However, these minor discrepancies should be resolved prior to the July 1, 1986 implementation of those RETS sections which reference the ODAM.

SUMMARY

The proposed changes to the radiological effluent technical specifications for Cooper Nuclear Station have been reviewed, evaluated, and found to be in compliance with the requirements of the sections of 10 CFR Part 20 and 10 CFR Part 50 referenced above and with the guidelines of NUREG-0133 and NUREG-0473 (Cooper is a boiling water reactor) and thereby fulfill all the requirements of the regulations to radiological effluent technical specifications.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational 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.

6.0 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 this amendment will not be inimical to the common defense and security or the health and safety of the public.

Attachments:
Technical Evaluation Report
Review of ODCM

Principal Contributors: C. Miller, C. Willis, W. Meinke and F. Congel

Dated: December 24, 1984

EGG-PBS-6612

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) IMPLEMENTATION - COOPER NUCLEAR STATION

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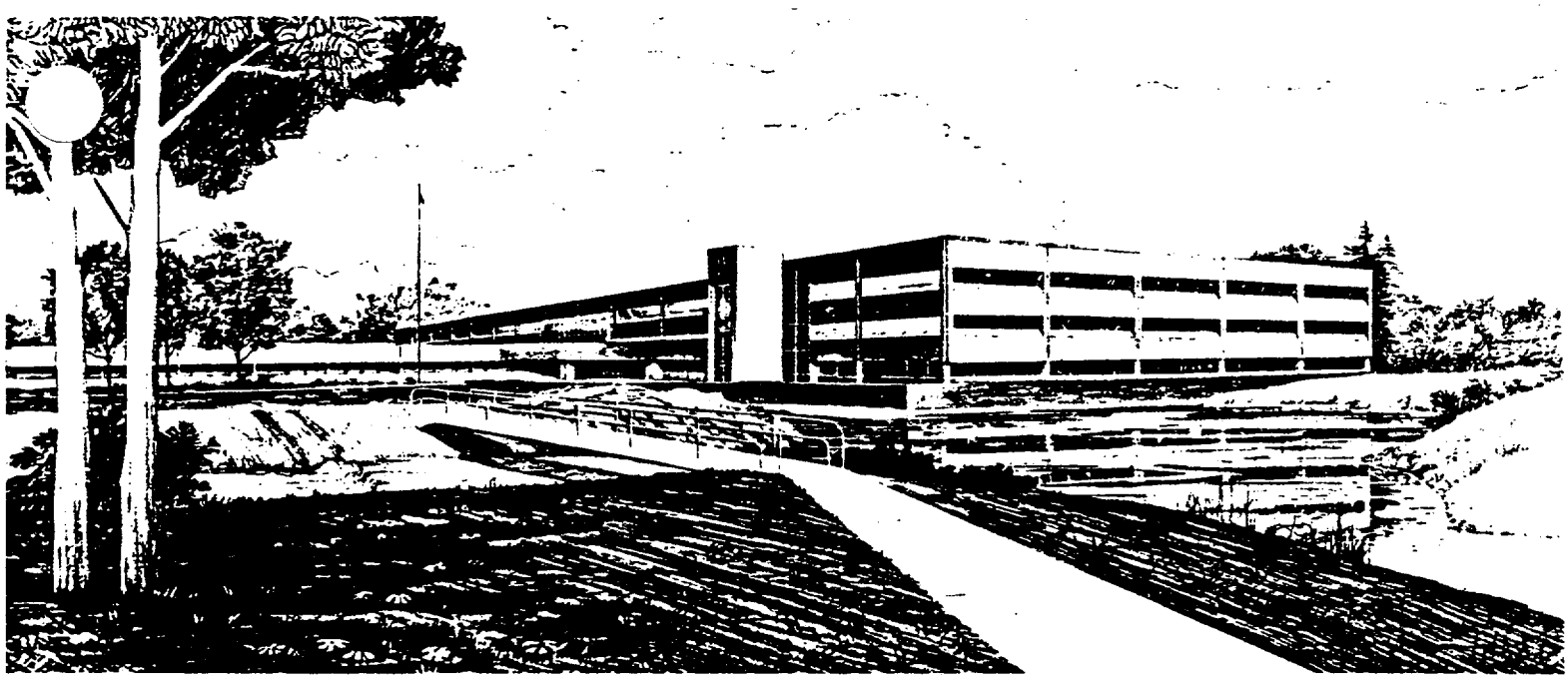
Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



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

EG&G Idaho

ATTACHMENT 1

EGG-PBS-6612

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) IMPLEMENTATION - COOPER NUCLEAR STATION

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Published May 1984

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Prepared for the
U. S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6459

ABSTRACT

A review of the Radiological Effluent Technical Specifications (RETS) for the Cooper Nuclear Station 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-0473, Revision 3, "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors." Draft submittals were discussed with the Licensee by both EG&G and the NRC staff 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 NRC review guidelines. The proposed Offsite Dose Calculation Manual was reviewed and generally found to be in compliance with the NRC review guidelines.

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.

