ATTACHMENT 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

(2)

FEB 9 1984

MEMORANDUM FOR: Darrell G. Eisenhut, Director
Division of Licensing, NRR

FROM: Richard C. Lewis, Director
Division of Project and Resident Programs

SUBJECT: COMMENTS ON DRAFT APPENDIX A TECHNICAL SPECIFICATIONS, GRAND GULF UNIT 1, DOCKET NO. 50-416

Our comments on the second proof and review of the Grand Gulf Unit 1 Technical Specifications are enclosed. This review was performed on the Technical Specification provided in C. O. Thomas' letter dated October 4, 1983.

The comments contained herein were discussed with M. Houston and D. Hoffman of your staff during our recent meeting at the Grand Gulf site. Should NRR wish to discuss these issues further, we would be pleased to meet with you.

R.C. Lewis
Richard C. Lewis

Enclosure:
Comments on Grand Gulf Unit 1
Draft Technical Specifications

cc: D. M. Houston, LPM
Grand Gulf Nuclear Station

8404020087

CONTACT: C. Julian
242-5535

ENCLOSURE

SPECIFIC COMMENTS

Page	TS Item	Comment
3/4 5-1	3.5.1.C	Change to read "...taking suction from the condensate storage tank and...". It is our understanding that the CST is the normal water source for HPCS rather than the suppression pool.
3/4 5-6	3.5.2.e.1 & 2	Change to read "1. From the condensate storage tank, or 2. From the suppression pool when the condensate storage tank contains less than 170,000 available gallons of water."
3/4 5-5	4.5.1.e	Add the following: "The HPCS system shall be determined OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be operable."
3/4 8-9	3.8.1.2.b	Change to read "Diesel Generator 11 or 12, and...".
3/4 8-9	3.8.1.2.b.2.a	Change to read "48,000 gallons of fuel for each operable diesel generators 11 or 12".
3/4 8-9	3.8.1.2 ACTION a.	Change to read "with all offsite circuits inoperable or with diesel generators 11 and 12 inoperable, suspend...".
3/4 8-9	3.8.1.2 ACTION b.	Change to read "with diesel generator 13 inoperable, restore...".
3/4 1-9	4.1.3.3.b.2	Recommend that this surveillance be deleted. The Technical Specification, as written, provides no acceptance criteria nor required actions. During informal communication with NRR we have been told this was put in the TS to gather data to determine the need for a definitive TS on accumulator check valve leakage. It is our opinion that this specification is unnecessary.

3/4 3-45	Table 3.3.5-1	For initiation of RCIC, the number of minimum operable channels per trip system should be increased from 2 to 4 for Reactor Vessel Water Level Low Low Level 2. This change is needed to conform to plant design.
3/4 6-29	Table 3.6.4-1	MP&L letter AECM-83/0540 of September 12, 1983, identified 12 penetrations with associated containment isolation valves that should have been pneumatically tested but were tested hydrostatically. MP&L committed to test those valves pneumatically the next time and to request appropriate changes to the TS. We should take action to correct these TS deficiencies. MP&L letter AECM-83/0592 of September 20, 1983, to NRR identified 18 containment isolations that were pneumatically tested when the TS calls for hydrostatically testing them. Region II feels this is conservative since pneumatic testing is a more sensitive method; however, this is a deviation from their TS. The status of this item should be reviewed with the licensee to insure that the TSs are correct and that they were met. Notes (c) and (d) on page 3/4 6-29 appear to be identical and so both are not needed.
3/4 7-10	3.7.4	Our staff questions whether hydraulic snubbers and mechanical snubbers should be given separate functional test frequencies and acceptance criteria. The bases for the sampling program for functional testing is not clear. It is also not consistent with other recently approved TSs such as McGuire 2 and St. Lucie 2. We understand that MP&L has been informed recently of another sampling program TS now considered acceptable to NRR and may revise the snubber TS in its entirity. We urge that the NRC strive for some consistency in snubber TS for new plants and terminate the numerous revisions of the acceptable TS standard in the interest of regulatory uniformity. We observe that the Grand Gulf TS does not address hard to remove snubbers or the ALARA aspect of snubbers located in high radiation areas. Line 14 of paragraph b on page 3/4 7-10 says "part" and should read "port".

6-7 6-9	6.5.1.3 6.5.2.3	The wording of these statements should be clarified. It allows alternate members to be appointed in writing but does not specify their terms, qualifications, limitations, etc.
3/4 7-45	3/4.7.9	This specification is apparently intended to limit spent fuel pool temperature to less than 150°F. We assume that the intent is to prevent evolution of radioactive material from the pool water at elevated temperatures, as stated in the basis. It is not clear why the ACTION statement calls for reactor shutdown if the temperature limit is exceeded. We recommend that NRR confirm the intent of this specification and revise it to clearly state the intent. If intended, there should be a statement that TS 3.0.3 and 3.0.4 do not apply.
3/4 7-46	3/4.7.10	This TS on access road embankment stability should either be clarified or deleted. If the culvert is found to be blocked, no time frame is given in which it must be cleared. There should be a statement that 3.0.3 and 3.0.4 do not apply. Otherwise, culvert blockage can lead to a shutdown of the reactor which we do not believe is intended. There is no basis statement for this TS in the bases section.
3/4 1-16	3.1.4.2	The TS on Rod Pattern Control System should include a statement of when it is permissible to use the installed bypass switches to remove a particular control rod from the influence of RPCS. One violation has already been identified by Region II for improper use of these switches. A proposed new TS 3.1.4.3 is included below for your consideration.

PROPOSED NEW TS ON RPCS BYPASS SWITCHES

3.1.4.3 The individual rod position bypass switches may be activated only under the following conditions.

1. To reposition a misaligned control rod believed to be operable.
2. After performing action a.2 of 3.1.3.4.
3. After performing action a.3 of 3.1.3.5.
4. After performing action b of 3.1.4.2.
5. In conjunction with special test exception 3.10.2.

APPLICABILITY: OPERATING CONDITIONS 1, 2 AND 5.

SURVEILLANCE REQUIREMENTS

4.1.4.3 The position of each rod position bypass switch must be confirmed:

a. By independent verification by an SRO when the position is changed.

b. At least once per 24 hours by observing and recording switch position.

GENERAL COMMENTS

We are aware that in response to concerns raised over isolation of instrument air during containment isolation, the licensee has committed to NRR to install a pressure gauge on the instrument air supply as it enters containment and to read it periodically. We believe that such a commitment should be either a TS or license condition and that the surveillance frequency and acceptance criteria must be spelled out clearly, so that we can verify conformance.

All matters addressed above in our specific comments have been discussed with Mr. D. Hoffman, NRR during our recent meeting of 1/23-24/84 at the Grand Gulf site. We feel this meeting was very beneficial and we appreciate the cooperation shown by Mr. Hoffman in reviewing our comments on the Grand Gulf TS.

During our visit we received from the licensee a list of TS issues that they feel need clarification or revision. Certain of those items we feel need prompt attention to change the TS. We have selected from the licensee's list those that we feel need action and listed them below.

MP&L'S TECHNICAL SPECIFICATION AREAS OF CONCERN

Tech Spec	Discussion:
3.5.1 (ECCS-Operability)	Tech Spec requires at least 7 operable ADS valves in Division 1 and 2. However, the bases indicates that the ADS controls only 7 selected SRVs while the FSAR only takes credit for 6; thus allowing one valve out-of-service for 14 days. It thus appears that the Tech Spec should require 8 operable ADS valves since ADS controls 8 valves, not 7, and the FSAR assumes loss of one ADS valve.
4.6.6.3.d.3.c.d (SGTS Surveillance)	Surveillance deals with SGTS initiation on Fuel Handling Area Pool Sweep and Ventilation Exhaust Radiation High. The correct terminology should be radiation high-high instead of just high.
3.3.2-Table 3.3.2-1.4.h Item 4.h (Isolation Actuation Instrumentation)	The table lists the minimum operable channels per trip system as 'NA' for RWCU isolation from a SLCS initiation. Hence, the action statements for the Tech Spec does not really apply and the channel can remain out-of-service indefinitely with no action required. This should be corrected.

3.3.6 (Control Rod Block Inst.) 3.3.7.6 (SRM) 3.9.2 (Refueling Inst.)	The action statement requires different numbers of SRMs operable during same modes of operation. 3.3.6 requires 4 SRMs during Ops. Conds. 2,5; 3.3.7.6 requires 3 SRMs during Ops. Conds. 2,3,4; while 3.9.2 requires 2 SRMs during Ops. Conds. 5. TS needs to be made consistent.
3.3.7.7 (TIP System)	The Tech Spec only requires 3 TIPs operable to map the core and is required only when recalibrating LPRMs. In order to calibrate all LPRMs and map the whole core, 5 TIPs with indexing equipment is required. The TS should be changed to require 5 TIPs, or deleted. The requirement to calibrate LPRMs suffices and the TIP TS is not necessary.
3.3.6 Table 3.3.6-1.4.a (Control Rod Block Instrumentation)	The superscript (d) is not correct. The IRM detector not full in Control Rod Block is not bypassed on range 1 of the IRMs. It is, however, bypassed when the associated detectors are bypassed or the Reactor Mode Switch is in run (LIC NS-83125).
3.6.3.3.b (Suppression Pool Cooling)	Tech Spec deals with a flow path from Suppression Pool through a SSW heat exchanger. There is no SSW heat exchanger; it should read a RHR heat exchanger.
3.3.2 All Tables (Isolation Actuation Instrumentation)	The associated Tech Spec Tables deal with MSIV closure on HI RAD; however, MSL RAD INOP also gives an MSIV closure. This is not the only case. Do all isolation signals need to be identified? (Another example is RCIC isolation high temp.) We need to closely review the RPS and isolation systems to ensure that all automatic functions credited in the FSAR are required by TS.
4.1.3.1.4.b (Control Rod Operability) Scram Discharge Volume	Tech Spec surveillance verifies proper level sensor response by the performance of a Channel Functional test. However, the definition of a Channel Functional does not include the sensor. IE Bulletin 80-17 required all BWRs to begin monthly functional tests of the SDV which actuates the sensor and verify the desired result. Grand Gulf TS should be clarified to impose a comparable test requirement.

3.3.8 (CTMT and Drywell Press Setpoints)	Tech spec should be changed to take into account barometric pressure influence on containment pressure. Licensee has determined that installed instruments are absolute and not relative, thus changes in barometric pressure can cause nonconservative setpoints. Change needs to be made to Tech Spec concerning CTMT press initiation setpoint which may be wrong for containment spray, as identified by licensee.
4.6.7.3.b.1 (Drywell Purge)	Surveillance requires reverifying a flow rate of 1000 cfm every 18 months. However, the flow rate should be considered in scfm since the flow rate is temperature dependent.
4.4.4.c (Chemistry)	Surveillance specifies action to be taken in the event the continuous conductivity recording monitors are inoperable for up to 31 days. What action must be taken at the end of the 31 day INOP period. TS language should be clarified.
6.15 (Major changes Radwaste reporting requirements)	What constitutes major Radwaste changes? TS language should be clarified.
3.3.4.1 (Recirc pump trip ATWS)	The associated action statements (3.4.4.1.b and c) are not consistent with the system design. For example, if one trip channel is INOP, then the INOP channel is supposed to be tripped; this action trips one recirc pump. However, if two channels are INOP (depending on channels) then the trip system may be declared INOP and you have 72 hours to declare it operable. This is a prominent case in which the TS was written without accurate knowledge of the actual system design.
3.4.2.1 & 3.4.2.2 (SRVs)	The Tech Spec action statements seem to address only one trip system of relief logic and Lo-Lo set function logic. However, the plant design provides (two) redundant trip systems; hence, Tech Specs may be more restrictive than necessary.

3.4.1.2 (Jet Pumps)	Tech Spec requires Jet Pumps to be operable in Ops Conditions 1 and 2; however, the jet pumps are not required to be demonstrated operable until prior to exceeding 25% thermal power per surveillance 4.4.1.2. If it is acceptable to change modes during startup and wait until 25% power to perform the surveillance, then the TS should say so.
3.4.6.1 (Rc. Coolant System) Pressure - Temperature Limit Curve	The Tech Spec and associated surveillance requires reactor coolant system temperature and pressure to be in accordance with associated figures. The figure denotes RPV metal temperature. Does the statement reactor coolant system temperature address the monitoring of reactor coolant temperature or RPV metal temperature? Also, Figure 3.4.6.1-1 is ambiguous in that it does not label acceptable and unacceptable regions. Is BB' the B curve referred to in the TS verbage?
4.8.1.1.2.C (A. C. Sources)	ASTM-D270-1975 should be ASTM-D270-1965 (Reapproved 1980).
6.10.2.1 (Records)	Numbers for Table 3.7.5-1 and 3.7.5-2 are incorrect references. Should be 3.7.4-1 and 3.7.4-2.
3.4.3.2-Table 3.4.3.2-1 (RCS-Leakage)	This item is included with the generic RCS leakage spec implementation problems; however, it includes some specific problems with the table: 1. RCS Pressure Isolation Valves does not address all specific valves (i.e., E12-F041 should be E12-F041 A, B, C; E12-F042 should be E12-F042 A, B, C; E12-F050 should be E12-F050 A, B; E12-5053 should be E12-F053 A, B).
Miscellaneous	Tech Specs makes references to redundant channels in trip system. What is the definition of a redundant channel in a trip system? In addition, Tech Spec allows a channel to be in an inoperable status (during testing) without placing trip system in the tripped condition provided at least one other operable channel in the same trip system is monitoring that parameter. What exactly does this mean? This needs clarification throughout the TS.

3.6.4-Table 3.6.4-1 (CTMT/DRWL Isol Valves)	Tech Spec Table had .losing times of 60 seconds for 1E61-F003A,B and 84 seconds for E61-F005. This is not consistent with the 30 second blowdown criteria assumed during a LOCA.
Table 1.2 (Operational Conditions)	In order to perform neutron monitoring channel functional tests (SRMs, IRMs, APRMs), the reactor mode switch needs to be placed in the refueling position while in Ops Conditions 3 and 4. TS does not currently allow this and should be changed.
3.4.3.1, 3.4.3.2 (RCS-Leakage)	RCS leakage Tech Spec implementation problems 1) Spec list channel checks requirement for drywell air cooler condensate flow instrument loop, but gives no instructions if loop is inop; and 2) the bases includes the leakage through the high/low pressure interface valves as identified leakage. Hence, this leakage must be included in the total leakage rate allowed if this is intended.
3.3.8 (CTMT Spray)	CTMT Spray Timer Actuation Setpoint. On hold pending resolution of CTMT Spray Timer issue. Licensee has identified and reported that with worst case tolerances, installed equipment may not meet the present TS.
3.3.2-Table 3.3.2-1 (Isolation Instrumentation)	Changes to Rosemont Trip Unit and Riley Temp Switch Calibration frequency. Surveillance frequencies need to be shortened to be consistent with vendor recommendations.
4.3.7.1-Table 4.3.7.1-1 (Radiation Monitoring)	Surveillance list calibration frequency for Carbon Bed Vault Radiation Monitor as 18 months. Manufacturer recommends 12 months.
4.3.7.2-Table 4.3.7.2-1 (Seismic Monitoring)	Change associated Table to reflect instruments (total 3) deleted and ranges changed according to Design Change Package.
3.3.2-Table 3.3.2-1 Isolation Instrumentation)	E61-F009,57,10, & 56 receive Group 5 signals. Clarifying note should be placed here or on Table 3.6.4-1.
3.4.1.3 & Bases (Recirc Loop Flow).	Change recirculation flow to Jet Pump flow and rated core flow to EFF. Core Flow.

3.3.3-Table 3.3.3-2 (LOP-Division 3) Item D.2	Bechtel Engineering commented that contrary to TS, offsite power trips without a time delay when Division 3 voltage drops to 3045 volts. Table 3.3.3-2 should be revised.
3.4.4 (Chemistry)	It is not clear if the 336 hrs/year is to be taken individually or jointly for conductivity and chloride. If read verbatum, it tends to indicate that you can go for 336 hrs/year when both chloride and conductivity are above TS limit and no time limit exists, if only one exceeds the limit. TS should be changed to clarify.
4.4.1.1.a & b (Recirc Loop FCV Movement)	Fail "as is" should be clarified to mean does not drive in either direction. Changing "Average" to "Maximum" would clarify surveillance with respect to what is actually measured.
4.11.1.1.2 (Liquid Effluents Concentration)	If post release-analysis shows concentrations greater than MPC (10 CFR 20 Table II, Col. 2, App. B), ACTION under 3.11.1.1 is impossible. TS language should be clarified.
3.4.3.2-Table 3.4.3.2-1	2) RCS Interface Valves: a) F51-F064 should be E51-F064 b) The alarm setpoint of -480 for RCIC should be NA, as there is no alarm for RCIC.
3.3.7.12-Table 3.3.7.12-1 (Radioactive Gaseous Effluent Monitoring)	Associated Table includes some instrument channels for which the alarm/trip setpoints are not calculated in accordance with ODCM.
4.3.4.2.3 (EOC Recirc Pump Trip)	Surveillance requires response time testing of trip functions for Turbine Stop Valve and Turbine Control Valve closures every 18 months, while the next sentence states that each test shall include at least the logic of one type of channel input (turbine control valve fast closure or turbine stop valve closure) such that both are tested every 36 months. This should be clarified.
3.2.2 (APRM Setpoints)	Change ACTION to agree with STS (6 hours vs. 2 hours). Grand Gulf TS appears to be more restrictive than that of other BWRs.

4.3.7.7 (TIP)	Resolve use of TIP and process computer for monitoring functions. Surveillance is inappropriate.

ATTACHMENT 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

MAR 13 1984

Mississippi Power and Light Company
ATTN: Mr. J. B. Richard
Senior Vice President, Nuclear
P. O. Box 1640
Jackson, MS 39205

Gentlemen:

SUBJECT: REPORT NO. 50-416/84-06

On February 21-24, 1984, NRC inspected activities authorized by NRC Operating License No. NPF-13 for your Grand Gulf facility. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed inspection report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

Within the scope of the inspection, no violations or deviations were identified.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in NRC's Public Document Room unless you notify this office by telephone within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of the letter. Such application must be consistent with the requirements of 2.790(b)(1).