CONTENTS

	Page
Abstract	i
Foreword	ii
1. INTRODUCTION	1
1.1 Purpose of the Technical Evaluation	1
1.2 Generic Issue Background	1
1.3 Plant-Specific Background	4
2. REVIEW CRITERIA	6
3. TECHNICAL EVALUATION	8
3.1 General Description of Radiological Effluent System	8
3.2 Radiological Effluent Technical Specifications	12
3.3 Offsite Dose Calculation Manual	23
4. CONCLUSIONS	26
5. REFERENCES	28

FIGURES

1. Liquid Radwaste System and Effluent Pathways	9
2. Gaseous Radwaste System and Effluent Pathways	11

TABLE

Number	Title	Page
1	Correspondence of Provisions of NUREG-0473, The Licensee's Current Technical Specifications and the Licensee's Proposal for Cooper Nuclear Station	27

1. 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 Cooper Nuclear Station (CNS) with regard to Radiological Effluent Technical Specifications (RETS), and the proposed Offsite Dose Assessment Manual (ODAM*).

The evaluation used criteria proposed by the Nuclear Regulatory Commission (NRC) staff in the model Technical Specifications for boiling water reactors (BWRs), NUREG-0473,[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

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* The Licensee has elected to use the term "Offsite Dose Assessment Manual" which is equivalent to the NRC's "Offsite Dose Calculation Manual". These terms will be used interchangeably throughout this report.

deferred until the NRC completed the model RETS.

The model RETS deal with radioactive waste management systems and environmental monitoring. Although the model RETS address 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 Standards for Nuclear Power Operations," and submit reports to the NRC when the 40 CFR Part 190 limits have been or may 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.

The NRC position on the model RETS was established in May 1978 when the NRC's Regulatory Requirements Review Committee approved the model RETS: NUREG-0472 for PWRs and NUREG-0473 for BWRs. 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 the 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 Process Control Program (PCP) were issued in February 1979 to each utility at individual meetings. NUREG-0473, Revision 2,[1] was 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. Reviews leading to ultimate implementation of these documents were initiated by the NRC in September 1981 using subcontracted independent teams as reviewers.

As the RETS review progressed, feedback from the licensees led the NRC to modify some of the provisions in the February 1, 1980 versions of the model RETS to clarify specific concerns of the licensees and thus expedite the reviews. Starting in April 1982, the NRC distributed revised versions of the model 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.[10] Some interim changes regarding the Radiological Environmental Monitoring Section were issued in August 1982.[11] With the incorporation of these changes, the NRC issued Draft 7'' of Revision 3 of NUREG-0473[12] in September 1982 to serve as new guidance for the review teams.

1.3 Plant - Specific Background

In conformance with the 1975 directive[4] Nebraska Public Power District (NPPD), the Licensee of the Cooper Nuclear Station (CNS), filed with the commission on June 4, 1976[13] information necessary to permit evaluation of the CNS Technical Specifications with respect to the requirements of 10 CFR Part 50, Section II.A, II.B, and II.C of Appendix I. Additional information was furnished by letters dated February 14, 1977,[14] June 27, 1977,[15] November 14, 1977,[16] and February 8, 1978.[17] These submittals showed that the radioactive waste treatment systems installed at CNS are capable of maintaining releases of radioactive materials in effluents to ALARA levels in conformance with the requirements of 10 CFR Part 50, Section 50.34a.[18] NPPD did not submit proposed new RETS at this time.

Proposed RETS were submitted in 1979 to NRC and revised January 7, 1980.[19] EG&G Idaho, Inc. (EG&G), selected as an independent task review team, initiated a review and evaluation of the submittal. This submittal was compared with the model RETS and assessed for compliance with the requirements of 10 CFR 50, Appendix I, and 10 CFR 50, Appendix A.

Review comments and questions dated March 19, 1982[20] were mailed to the NRC and the Licensee prior to a site visit to the Cooper Nuclear Station. The site visit was arranged for the purpose of resolving questions identified in the March 1982 review of the January 1980 CNS RETS submittal.

During the site visit on April 5 and 6 of 1982, technical discussions resolved many of the shortcomings of the Cooper Nuclear Station RETS identified in the March 1982 review.

On June 9, 1982 the Licensee submitted revised RETS[21] to the NRC, addressing most of the discrepancies discussed during the site visit. These revised RETS were reviewed by the EG&G team and review comments and questions were transmitted to the NRC and J. W. Pilant of NPPD on October 11, 1982.[22] These comments and questions were discussed in a telephone conference between the Licensee and EG&G on November 5, 1982,[23] and between the NRC and EG&G on November 16, 1982.[24] A telephone conference was held between the NPPD consultant and EG&G on November 18, 1982[25] to discuss the items still outstanding. On December 20, 1982[26] EG&G forwarded a RETS status report to the NRC. A conference was held between EG&G and NRC personnel on January 11-13, 1983[27] to discuss the items identified in the status report. Following the conference, a letter identifying all unresolved issues in the Licensee's RETS, was transmitted to the NRC on February 15, 1983.[28]

In response to a request from the NRC,[29] the Licensee submitted revised RETS proposals on September 12, 1983.[30] This submittal was reviewed and information on a few remaining unresolved items was forwarded[31] to the NRC for final decisions. These issues were resolved in discussions between the NRC staff and representatives of NPPD, and appropriate modifications to the Cooper Nuclear Station RETS were submitted to the NRC on March 7, 1984[32] and April 10, 1984.[33] Resolution of the outstanding RETS issues allowed preparation of a TER for submittal to the NRC. The proposed RETS, submitted March 1984, with the April 1984 revisions are evaluated in Section 3.