Should you have any questions concerning this letter, please contact us.

Sincerely,

David M. Verrelli

David M. Verrelli, Chief
Project Branch 1
Division of Project and
Resident Programs

Enclosure:
Inspection Report No. 50-416/84-06

cc w/encl:
J. E. Cross, Plant Manager
Ralph T. Lally, Manager of Quality
Middle South Services, Inc.

~~8404020091~~

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report No.: 50-416/84-06

Licensee: Mississippi Power and Light Company
Jackson, MS 39205

Docket No.: 50-416

License No.: NPF-13

Facility Name: Grand Gulf 1

Inspection at Grand Gulf site near Port Gibson, Mississippi

Inspectors: C. Julian for 3/13/84
S. Butler Date Signed

C. Julian for 3/13/84
J. Caldwell Date Signed

C. Julian for 3/13/84
R. Carroll Date Signed

C. Julian for 3/13/84
M. Hunt Date Signed

C. Julian 3/13/84
C. Julian Date Signed

C. Julian for 3/13/84
H. Krug Date Signed

C. Julian for 3/13/84
N. Merriweather Date Signed

C. Julian for 3/13/84
W. T. Orders Date Signed

C. Julian for 3/13/84
A. Wagner Date Signed

Approved by: [signature] 3/13/84
D. M. Verrelli, Branch Chief Date Signed
Division of Project and Resident Programs

~~8404240495~~

SUMMARY

Inspection on February 21-24, 1984

Areas Inspected

This special announced inspection involved 234 inspector-hours on site in the area of verification of the accuracy of the Technical Specification.

Results

Of the areas inspected, no violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. E. Cross, Plant Manager
   *R. F. Rogers, Assistant Plant Manager - Operations
   *C. R. Hutchinson, Assistant Plant Manager - Maintenance
   *J. W. Yelverton, Assistant Plant Manager - Support
   *J. C. Roberts, Technical Support Staff
   *F. M. Walch, Maintenance Superintendent
   *G. A. Zinke, Technical Engineering Supervisor
   *L. F. Daughtery, Compliance Superintendent
   *J. D. Bailey, Compliance Coordinator

   Other licensee employees contacted included numerous engineers, operators, mechanics, security force members, and office personnel.

   Other Organizations

   *M. G. Farschon, General Electric Site Operations Manager

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on February 24, 1984, with those persons indicated in paragraph 1 above. The Technical Specification (TS) discrepancies were described to plant management by the inspectors. NRC representatives stated that the problems found are indicative of the need for another review of Technical Specifications to find and correct any errors.

3. Licensee Action on Previous Enforcement Matters

   Not inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Suppression Pool and Containment Spray

The inspectors compared applicable sections of the Final Safety Analysis Report (FSAR), as-built drawings, surveillance, and operating procedures and actual plant systems to Technical Specifications (TS) associated with the suppression pool and containment spray. The following are discrepancies that were identified:

a. FSAR Section 6.2.7.5 indicates that the suppression pool level indication system is made up of four level detector channels, (two detector channels per division). It also indicates that each of these channels provides a high-water-level alarm, low-water-level alarm, low-low-water-level alarm, as well as a signal to open suppression pool makeup valves.

   In actuality, there are three active level detector channels per division. Two channels are wide range and one channel is narrow range. There is also one additional channel per division which is only used for indication at the remote shutdown panel. Each wide range channel supplies input to their respective division's suppression pool makeup system in one out of two logic as well as providing a low-low-level alarm at 16'10". The narrow range channel in each division provides the divisional low-level and high-level alarms (18'5½" and 18'9", respectively). TS are written to conform to the FSAR but are not in clear agreement with the actual plant design.

b. TS 3.5.3 (ECCS), 3.6.3.1 (Depressurization Systems), 3.3.7.5 (Accident Monitoring Instrumentation), and 3.6.3.4 (Suppression Pool Makeup) all relate to required operability of suppression pool and level instrumentation. They do not recognize the difference between narrow and wide ranges and therefore do not identify what level detector channel is to be used to meet the TS operability requirement. As a result, divisional operability is left to the interpretation of the reader in the action statements as well as in the surveillance requirements.

c. In none of the above listed TS is the level instrumentation required to initiate automatic suppression pool makeup addressed as a requirement for suppression pool operability. This is an accident mitigation function and should have an associated surveillance. It would logically follow that at least TS 3.6.3.4 (Suppression Pool Makeup) should include the wide range level instrumentation as part of its operability requirement and a surveillance should be included. In TS 3.3.7.5 (Accident Monitoring Instrumentation) only two suppression pool level detectors are required, and a seven day Action Statement applies if only one is available. In reality, it appears this should read that two wide range level channels per division are required, minimum channels operable per division is one, and if only one division is operable, then the 7-day Action Statement applies.

d. By annotating what level detectors are required in the daily operating log and surveillance procedures, the license has made an effort to compensate for these unclear technical specifications. In spite of this, some problems were observed. The daily operating log indicates that for operability statement "a" of TS 3.5.3, the narrow range level detectors are to be used to verify that suppression pool level is ≥ 18'4 3/4" (Condition 1, 2, or 3); for operability statement "b", the wide range level detectors are to be used to verify that suppression pool level is ≥ 12'8" (Condition 4 or 5) since narrow range indication does not go down this far. However, wide range is calibrated for post accident temperature (170°F), thereby indicating approximately 3" higher suppression pool level than what is actually present under normal conditions. The licensee has agreed to resolve this temperature calibration issue. This will be identified as Inspector Follow-up Item (IFI) 416/84-06-01.

e. Furthermore, since the wide range indication is utilized by the licensee in conditions 4 or 5, a channel calibration per surveillance requirement 4.5.3.1.b.3 (ECCS) is required. A channel calibration is performed by surveillance procedure 06-IC-IE30-R-0001, but only for surveillance requirements 4.3.7.5 (Accident Monitoring Instrumentation) and 4.6.3.4.c (Suppression Pool Makeup). The fact that this surveillance procedure does not recognize surveillance requirement 4.5.3.1.b.3 (ECCS) further demonstrates the need for individual level instrumentation identification in these associated suppression pool TS. It is also important that all TS relating to the suppression pool cross reference each other. As it stands now, only 3.6.3.1 and 3.5.3 reference one another. The FSAR states that the level sensors are spaced 90 degrees apart around the pool. Actually, the two groups of sensors are spaced 180 degrees apart.

f. TS 4.5.3.1.a.2 contains an apparent typographical error. The licensee noted that a change request has been prepared to surveillance requirement 4.5.3.1.a.2 (ECCS) to indicate suppression pool level as 12'8", in lieu of 12'5" (IFI 416/84-06-02).

g. TS 3.6.3.1 (Depressurization Systems) and 3.3.7.5 (Accident Monitoring Instrumentation) specify suppression pool temperature requirements. There are actually installed 24 temperature detectors/alarms, (2 divisions with 6 pairs per division). The suppression pool is azimuthally divided into six sectors, with two pairs (one pair per division) in each sector. By licensee designation, 12 of these detectors are used to meet TS 3.6.3.1 and the other 12 are used to meet TS 3.3.7.5. Consequently, only 12 channels undergo the channel functional test required by surveillance requirement 4.6.3.1.c (Depressurization Systems). 4.3.7.5 (Accident Monitoring Instrumentation) does not require a functional test. Neither TS indicates which temperature channels are to be used; therefore, leaving divisional/sector operability to the interpretation of the reader in the Action Statements as well as in the surveillance requirements. In fact, surveillance requirement 4.6.3.1.c implies that you can use any 12 temperature channels as long as there are two channels in each sector. Since only 12 channels receive functional testing, only these 12 should be credited by TS.

TS Table 3.3.7.5-1 apparently should state as "required number of channels" 12,2/sector rather than the present 6,1/sector. Then the present statement of 6,1/sector for "minimum channels operable" would allow operation for up to 7 days in an Action Statement.

It was further observed in TS 3.6.3.1 that the combining of action statements on suppression pool level and temperature instrumentation with the use of "and/or" was very ambiguous.

h. TS 3.6.3.2 (Containment Spray) contains an error and the licensee stated that a change has been prepared to operability statement 3.6.3.2.b to indicate the use of a "RHR" heat exchanger, in lieu of a "SSW" heat exchanger (IFI 416/84-06-03). Another inconsistency in surveillance requirement 4.6.3.2.b was pointed out by the inspectors.

In order to demonstrate operability of containment spray, this surveillance requires verification that each RHR pump develops a flow of at least 5650 GPM while recirculating water through the RHR heat exchanger to the suppression pool. This is accomplished by surveillance procedure 06-OP-1E12-Q-0023, where the same recirculation flow verification is used to determine LPCI and suppression pool cooling operability. However, surveillance requirements 4.5.1.b.2 (LPCI) and 4.6.3.3.b (Suppression Pool Cooling) specify a recirculation flow of at least 7450 GPM. The FSAR indicates that containment spray flow emitting from the spray nozzles into the containment is 5650 GPM. This would imply that a RHR pump flow capability of 7450 GPM is reduced to 5650 GPM after passing through the piping and containment spray nozzles. Therefore, one would suspect that surveillance requirement 4.6.3.2.b (Containment Spray) should also require a RHR recirculation flow acceptance criteria of at least 7450 GPM. At the time of the inspection, the licensee was unable to provide their spray flow analysis to justify the lower RHR recirculation flow of 5650 GPM.

In regards to an inoperable train, Action statements of TS 3.5.1 (LPCI) and 3.6.3.3 (Suppression Pool Cooling) allow for seven day continued operation when only one train is available. Since containment spray is more important, i.e., has less redundancy, action statement 3.6.3.2.a (Containment Spray) only allows a 72 hour Action period when one train is inoperable. A review of the RHR Pump operability data sheets in surveillance procedure 06-OP-1E12-Q-0023 revealed an allowance of 96 hours to analyze test results. In essence, this allows an additional

96 hours possible delay to the Action periods discussed above. Licensee representatives agreed to review this matter and make appropriate changes to the surveillance procedure (IFI 416/84-06-04).

6. a. TS 3/4.6.6.2 - Secondary Containment Automatic Isolation Dampers/Valves

The inspector compared the TS list of automatic valves and dampers to the licensee's surveillance procedures. The completed results of the most recent surveillance on secondary containment isolation were reviewed to see that the valve lists and identification are compatible. Approximately 5% of the valves and dampers were examined in the plant by the inspector. No discrepancies were identified.

The inspector asked licensee representatives if an actual plant walk-down had been conducted by the licensee to verify the accuracy of the TS lists of primary, drywell, and secondary valves. Licensee representatives stated that walkdowns were done at various times to resolve specific questions, but no comprehensive effort could be identified which had as its objective the verification of the TS tables. The inspector stated that, although this is not a regulatory requirement, it would seem to be a prudent action to confirm TS accuracy. Licensee representatives agreed to consider further action (IFI 416/84-06-05).

b. TS 3/4.6.6.3 - Standby Gas Treatment

The inspector reviewed the surveillance procedure for SGTS. The TS surveillance requirements and the implementing procedures appear adequate to ensure SGTS reliability. The inspector walked down the majority of the SGTS hardware in the plant to ensure that the hardware is compatible with the TS. No discrepancies were observed.

c. TS 3/4.6.7.1 - Hydrogen Recombiner

The inspector examined the two hydrogen recombiner systems installed in the plant to ensure compatibility with the TS. The completed results of the preoperational tests of this equipment were reviewed to ensure that the recombiners are capable of performance described in the surveillance section of the TS. No discrepancies were observed.

7. TS 3/4.8 - Emergency Power Supplies

The inspectors selected several sections of the TS and the corresponding surveillance procedures for examination to verify the adequacy of the procedures and the TS as they relate to the existing equipment. The following TS and surveillance procedures were examined and evaluated.

TS Sections

3/4.8.1	AC Sources - Operating
3/4.8.2	DC Sources - Operating
3/4.8.3	Onsite Power Distribution Systems (Operating)
3/4.8.4	Electrical Equipment Protective Devices
	Primary Containment Penetration Conductor Protective Devices
	Motor Operated Valve Thermal Overload Protection
	Reactor Protection System Electric Power Monitoring

Surveillance Procedures

06-OP-1R20-W-0001	Plant AC and DC Electrical Power Distribution Weekly Lineup
06-EL-1L51-R-0001	125V Battery Charger Capability Test
06-IC-1C71-SA-1001	RPS Electrical Protection Assembly Channel Functional Test
06-EL-1C71-R-0012	RPS Electrical Protection Assembly Calibration
06-EL-1L11-O-0001	125V Battery Capacity Discharge Test
06-EL-1R65-Q-1001	MOV Thermal Overload Protection Device
06-EL-1R65-R-0001	MOV Thermal Overload Protection Device

As a result of this review, the following discrepancies were identified:

a. Surveillance procedure 06-EL-1L51-R-0001 appears to be inadequate in that the battery chargers are never tested at the equalizing voltage (140 VDC ± 1 volt). The chargers are only tested at 105 volts at 400 amperes for two (2) hours. In addition, in the battery discharge test, there is no time limitation specified for when the batteries must be recharged to full capacity (IFI 416/84-06-06).

b. An apparent typographical error was found in TS Table 3.8.4.2-1. The B designation was omitted from valve number QSP415189B. Licensee representatives have since stated informally that the TS is correct. This will be confirmed during a future inspection.

c. TS requirement 3.8.4.3 appears to be inappropriate for the way the RPS electrical power monitoring assemblies (EPAs) are designed. Two EPAs are in series which means both units must be operable to supply power to the RPS bus. The Action statement in the TS requiring only one (1) EPA unit to be restored to service when two are inoperative does not seem appropriate for the circumstance. There is no provision for manual bypass of the individual EPA units.

   Surveillance procedure 06-IC-1C71-SA-1001 appears inadequate in that it only requires testing of the EPAs that are not providing power to the Reactor Protection System (RPS) bus. The procedure does not assure that the EPAs associated with the normal power supply (MG sets) will be tested during the six month surveillance test as required by TS Section 4.8.4.3.a (IFI 416/84-06-07).

The NRC inspectors also performed walkdowns of the systems identified above to randomly verify that equipment described in the TS was actually installed in the plant. All equipment examined in the plant was found to be properly identified in the TS with the exception of the items discussed above.

8. ECCS Systems and Actuation Instrumentation

a. TS 3/4.5.1 ECCS - Operating

The requirements for ADS operability contained in paragraphs 3.5.1.a.3 and 3.5.1.b.2 were reviewed. The TS paragraphs require "at least 7 operable ADS valves". This number appears to be incorrect. The Safety Evaluation Report page 6-22 states that the ADS employs eight of 20 SRVs. The action statement paragraph e.1 allows the operation up to 14 days with only six ADS valves operable, and up to 12 hours with five or less ADS valves. This appears to be an unacceptable TS (IFI 416/84-06-08).

A review of the paragraph 4.5.1.b pump testing criteria was conducted. Significant inconsistencies were noted in the pump flow characteristics for the following pumps. The TS for High Pressure Core Spary requires at least 7115 gpm with 182 psid, while the SER page 6-21 states 7115 gpm with 540 psid, and the FSAR Figure 6.3-2 lists 7115 gpm with approximately 387 psid. The TS for Low Pressure Core Spray requires at least 7115 gpm with 261 psid, while the SER page 6-22 states 7115 gpm with 340 psid and the FSAR lists 7115 gpm with approximately 311.6 psid. The TS required flows appear considerably less conservative than either the SER or FSAR (IFI 416/84-06-09).

b. TS 3/4.3.3 Emergency Core Cooling System Actuation Isolation

The inspector verified the incorporation of the following instrument surveillances of TS Tables 3.3.3-1, 3.3.3-2, and 4.3.3.1-1 into the plant's surveillance program.

LPCI Pump A Start Time Delay Relay
ADS Times
Drywell Pressure High
Reactor Vessel Water Level - Low, Low, Level 2

The following surveillance procedures were reviewed to ensure the required TS frequencies, trip setpoints, and allowable values were correctly incorporated.

06-IC-1B21-R-0012, Rev. 22, Reactor Vessel Water Level Calibration
06-IC-1B21-M-1010, Rev. 21, TCN9 Reactor Vessel Water Level (HPCS)
06-EL-1B21-M-0001, Rev. 21, TCN3 ADS Times Functional Test and Calibration

06-OP-1000-D-0001, Rev. 20, TCN27 Daily Operating Log (Items 64 & 16)
06-IC-1B21-R-0009, Rev. 21, TCN3 Drywell High Pressure Calibration (ECCS)
06-IC-1B21-M-1011, Rev. 20, TCN4 Drywell High Pressure (HPCS) Functional Test
06-EL-1E12-M-0001, Rev. 22, RHR Pump Start Time Delay Relay Functional
06-EL-1E12-M-0001, Rev. 21, RHR Pump Start Time Delay Relay Calibration

The following Drywell Pressure Transmitters and Reactor Vessel Level Transmitters were reviewed for proper field installation in accordance with the as-built drawing and the piping and instrumentation diagrams. Logic diagrams were reviewed for actuation of the appropriate equipment. No deficiencies were noted.

Division I	PT	NO94A
Division I	PT	NO94E
Division II	PT	NO94B
Division II	PT	NO94F
Division III	PT	NO67C
Division III	PT	NO67G
Division III	PT	NO67L
Division III	PT	NO67R

Division I	LT	NO91A
Division I	LT	NO91E
Division II	LT	NO91B
Division II	LT	NO91F
Division III	LT	NO73C
Division III	LT	NO73G
Division III	LT	NO73L
Division III	LT	NO73R

The licensee has previously identified a problem on instruments which utilize atmospheric pressure as one side of a differential pressure detector instrument. Due to a possible low atmospheric pressure condition around the plant, certain detectors may be as much as .5 psig nonconservative. This includes drywell pressure and containment spray. The licensee has not yet submitted all the appropriate changes at this time (IFI 416/84-06-10).