The Licensee submitted a proposed ODAM in January 1980.[19] The submittal was reviewed and review questions[20] were mailed to the NRC and the Licensee prior to the site visit in April 1982. In July 1982, the Licensee submitted a revised ODAM to the NRC.[21] This revision was reviewed by EG&G and comments and questions were transmitted to the NRC and J. W. Pilant of NPPD on October 11, 1982.[22] These comments and questions were discussed in telephone conferences between EG&G personnel,

representatives of NPPD, and the NRC staff.[23,24,25] These telephone conferences were followed by a status report to the NRC on December 20, 1982[26] and a conference of EG&G personnel with the NRC staff on January 11-13, 1983.[27] Following this conference, a letter identifying all unresolved issues in the Licensee's ODAM was transmitted to the NRC on February 15, 1983.[28] In response to a request from the NRC[29], the Licensee submitted a revised ODAM submittal on September 12, 1983.[30] A review of this proposed ODAM was transmitted to the NRC by EG&G on November 10, 1983.[34] In March 1984 the NRC forwarded a revised ODAM dated January 1984 to EG&G. This ODAM was reviewed and the review transmitted to the NRC on April 9, 1984.

2. REVIEW CRITERIA

Review criteria for the RETS and ODCM were provided by the NRC in two documents:

1. NUREG-0473, RETS for BWRs,

2. 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.[35]

3. Offsite radiation doses shall be ALARA.

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

5. Radwaste tanks 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-0473, as revised, the NRC staff issued guidelines,[36,37] clarifications,[38,39] branch positions,[40,41] and NUREG-0543,[42] 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 Contents of the Offsite Dose Calculation Manual;[43] NUREG-0133;[9] and Regulatory Guide 1.109.[44] The format for the ODCM is left to the Licensee and may be simplified by tables and grid printouts.

3. 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 the Cooper Nuclear Station generating plant, a BWR. The unrestricted area boundary for both liquid and gaseous effluent coincides with the property line as shown in Figure 1.1 of the Technical Specifications.

3.1.1 Radioactive Liquid Effluents

As shown in Figure 1, there are two radioactive liquid effluent release paths to the circulating water discharge canal: the liquid radwaste effluent line and the service water system. The liquid radwaste effluent line is the common release line for potentially radioactive liquids from the waste sample tanks, floor drain sample tank, and the laundry drain tank. All releases from these systems are by batch mode. Sampling and analysis is performed prior to discharge to ensure 10 CFR 20 limits are not exceeded, and all releases are monitored at the radwaste effluent line. Water that may be contaminated with radioactive material is either reused or discharged to the Missouri River after dilution in the circulating water discharge canal.