9. Drywell and Primary Containment Integrity

An inspection was performed of the following sections of the Grand Gulf TS:

SECTION	SUBJECT	PAGES
3/4 6.1.1	Primary Containment Integrity	3/4 6-1
3/4 6.1.2	Containment Leakage Rates	3/4 6-2, 3, 4

3/4 6.1.3	Containment Air Locks	3/4 6-5, 6
3/4 6.1.4	MSIV Leakage Control System	3/4 6-7
3/4 6.1.6	Containment Structural Integrity	3/4 6-9
3/4 6.1.7	Containment Internal Pressure	3/4 6-10
3/4 6.1.9	Containment Purge System	3/4 6-12

Emphasis was placed on the following specifics:

(1) Literal correspondence between the TS, and the installed hardware configuration.

(2) Adequacy and completeness of the surveillance requirements.

(3) Review of associated surveillance procedures and results generated by their execution.

(4) Adequacy and completeness of the Action Statements.

(5) Familiarity of licensee personnel with the TS and the associated hardware systems and testing requirements.

The following discrepancies were identified:

a. Surveillance Requirement 4.6.1.4 concerns the operability of each MSIV leakage control subsystem. Item C address the functional testing of the subsystem heaters but does not acknowledge that there are no heaters on the outboard subsystem; whereas there are four on the inboard subsystem. Section 6.7 of the FSAR reveals that no heaters are required in the outboard system, which is not identical to the inboard system in a number of aspects. Additionally, licensee surveillance procedures accurately reflect the existing hardware configuration. Clarification of the wording of the TS will resolve the ambiguity (IFI 416/84-06-11).

b. Surveillance Requirements 4.6.1.1.a requires a leak rate retest of the equipment hatch seals every time each penetration subject to a Type B test, except the containment air locks, is reclosed.

   This is not what was intended as it would require a retest of the equipment hatch seals following the opening of Type B pentration areas such as:

   (1) electrical penetrations

   (2) ECCS test return line orifice plate

   (3) fuel transfer tube

Surveillance Requirement 4.6.1.1.b is vague as to what must be secured in position, and how. Correction of the wording in the TS will resolve these ambiguities (IFI 416/84-06-12).

10. TS 3/4.3.2 Isolation Actuation Instrumentation

TS 3/4.3.2 Isolation Actuation Instrumentation was reviewed to determine if the requirements entailed therein are clear, if the LCOs are realistic, if the channels and trip systems appear technically adequate, if there are procedures for performing the surveillances, and if those requirements can be performed.

Seventeen applicable surveillance procedures were analyzed for technical adequacy and incorporation of TS acceptance criteria. Ten channels and associated trip systems were analyzed through examination of electrical prints, logic diagrams, and system descriptions for technical adequacy. The review revealed that TS 3/4.3.2 appeared technically adequate, the requirements realistic, and there were procedures for performing those selected requirements reviewed.

The procedures reviewed, with those exceptions to be detailed, appeared adequate. The procedures reviewed included but were not limited to:

06-IC-1B21-M-1004

06-IC-1B21-M-1010

06-IC-1C71-M-0001

06-IC-1B21-M-1004

06-IC-1E31-M-0003

06-OP-1000-D-0001

06-IC-1E31-M-1001

06-OP-1G33-M-0002

06-IC-1E31-M-0023

06-IC-1321-M-1003

Of those procedures, some discrepancies were identified in operation surveillance procedure 06-OP-1000-D-0001, which entails the operations' semi daily surveillance (channel checks) as required by TS Table 4.3.2.1-1. The item numbers, detailed below refer to the TS table specific requirement, by line number. The discrepancies are as follows:

a. With regard to items 1.b through 1.g and 5.m it was observed that the procedure line item (as referred to by the TS cross reference index) does not conform to the TS requirement. This appears to be an error in the TS cross reference document.

b. Line item 63 of procedure 06-OP-1000-D-0001 stipulates a channel check of slave trip unit B21-LS-693 to level instrument B21-LIS-N691 at $\leq$ 53.5". The slave trip unit referenced on the semi-daily surveillance sheet has undergone a station modification such that it no longer is slave unit to instrument B21-LIS-N691, as the procedure indicates, but now is slave to instrument B21-LIS-N695. There are three problems associated with this issue, of serious concern.

    (1) The station modification was completed, according to the licensee, in December 1983; however, the procedure has yet to be changed to reflect the modification.

    (2) The operations staff, if they were to be following the procedure, would be noting in mode 4 that slave unit B21-LS-693 is not in alarm as it should be since the master trip unit B21-LIS-N691 is tripped and is the procedure referenced master trip for that slave. They are not noting this fact.

    (3) The operations staff has been ignoring the procedural requirements simply because they know the modification, discussed above, is installed and as such that slave trip unit should not be in alarm. The operations staff however, has not in the period since December initiated a procedure change to reflect the station modification.

c. Further, TS item 4.3.2.1-1.f requires that a semi-daily channel check be performed on high drywell pressure, ECCS, Division 3. The TS cross reference index refers to line item 15 of procedure 06-OP-1000-D-0001 for the performance of that required surveillance. That line item number appears to test reactor vessel level 2 and 8.

d. Line items 15 and 63 (reactor vessel level 8) require a semi-daily channel check of those applicable instruments at $\leq$ 53.5". However, the instruments referred to in the procedure for those channel checks are calibrated for elevated temperatures, and in modes 4 and 5, are pegged high such that the required channel check can never be satisfactorily performed. Here again, although operations staff has known of this inadequacy, no procedure change has been implemented to resolve the inadequacy, i.e., refer to an instrument which reads correctly in modes 4 and 5.

In summary, the TS (3/4.3.2) appears to be adequate although there are some inadequacies in the implementation of the requirements. Licensee representatives agreed to review and correct as necessary the semi-daily log sheet and the TS cross reference document (IFI 416/83-06-13).

11. TS 3.6.4 - Containment and Drywell Isolation Valves

The inspector reviewed TS 3.6.4 "Containment and Drywell Isolation valves" and Tables 3.6.4-1 to verify that the licensee has adequately identified all primary containment and drywell penetrations and associated isolation valves and included them in the TS. In addition, the inspector reviewed the limiting condition for operation and surveillance requirement for TS 3.6.4 to ensure that they were appropriate and being properly implemented by the licensee.

The inspector reviewed Final Safety Analysis Report (FSAR) Table 6.2-44 "Containment Isolation Valve Information" and Figures 6.2-76 thru 80 "Containment Leak Rate Test System" and compared them to TS Table 3.6.4-1, plant surveillance procedures and selected "as built" penetrations observed in the plant. No discrepancies were identified. The inspector reviewed the following plant surveillance procedures:

06-OP-1M10-M-0001	"Containment and Drywell Penetration Isolation Monthly Check"
06-OP-1M10-C-1001	"Containment and Drywell Penetration Isolation Cold Shutdown Check"
06-OP-1B21-R-0006	"Containment, Drywell and Auxiliary Building Isolation Valves Functional Test"
06-OP-1B21-C-0003	"Nuclear Boiler Valve Operability"

The procedures were reviewed to determine if they adequately fulfilled the associated surveillance requirements and corresponded to the TS Table 3.6.4-1 of isolation valves. Some minor discrepancies were identified and resolved. No valves were identified that were not covered by procedure or TS.

The inspector toured the Unit 1 Containment and Drywell and randomly selected approximately 45 isolation valves and verified that they were included in TS Table 3.6.4-1 and plant surveillance procedures. No discrepancies were identified. The comments of paragraph 6.a and the IFI identified there also include this large group of valves.

ATTACHMENT 5

29. (GGNS - X43)

SUBJECT: Technical Specification 5.1.3 and Figures 5.1.1-1 and 5.1.3-1, pages 5-1, 5-2 and 5-4.

DISCUSSION: This change involves correction of terminology and a typo. Under Note 3 for Figure 5.1.1-1 the reference to Figure 5.1.4-1 should be changed to 5.1.3-1. The words "Unrestricted Area Boundary" should be changed to "Effluent Release Boundary" in 4 places.

JUSTIFICATION: The terminology "Unrestricted Area Boundary" is not appropriate for gaseous and liquid effluents and is inconsistent with the terminology used in the Emergency Plan. The appropriate terminology is "Effluent Release Boundary". The Effluent Release Boundary is shown in Figure 5.1.3-1 instead of the non-existent Figure 5.1.4-1.

SIGNIFICANT HAZARDS CONSIDERATION:

This change is purely administrative in nature and involves only the correction of terminology and a typo. This change does not involve the reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards consideration.

3/9/84

This change is not acceptable. If the change were made there would still be a number of terms of this type in the Grand Gulf Tech Specs that are not defined.

We have run into these problems a number of times. Late in 1982 and early 1983, after the Grand Gulf T/S had been issued, we changed our model RETS to develop all of our radiological and effluent Tech Specs around 3 definitions — Member of the Public, Site Boundary, Unrestricted Area. Over 40 plants are now using these concepts and definitions in their T/S, and 20 more operating reactors are addressing them as we negotiate their RETS.

We recommend ~~that~~ Grand Gulf commit to this system, used in NUREG-0472, Rev 3 Draft and make the changes indicated on the attached T/S sheets.

W. Meinke RAB

29. (GGNS-X 43)

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

EFFLUENT RELEASE ~~UNRESTRICTED AREA~~ BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The effluent release ~~unrestricted area~~ boundary for gaseous effluents and for liquid effluents shall be as shown in Figure 5.1.3-1. The gaseous effluent release points are shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment is a steel lined, reinforced concrete structure composed of a vertical right cylinder and a hemispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head which contains an approximately eighteen to nineteen foot deep water filled suppression pool connected to the drywell through a series of horizontal vents. The containment has a minimum net free air volume of 1,400,000 cubic feet. The drywell has a minimum net free air volume of 270,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

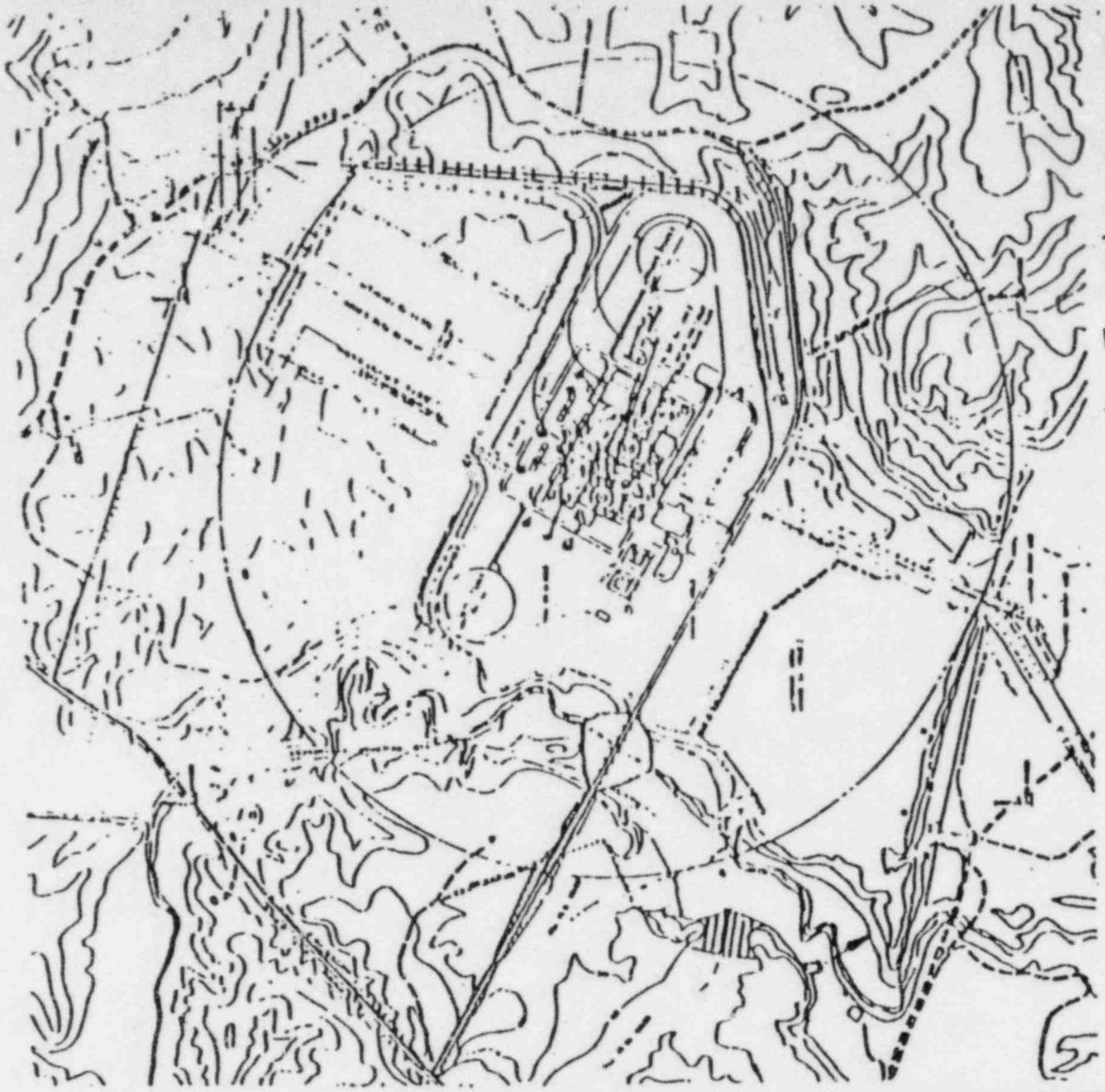
a. Maximum internal pressure:
   1. Drywell 30 psig.
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external-to-internal differential pressure:
   1. Drywell 21 psid.
   2. Containment 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building and the Enclosure Building, and has a minimum free volume of 3,640,000 cubic feet.



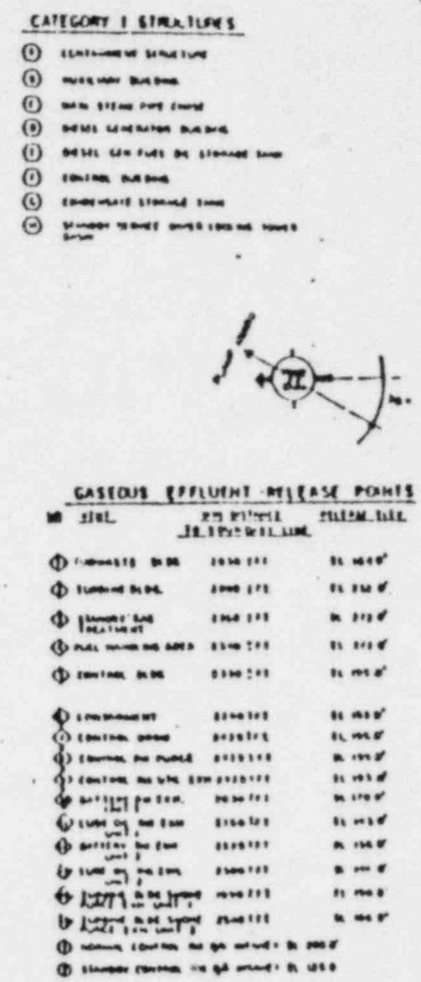
NOTES

1 GRID COORDINATES SHOWN ARE BASED ON MISSISSIPPI COORDINATE SYSTEM, WEST ZONE.

2 DATUM FOR ELEVATIONS SHOWN IS MEAN SEA LEVEL EL. 0.0'.

3 SEE FIGURE ~~5.1.4-1~~ 5.1.3-1 FOR ~~UNRESTRICTED AREA~~ EFFLUENT RELEASE BOUNDARY

4 SEE FIGURE 5.1.2-1 FOR METEOROLOGICAL TOWER LOCATION



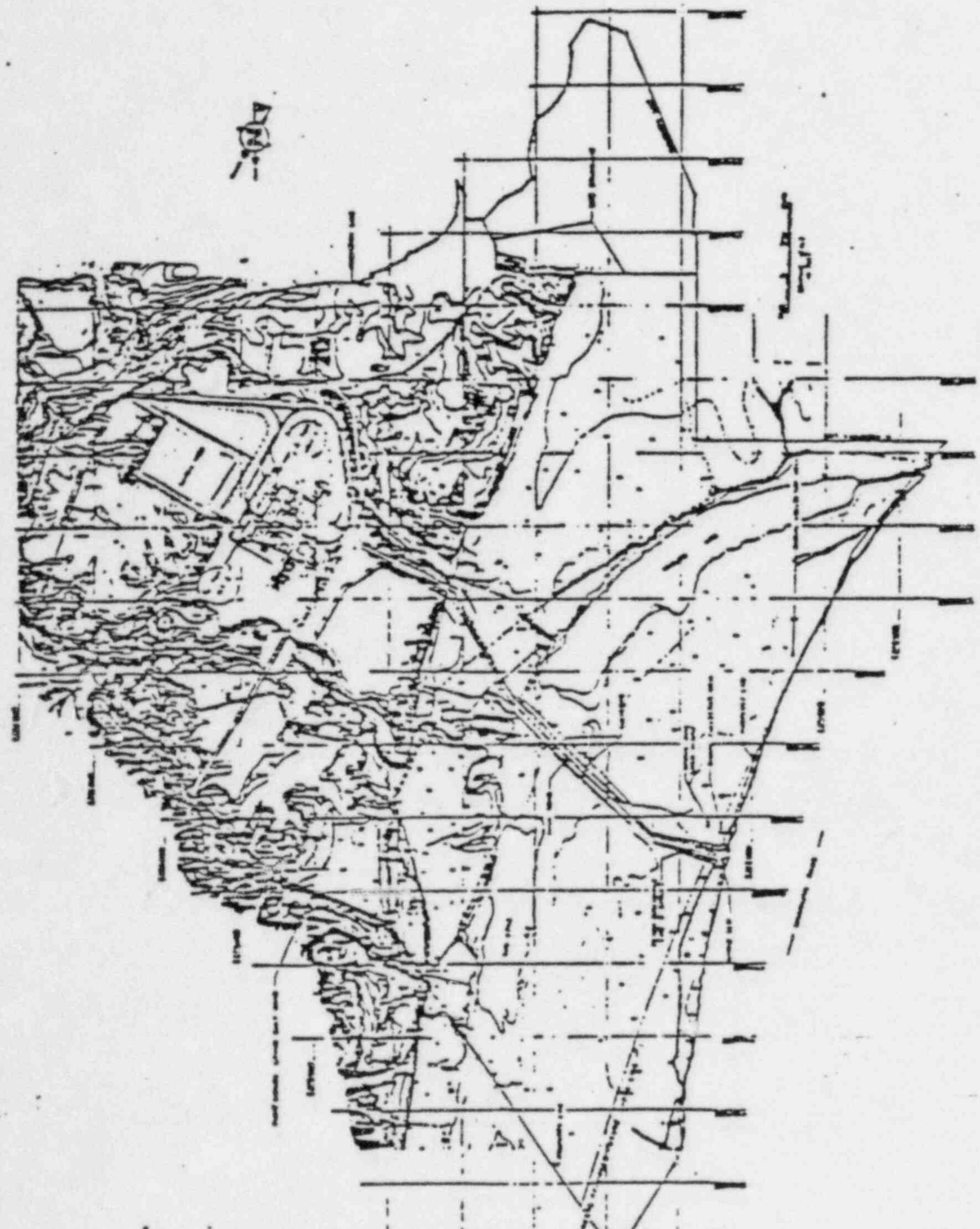
29. (GGNS-X43)