High purity wastes are processed through filters and ion exchange

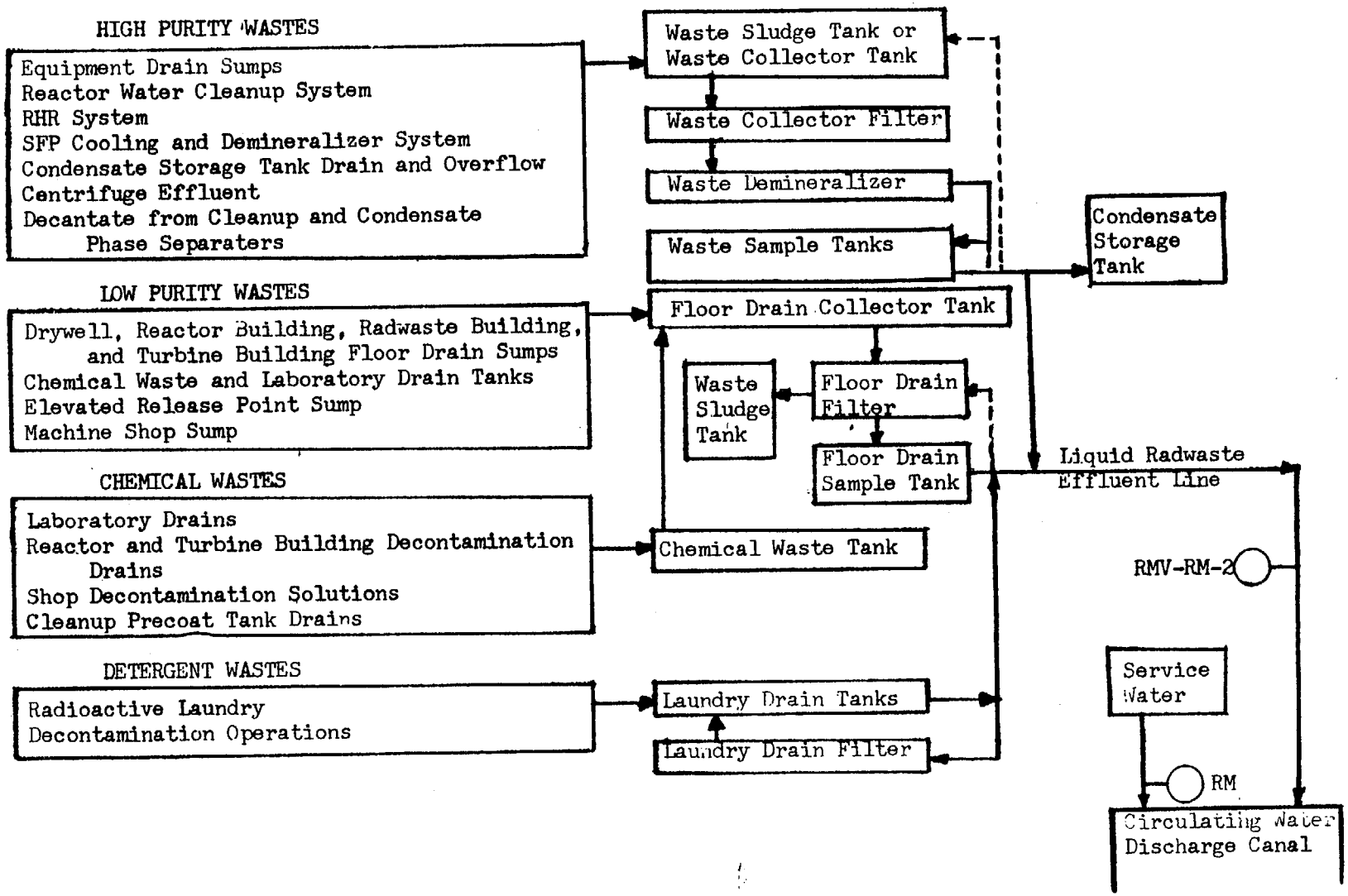

Figure 1. Liquid Radwaste System and Effluent Pathways

demineralizers and then sampled at the waste sample tanks. If the waste is suitable for reuse it is transferred to the condensate storage tank for reuse. If the water is not suitable for reuse it is normally returned to the system for additional processing. A pathway exists by which these wastes can be discharged to the circulating water discharge canal.

Low purity wastes are collected in the floor drain collector tank, then filtered and stored in the floor drain sample tank for sampling and analysis before being released to the circulating water discharge canal. Chemical wastes are normally neutralized, then transferred to the floor drain collector tanks for processing through the floor drain filter. They are then collected in the floor drain sample tank and sampled before discharge. If radioactivity content precludes discharge, the chemical wastes are processed and disposed of as solid waste.

Detergent wastes are collected in the laundry drain tanks, where they are sampled and then released to the circulating water discharge canal.

The normally non-radioactive releases from the service water system are monitored during discharge to the circulating water discharge canal.

3.1.2 Radioactive Gaseous Effluents

As shown in Figure 2 there are four radioactive gaseous effluent release points at the CNS. They are (a) elevated release point (ERP), (b) the reactor building vent, (c) the turbine building vent, and (d) the radwaste building vent. The sources of radioactive gases, the treatment given the gases, and their paths to the release points are outlined in Figure 2.

The condenser SJAE system is the major source of radioactive noble gases in the ERP effluents, which also include releases from the startup mechanical vacuum pump, the gland seal condensers, and the standby gas

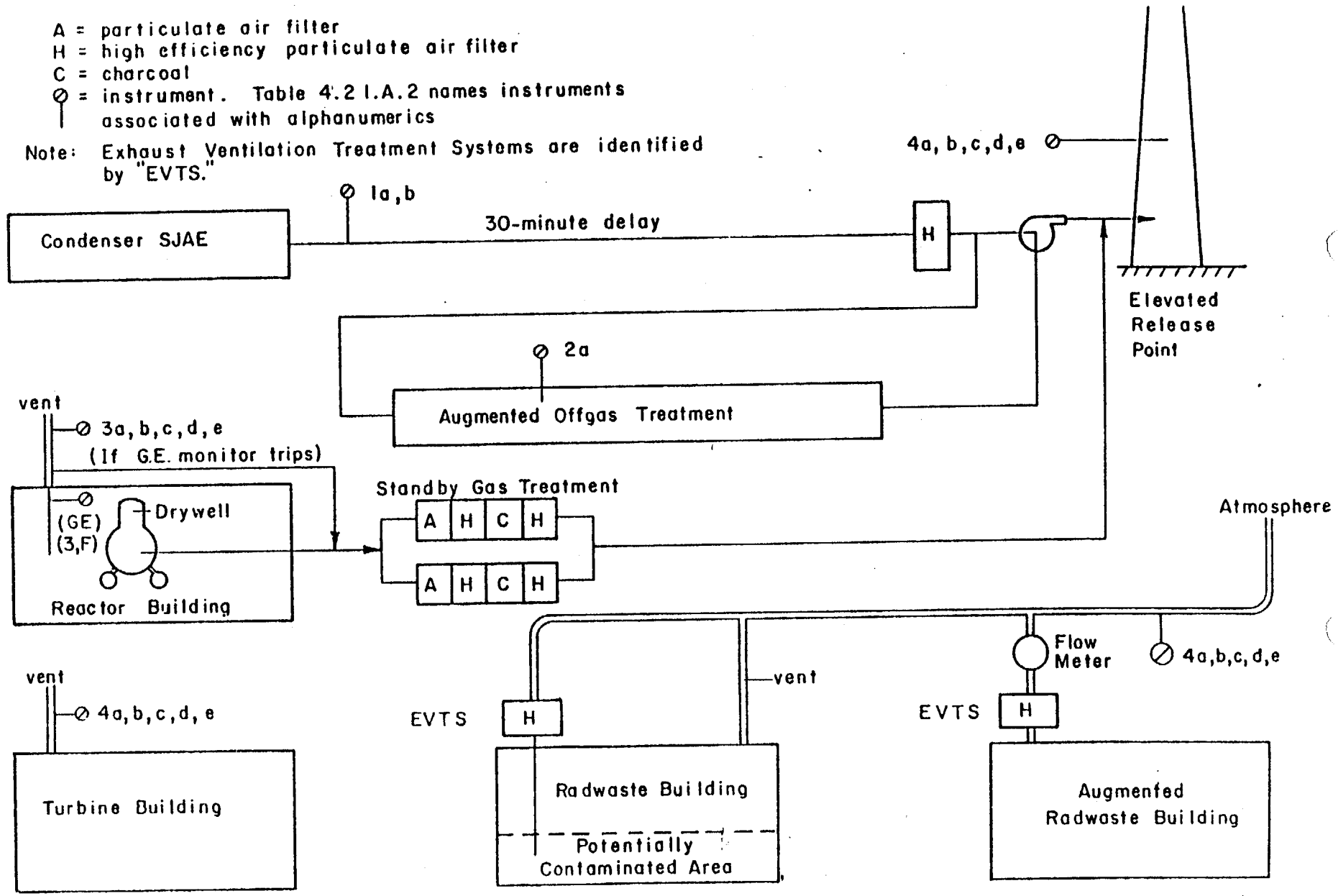

Figure 2. Gaseous Radwaste System and Effluent Pathways

treatment system. Gas from the drywell is routed through the standby gas treatment system before release at the ERP. Effluents from the HPCI gland seal are routed through a 1-minute delay line before being discharged.

Gaseous effluents from the radwaste building and the turbine building are released from the radwaste building vent and turbine building vent, respectively. Effluents from the reactor building are normally released at the reactor building vent, but may be routed through the standby gas treatment system. All radioactive gaseous effluent release points are monitored.

3.2 Radiological Effluent Technical Specifications

The following subsections describe the primary objectives of each section of the model RETS and summarize the commitments of the Licensee's RETS. A cross reference between 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 (MPCs) listed in 10 CFR Part 20.

3.2.1.1 Radioactive Liquid Effluent Instrumentation

The radioactive liquid effluent lines leading to the common release point at the CNS site are monitored during releases with adequate

instrumentation surveillance being performed. The locations of the monitors are shown in Figure 1.

Radiation readings above the monitors' setpoints initiate control room alarm annunciation. The radwaste effluent line monitor also initiates automatic termination of release when the setpoints are exceeded. The proposed applicability of monitoring "during release" for the liquid radwaste effluent line is acceptable since only batch releases are possible via this pathway. The RETS contain a commitment to perform surveillance of the monitoring instrumentation that ensures they will be OPERABLE.

3.2.1.2 Radioactive Gaseous Effluent Instrumentation

The radioactive gaseous effluent release points are monitored with adequate instrument surveillance being performed.

The main source of radioactive gaseous effluents from the elevated release point (ERP) is the unit's off-gas system. The off-gas system has its own radiation monitor capable of isolating the off-gas release pathway. The monitor is located between the steam jet air ejector and the 30-minute delay. A high gross activity release condition at this monitor (>1 Ci/sec) gives control room alarm annunciation. If the high radiation condition persists for longer than 15 consecutive minutes the off-gas isolation valve will close, reactor shutdown will be initiated immediately, and the reactor will be placed in cold shutdown within 24 hours. (See Table 3.2.D of the Technical Specifications.)

The ERP, reactor building ventilation, radwaste building ventilation, and turbine building ventilation noble gas monitors give control room alarm annunciation at high radiation levels. Each monitoring system has a gas monitor, an iodine sampler, a particulate sampler, and a flow rate measuring device. A separate noble gas monitor in the reactor building ventilation system will also isolate the reactor building and route the

effluent through the standby gas treatment system. The Licensee's RETS state that monitoring and sampling will be performed "during releases via this pathway" for all effluent monitors.

3.2.1.3 Liquid and Gaseous Instrumentation Setpoints

The setpoints for the radioactivity monitors at each release point are established to prevent exceeding concentrations in liquid releases or corresponding dose rates for gaseous releases of 10 CFR Part 20 in unrestricted areas. Figure 1.1 of the Technical Specifications shows the unrestricted area boundaries. The setpoints for the liquid and gaseous effluent instrumentation will be determined in accordance with the ODAM.

The Licensee's RETS submittal on liquid and gaseous effluent monitoring instrumentation and their corresponding setpoints have satisfied the provisions and meets the intent of NUREG-0473.

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 20.106 limits, as determined for reasonable resolution times for the measurements, and if the concentration of liquid effluents released to the unrestricted areas exceeds these limits, the cause will be attended to without delay and the concentration will be restored to within the above limits. Both batch and continuous releases are sampled and analyzed periodically in accordance with an acceptable sampling and analysis program.

Therefore, the Licensee's RETS submittal on liquid effluent concentrations meets the intent of NUREG-0473.

3.2.2.2 Gaseous Effluent Dose Rate

The Licensee's RETS include a commitment to maintain the offsite gaseous dose rates from the site to areas at and beyond the site boundary to within 10 CFR Part 20 limits, and if the dose rates due to gaseous effluents exceed these limits the release rate will be decreased to comply with the limits.