EXCLUSION AREA AND GASEOUS EFFLUENT RELEASE POINTS

FIGURE 5.1.1-1

Exclusion Area radius is 696 meters from the centerline of the Unit 1 Reactor

29. (GGNS-X43)



EFFLUENT RELEASE
~~UNRESTRICTED-AREA~~ BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

FIGURE 5.1.3-1

NUREG-0926

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Technical Specifications

Grand Gulf Nuclear Station, Unit No. 1

Docket No. 50-416

Appendix "A" to
License No. NPF-13

Grand Gulf -1
Tech Specs
as of 3/8/84
Every page checked vs
Master T/S thru date +
attached is complete
thru Amdt #12

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

June 1982

Problems with

Member of the Public
Site Boundary
Unrestricted Area

Recommended changes
are on attached pages.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc., of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

Add
MEMBER OF THE PUBLIC → See pg 1-2 of Model

MINIMUM CRITICAL POWER RATIO

1.24 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

SHUTDOWN MARGIN

1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY ⟶ See page 1-3 of Model

SOLIDIFICATION

1.39 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

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

STAGGERED TEST BASIS

1.41 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

1.42 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.43 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

Add UNRESTRICTED AREA ⟶ See page 1-4 of Model.

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released ~~from the site~~ to unrestricted areas (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

CAPS

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released to UNRESTRICTED AREAS ~~from the site~~ exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11.1.1.1-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11.1.1.1-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC ~~an individual~~ from radioactive materials in liquid effluents released, from each reactor unit, ~~from the site~~ to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.1.2.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

from each reactor unit,

3.11.1.3 The liquid radwaste system components as specified in the ODCM shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the cumulative projected dose due to the liquid effluent ~~from the site~~ (see Figure 5.1.3-1) in a 31 day period would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

to UNRESTRICTED AREAS

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.11 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days, in accordance with the ODCM. CAPS

4.11.1.3.2 The liquid radwaste system components specified in the ODCM shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquids during the previous 92 days.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate (to areas at and beyond the SITE BOUNDARY) due to radioactive materials released in gaseous effluents from the site (see Figure 5.1.3-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines, tritium and all radioactive materials in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SUREVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to radioiodines, tritium and to radioactive materials in particulate form with half lives greater than 8 days, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2.1.2-1.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

to areas at and beyond the SITE BOUNDARY

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, ~~from the site~~ (see Figure 5.1.3-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from the radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions from gaseous effluents for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, ~~from the site~~ (to areas at and beyond the SITE BOUNDARY) (see Figure 5.1.3-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, or radioactive materials in paticulate form, with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.3.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations. Cumulative dose contributions from tritium, radioiodines, and radioactive materials in particulate form wth half-lives greater than 8 days for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

R IOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

from each reactor unit, to areas at and beyond the SITE BOUNDARY

3.11.2.5 The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected cumulative dose due to gaseous effluent releases ~~from the site~~ (see Figure 5.1.3-1) in a 31 day period would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.*

ACTION:

a. With the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days, or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.11 are not applicable.

SURVEILLANCE REQUIREMENTS

from each reactor unit to areas at and beyond the SITE BOUNDARY

4.11.2.5.1 Doses due to gaseous releases ~~from the site~~ shall be projected at least once per 31 days in accordance with the ODCM.

4.11.2.5.2 The VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 30 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*Not applicable to Turbine Building ventilation exhaust unless filtration media is installed

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitments over 12 consecutive months to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to: a. less than or equal to 25 mrem to the total body or any organ (except the thyroid), b. less than or equal to 75 mrem to the thyroid. CAPS

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report reuqired by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence and exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (i.e., dose) to a member of the public from uranium fuel cycle sources including all effluent pathways and direct radiation for a 12 consecutive month period that includes the release(s) coverd by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.00(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

CAPS

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance wth Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

a MEMBER OF THE PUBLIC

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

to UNRESTRICTED AREAS

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents ~~from the site~~ will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to ~~an individual~~, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air, i.e., submersion, was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

to UNRESTRICTED AREAS

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of ~~an individual~~ through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluent from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

a MEMBER OF THE PUBLIC

3/4.11.1.3 LIQUID WASTE TREATMENT

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

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

a MEMBER OF THE PUBLIC

3/4.11.2 GASEOUS EFFLUENTS

either within or

3/4.11.2.1 DOSE RATE

at any time at and beyond

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

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MEMBERS OF THE PUBLIC

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DOSE RATE (Continued)

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE - NOBLE GASES

to UNRESTRICTED AREAS

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

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3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

to UNRESTRICTED AREAS

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A. of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of ~~an individual~~ through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials

a MEMBER OF THE PUBLIC

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM (Continued)

areas at and beyond the SITE BOUNDARY.

are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revison 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive materials in particulate form and tritium are dependent on the existing radionuclide pathway to man in the ~~unrestricted area.~~ The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT AND VENTILATION EXHAUST TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of the system 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 the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the system were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup piping is maintained below the flammability limits of hydrogen. 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.

3/4.11.2.7 MAIN CONDENSER

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.

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitation of 40 CFR 190. The specification requires the preparation and submittal of a special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation which is part of the nuclear fuel cycle.

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

Suggest using words of model page 5-1

UNRESTRICTED AREA BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The unrestricted area boundary for gaseous effluents and for liquid effluents ~~shall be as~~ shown in Figure 5.1.3-1. The gaseous ~~effluent~~ release point~~s are~~ shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment is a steel lined, reinforced concrete structure composed of a vertical right cylinder and a hemispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head which contains an approximately eighteen to nineteen foot deep water filled suppression pool connected to the drywell through a series of horizontal vents. The containment has a minimum net free air volume of 1,400,000 cubic feet. The drywell has a minimum net free air volume of 270,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell 30 psig.
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