The radioactive gaseous waste sampling and analysis program provides for adequate sampling and analysis of the discharges. Therefore, the Licensee's submittal on gaseous effluent dose rates meets the intent of NUREG-0473.

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 the dose specifications of NUREG-0473, and are in accordance with 10 CFR Part 50, Appendix I and 40 CFR Part 190.

The Licensee's RETS include commitments (a) to maintain doses due to liquids effluents to within the NUREG-0473 quarterly and annual dose criteria, (b) to maintain noble gas air doses in unrestricted areas to within the NUREG-0473 quarterly and annual dose criteria, (c) to maintain the dose level due to release of iodine-131, iodine-133, and materials in particulate form with half-lives greater than eight days to within the NUREG-0473 quarterly and annual dose criteria, and (d) to limit the annual dose to any member of the public due to release 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-0473.

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 radwaste treatment system when the pre-release analysis indicates a radioactivity concentration (exclusive of tritium and dissolved noble gases) of 0.01 μCi/ml or higher. This trigger point for treatment is justified by a cost benefit analysis contained in the basis statements. The Licensee's RETS include a commitment to submit a special report within 31 days of the end of the quarter in which the limit was exceeded if radwaste requiring treatment was discharged without treatment.

The Licensee has committed to make every reasonable effort to operate at least one train of the charcoal adsorbers in the offgas treatment system whenever the main condensor is in operation except during startup or shutdown with the reactor operating at less than 10% of rated power or when the system cannot function due to low offgas flow. A commitment is also made to submit a special report if gaseous wastes are discharged for more than seven days without treatment.

The CNS RETS also require the ventilation exhaust treatment system to be operated to reduce the radioactive materials in gaseous waste prior to discharge when the projected dose, due to gaseous effluent releases via vent exhaust to unrestricted areas, averaged over 31 days would exceed 0.3 mrem to any organ.

Therefore, the Licensee's RETS submittal on effluent treatment meets the intent of NUREG-0473.

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.

There are no permanent undiked outside storage tanks at CNS. The Licensee has committed to keep the contents of any unprotected temporary outside storage tanks below 10 curies: and if the contents exceeds 10 curies (excluding H-3 and dissolved noble gases) to immediately suspend additions to the tank, begin measures to reduce the contents to 10 curies without delay, and describe the events leading to the condition in the next semiannual Radioactive Materials Release Report.

Therefore, the Licensee's RETS submittal on tank inventory limits meets the intent of NUREG-0473.

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. Flammability curves[45] show that if H_2 is less than or equal to four percent, oxygen can be at any concentration and a flammable mixture will not result.

The Licensee has committed to limit the concentration of hydrogen in the offgas system downstream of the recombiners to $\leq$ 2% by volume. The hydrogen concentration will be determined by use of two hydrogen monitors, or by one hydrogen monitor or recombiner temperature sensor with daily sampling and analysis for hydrogen within the ensuing four hours.

Therefore, the Licensee's RETS submittal on explosive gas mixtures meets the intent of NUREG-0473.

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 implement the requirements of 10 CFR Part 20, Section 20.301, and 10 CFR Part 71.[40]

The Licensee's RETS include a commitment to process wet radioactive wastes in accordance with a PCP to ensure the solid waste shall meet the requirements of 10 CFR Part 61.56[44] before shipping from the CNS site.

Therefore, the Licensee's RETS submittal on solid radioactive waste meets the intent of NUREG-0473.

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 CFR Part 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 Licensee's RETS for a radiological environmental monitoring program have followed in general the intent of the model RETS and the

Branch Technical Position on the subject issued November 1979, [41] as applicable to the site, and have generally provided an adequate number of sample locations for pathways identified. 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 obtaining the appropriate annual information for a BWR. 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 environment program meets the intent of NUREG-0473.

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 Station Operations Review Committee (SORC) and the Safety Review and Audit Board (SRAB) as the entities comparable to the Unit Review Group (URG) and the Company Nuclear Review and Audit Group (CNRAG), respectively.

The SORC is responsible for reviewing every unplanned release of radioactive material, and reviewing ODAM and PCP procedures at least once every 24 months. Review of these procedures effectively consitutues review of changes in the ODAM and PCP since any significant changes in these documents would be reflected in the procedures.

The SRAB is responsible for auditing the radiological environmental monitoring program and the ODAM and their implementing procedures at least every 24 months. (The audit frequency was implemented in 1969.) The SRAB audits the solidification procedures, which would reflect any significant changes in the PCP. The quality assurance program is reviewed by the SRAB. (The QA program requires that this review be performed at least every 12 months.)

The Station Operations Review Committee and Safety Review and Audit Board encompass the total responsibility for reviews and audits specified in NUREG-0473. Therefore, the Licensee's requirements for audits and reviews meet the intent of NUREG-0473.

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 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, ODAM and QA programs. The Licensee's existing technical specifications state that records of off-site environmental monitoring surveys will be retained for the life of the plant.

Therefore, the Licensee's RETS submittal on procedures and records meets the intent of NUREG-0473.

3.2.11 Reports

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

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

1. Environmental Program Data (Annual Report)

   The Licensee's RETS includes a commitment that this report will be submitted prior to May 1 of each year. It will include:

- A summary description of the radiological environmental monitoring program.

- A map and table of distances and directions of locations of sampling stations.

- A summary of the land use census.

- Results of analyses of samples required by the radiological environmental monitoring program.

- An assessment of radiation doses to a member of the public likely to be most exposed due to radioactive liquid and gaseous effluents released from CNS during the year.

- Results of participation in the Interlaboratory Comparison Program.

- Deviation from the environmental sampling schedule.

- A report of all analyses in which the required LLD is not achieved.

- A report of any changes in sample locations.

2. Radioactive Effluent Release Report (Semiannual)

A report of radioactive material released from the Station shall be submitted to the NRC within 60 days after January 1 and July 1 of each year. It will include a summary by calendar quarter of liquid, gaseous, and solid wastes released from the station. The summary is to be in the format recommended in Regulatory Guide 1.21, Appendix B, Tables 1 and 2.

The report will also include summary descriptions of any changes to the PCP or ODAM. A summary description of meteorological data collected during the year will be included in the

Semiannual Report submitted within 60 days after January 1 of each year. The report will include a description of each unplanned release of radioactive materials that causes a limit in Specification 3.21.B.1.a, 3.21.B.2.a, 3.21.C.1.a, or 3.2.1.C.3.a. to be exceeded.

3. Special Reports

The Licensee's RETS include a commitment to file a special report, within 30 days of the time specified in each Technical Specifications under the following conditions:

- Exceeding the liquid effluent dose limits according to Specification 3.21.B.2.b.

- Exceeding the gaseous effluent dose limits according to Specifications 3.21.C.2.b or 3.21.C.3.b.

- Exceeding the total dose limits according to Specifications 3.21.B.2.a, 3.21.C.2.a, and 3.21.C.3.a.

- Exceeding the reporting levels for the radioactivity measured in environmental sampling program according to Specification 3.21.F.3.

- When radioactive liquid or gaseous effluents are discharged without treatment according to Specifications 3.21.B.2.d or 3.21.C.4.c.

Therefore, the Licensee's RETS submittal on reports meets the intent of NUREG-0473.

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 changes in the ODAM and PCP shall become effective upon review and acceptance by the Station Operations Review Committee (SORC). Change(s) in the ODAM and PCP shall be submitted to the NRC by inclusion in the next Semiannual Radioactive Material Release Report for the period in which the change(s) was made effective.

The Licensee will transmit information concerning major changes to the Radwaste Treatment System to the NRC in the form of updates to the FSAR.

Therefore, the Licensee's RETS submittal for these administrative controls meets the intent of NUREG-0473.

3.3 Offsite Dose Calculation Manual

Cooper Nuclear Station has chosen to use the title "Offsite Dose Assessment Manual" (ODAM) for their document equivalent to the "Offsite Dose Calculation Manual" required by NUREG-0473. As specified in NUREG-0473, 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 ODAM 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 locations of samples in support of the environmental monitoring program are also needed in the ODCM.

3.3.1 Evaluation

The ODAM submitted was reviewed and determined to be generally in compliance with the requirements of NUREG-0133.

The Licensee's ODAM addresses the equation in the addendum of NUREG-0133 to determine the alarm and trip setpoints for the liquid effluent monitors. The setpoints of the radiation monitors on the liquid radwaste effluent line and the service water effluent line are determined so the concentration of radioactive materials at the point of release to unrestricted areas is within the 10 CFR 20, Appendix B, Table II, Column 2 limits without dilution, dispersion, or decay of radioactive material in the river.

The Licensee's ODAM contains methodology for comparing the radioactivity concentrations in liquid effluents at the point of release to the 10 CFR Part 20 limits.

The Licensee's ODAM described methods for assuring that noble gas discharges are within the NUREG-0473 dose rate limits of 500 mrems/yr to the total body and 3000 mrems/yr to the skin by correctly determining the setpoints for the noble gas monitors. Methods are described by which the dose rate due to the release of I-131, I-133, tritium, and particulates with half-lives greater than eight days is assured to be within the NUREG-0473 limit of 1500 mrem per year to any organ by calculating the dose rate to a person using the appropriate selection of age group and pathway. (The resolution times for these dose rate determinations will be 3 months for H-3, SR-89, and Sr-90, and 31 days for other radionuclides,) as required in Specification 3.21.C.1.b.)

The ODAM contains methodology for demonstrating compliance to 10 CFR Part 50, Appendix I by calculating the dose or dose commitment at least once every 31 days for liquid and gaseous effluents. The ODAM methodology includes a near-field mixing factor for calculating doses from the fish or drinking water pathway. The cumulative doses calculated are compared to the RETS dose limits to ensure the limits have not been exceeded.

The Licensee's RETS contain a cost-benefit analysis that justifies usage of the liquid radwaste treatment when pre-release analysis indicates a radioactivity concentration of 0.01 μCi/ml or higher. Thus, dose projections in the ODAM to determine when to use the liquid radwaste treatment equipment would be superfluous. Methods are described for calculating the projected dose due to gaseous effluents during the current quarter. Dose projections for the current year can be made by the method described for the dose projection for the quarter.

Specific parameters of distance and the direction of locations of all sampling stations for the environmental monitoring program are provided in the Table and Figure C-2 of Appendix C to the ODAM. The ODAM also contains maps showing the unrestricted area boundary and the location of the elevated release point and the discharge canal. A simplified diagram of the gaseous effluent streams, treatment and monitoring equipment, and discharge points is included as Figure 3-1 in the ODAM. No comparable diagram is included for the liquid radwaste system.

The Licensee's ODAM contains methodology for demonstrating compliance with 40 CFR 190. The fish consumption pathway will be included only if a significant increase in fishing downstream in the river near the station occurs during the previous twelve months.

The Licensee's ODAM is generally in compliance with the NRC guidelines and uses methods consistent with the methodology and guidance prescribed in NUREG-0133 to demonstrate compliance to the NUREG-0473 requirements.

4.0 Conclusions

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

- The Licensee's proposed RETS for the Cooper Nuclear Station submitted March 7, 1984 and supplemented by the submittal dated April 10, 1984, meets the intent of the NRC staff's "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors," NUREG-0473.

- The Licensee's ODAM dated January 1984 generally uses documented and approved methods that are consistent with the methodology and guidelines in NUREG-0133 with the following exceptions:

  a. The ODAM allows a near-field mixing factor for the fish and drinking water-pathway dose calculations.

  b. The fish consumption pathway will be included when demonstrating compliance to 40 CFR 190 only if a significant increase in fishing occurs during the previous twelve months.

  c. No diagram of the liquid radwaste system is included.

A correspondence between (a) NUREG-0473, (b) the Licensee's current RETS, and (c) the Licensee's proposed RETS is shown in Table 1.

TABLE 1. CORRESPONDENCE OF PROVISIONS OF NUREG-0473, THE LICENSEE'S CURRENT TECHNICAL SPECIFICATIONS AND THE LICENSEE'S PROPOSAL FOR COOPER NUCLEAR STATION

RETS Requirement	NUREG-0473	Licensee's Current Technical Specifications (Appendix B)(a)	Licensee's Proposed Technical Specifications (Appendix A)
Effluent (Liquid)	3.3.3.10	3.2.D,(App.B)2.4.1.b.3	3.21.A.1
Instrumentation (Gaseous)	3.3.3.11	3.2.D,(App.B)	3.21.A.2
Concentrations in Liquids	3.11.1.1	(App.B) 2.4.1.b	3.21.B.1
Dose Rate for Gases	3.11.2.1	(App.B) 2.4.3.a	3.21.C.1
Offsite Doses from Liquids	3.11.1.2	-----	3.21.B.2
Offsite Doses from Gases	3.11.2.2	(App.B) 2.4.3.a	3.21.C.2
Offsite Doses from Iodine-131, etc.	3.11.2.3	(App.B) 2.4.3.a	3.21.C.3
Total Offsite Dose	3.11.4	(App.B) 2.4.3.a	3.21.D.1
Liquid Radwaste Treatment	3.11.1.3	(App.B) 2.4.1.b.4	3.21.B.2
Gaseous Radwaste Treatment	3.11.2.4	3.7.B	3.21.C.4
Ventilation Exhaust Treat	3.11.2.5	-----	3.21.C.4
Tank Inventory Limits	3.11.1.4	(App.B) 2.4.1.b.5	3.21.B.3
Explosive Gas Mixtures	3.11.2.6	(App.B) 2.4.3.a.9	3.21.C.5
Main Condenser Effluent	3.11.2.7	-----	3.21.C.6
Mark I or Mark II Containment	3.11.2.8	-----	3.21.C.7
Solid Radwaste (PCP)	3.11.3	-----	3.21.E.1
Radiological Environmental Monitoring	3.12.1	(App.B) 4.4	3.21.F.1
Land Use Census	3.12.2	-----	4.21.F.2
Interlaboratory Comparisons	3.12.3	----	3.21.G.1
Audits and Reviews	6.5.1 6.5.2	6.2,(App.B)5.1	6.2,6.11,6.12
Procedures and Records	6.8, 6.10	(App.B) 5.5.6	6.3.1, 6.3.2.F, 6.6.2.D
Reports	6.9	(App.B) 5.4.1	6.5.1
Other Administrative Controls (PCP, ODCM, Changes to Radwaste Systems)	6.13 6.14 6.15	(App.B) 5.4.2.C	6.9 6.10 6.11

a) Being deleted.
b) Concentration in gases.
(c) Not required.

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NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC): EGG-PBS-6612

4. TITLE AND SUBTITLE
RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) IMPLEMENTATION - COOPER NUCLEAR STATION

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
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5. DATE REPORT COMPLETED
MONTH: May YEAR: 1984

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Neutron and Radiation Physics
EG&G Idaho, Inc.
Idaho Falls, ID 83415

DATE REPORT ISSUED
MONTH: May YEAR: 1984

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Systems Integration
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.

13. TYPE OF REPORT

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

A review of the Radiological Effluent Technical Specifications (RETS) for the Cooper Nuclear Station 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-0473, Revision 3, "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors." Draft submittals were discussed with the Licensee by both EG&G and the NRC staff 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 NRC review guidelines. The proposed Offsite Dose Calculation Manual was reviewed and generally found to be in compliance with the NRC review guidelines.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report): Unclassified

20. SECURITY CLASS (This page): Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (11-81)