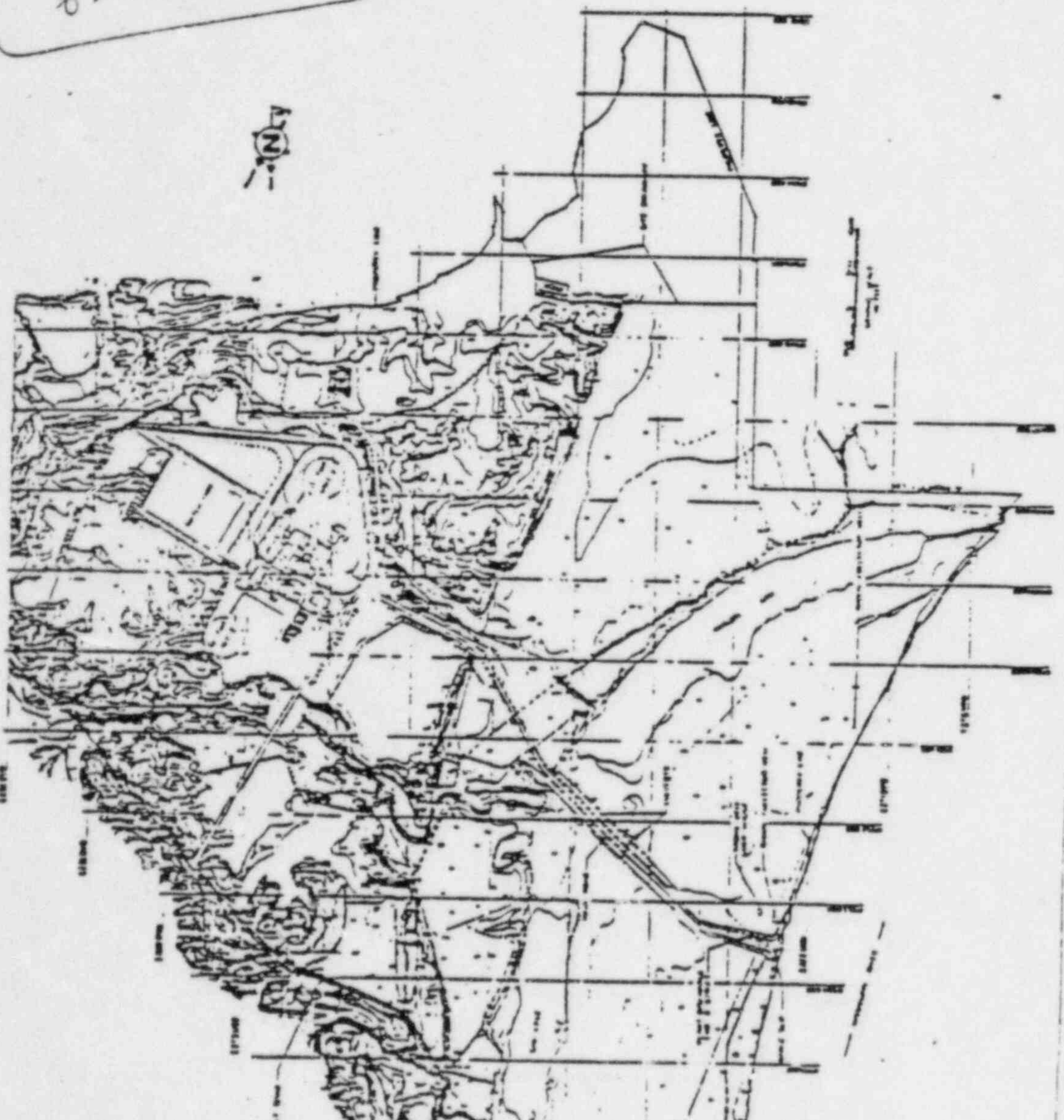
c. Maximum external-to-internal differential pressure:
   1. Drywell 21 psid.
   2. Containment 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building and the Enclosure Building, and has a minimum free volume of 3,640,000 cubic feet.

Note this map must be legible!
See instructions pg 5-2 of Model.

Illegible!



MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE ~~UNRESTRICTED AREA BOUNDARY FOR~~ LIQUID AND GASEOUS EFFLUENTS

FIGURE 5.1.3-1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

(5.1.3-1) ... This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure ~~5.1-3~~) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

CAPS

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

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclide (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),

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

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted area of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) or radioactive waste systems made during the reporting period.

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NUREG-0472
REVISION 3

STANDARD RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS FOR PRESSURIZED WATER REACTORS

JANUARY 1983

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NUREG-0472
REVISION 3

STANDARD RADIOLOGICAL EFFLUENT TECHNICAL

SPECIFICATIONS FOR PRESSURIZED WATER REACTORS

JANUARY 1983

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INDEX

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

NOTE: Add 3/4.3.3.10 and 3/4.3.3.11 with appropriate page numbers to Index section for Monitoring Instrumentation and its Bases; also add 5.1.3 to Section 5.0 Index and make appropriate additions to Section 6.0 Index.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DRAFT

INDEX

BASES

SECTION | PAGE

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

CHANNEL CALIBRATION

1.2 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.3 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.4 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

DOSE EQUIVALENT I-131

1.5 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" [or in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977].

FREQUENCY NOTATION

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

MEMBER(S) OF THE PUBLIC*

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)*

1.8 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.9 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.10 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

*Applies only to RETS, not STS.

1.0 DEFINITIONS (Continued)

PROCESS CONTROL PROGRAM (PCP)*

1.11 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low-level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low-level radioactive waste disposal sites.

PURGE - PURGING*

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

RATED THERMAL POWER

1.13 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ___ MWt.

SITE BOUNDARY*

1.14 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOLIDIFICATION*

1.15 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

SOURCE CHECK*

1.16 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

*Applies only to RETS, not STS.

THERMAL POWER

1.17 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNRESTRICTED AREA*

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

VENTILATION EXHAUST TREATMENT SYSTEM*

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

VENTING*

1.20 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM*

1.21 A WASTE GAS HOLDUP SYSTEM shall be 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.

*Applies only to RETS, not STS.

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TABLE 1.1

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	≥ 0.99	$> 5\%$	$\geq$ (350°F)
2.	STARTUP	≥ 0.99	$\leq 5\%$	$\geq$ (350°F)
3.	HOT STANDBY	< 0.99	0	$\geq$ (350°F)
4.	HOT SHUTDOWN	< 0.99	0	(350°F) $> T_{avg}$ $>$ (200°F)
5.	COLD SHUTDOWN	< 0.99	0	$\leq$ (200°F)
6.	REFUELING**	≤ 0.95	0	$\leq$ (140°F)

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 1.2

FREQUENCY NOTATION

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

*Applies only to RETS, not STS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 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, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, in lieu of a Licensee Event Report, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	28
b. Steam Generator Blowdown Effluent Line	1	29
c. Turbine Building (Floor Drains) Sumps Effluent Line	1	29
2. RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	1	30
b. Component Cooling Water System Effluent Line	1	30
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
a. Steam Generator Blowdown Effluent Line (alternate to item 1.b)	1	29
b. Turbine Building Sumps Effluent Line (alternate to item 1.c)	1	29
4. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown Effluent Line	1	31
c. Discharge Canal	1	31

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TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5. RADIOACTIVITY RECORDERS*		
a. Liquid Radwaste Effluent Line	1	28
b. Steam Generator Blowdown Effluent Line	1	32

*Required only if alarm/trip set point is based on recorder-controller.

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TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - 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 analyzed for radioactivity for up to 30 days at a lower limit of detection of no more than 10^{-7} microcurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram DOSE EQUIVALENT I-131.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of at least 10^{-7} microcurie/ml.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the radioactivity level is determined at least once per 4 hours during actual releases.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL SOURCE CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST
1. RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(3)	Q(1)
c. Turbine Building (Floor Drains) Sumps Effluent Line	D	M	R(3)	Q(1)
2. RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D	M	R(3)	Q(2)
b. Component Cooling Water System Effluent Line	D	M	R(3)	Q(2)
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR				
a. Steam Generator Blowdown Effluent Line (alternate to item 1.b)	D	N.A.	R	Q
b. Turbine Building Sumps Effluent Line (alternate to item 1.c)	D	N.A.	R	Q

TABLE 4.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D(4)	N.A.	R	Q
c. Discharge Canal	D(4)	N.A.	R	Q
5. RADIOACTIVITY RECORDERS*				
a. Liquid Radwaste Effluent Line	D	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D	N.A.	R	Q

*See footnote on page 3/4 3-73.

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TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur 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 setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

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INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, in lieu of a Licensee Event Report, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 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 at the frequencies shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Effluent System Flow Rate Measuring Device	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36
2A. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor (Automatic control)	1	**	39
b. Hydrogen or Oxygen Monitor (Process)	1	**	39
2B. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitors (Automatic control, redundant)	2	**	40, 42
b. Hydrogen or Oxygen Monitors (Process, dual)	2	**	40

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
4. VENT HEADER SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	38
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. CONTAINMENT PURGE SYSTEM (Continued)			
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
6. AUXILIARY BUILDING VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
7. FUEL STORAGE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
8. RADWASTE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
9. OTHER EXHAUST AND VENT SYSTEMS such as: STEAM GENERATOR BLOWDOWN VENT SYSTEM; TURBINE GLAND SEAL CONDENSER EXHAUST			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	41
c. Particulate Sampler	1	*	41
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36

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TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this WASTE GAS HOLDUP SYSTEM may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 40 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

TABLE 3.3-13 (Continued)

TABLE NOTATION

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 42 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Terminaticn of Release	P	P	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	P	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*
2A. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)					
a. Hydrogen Monitor (Automatic control)	D	N.A.	Q(4)	M	**
b. Hydrogen or Oxygen Monitor (Process)	D	N.A.	Q(4) or Q(5)	M	**
2B. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)					
a. Hydrogen Monitors (Automatic control, redundant)	D	N.A.	Q(4)	M	**
b. Hydrogen or Oxygen Monitors (Process, dual)	D	N.A.	Q(4) or Q(5)	M	**

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TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. VENT HEADER SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-13 (Continued)

RADIOACTICE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT PURGE SYSTEM					
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release	D	P	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
6. AUXILIARY BUILDING VENTILATION SYSTEM					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. FUEL STORAGE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
8. RADWASTE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
9. OTHER EXHAUST AND VENT SYSTEMS such as: STEAM GENERATOR BLOWDOWN VENT SYSTEM; TURBINE GLAND SEAL CONDENSER EXHAUST					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-13 (Continued)

TABLE NOTATION

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

(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 setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

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

TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

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

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

E is the counting efficiency, as counts per disintegration,

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

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

Y is the fractional radiochemical yield, when applicable,

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

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

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

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b]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 by a method described in the ODCM to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

[d]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 that is representative of the liquids released.

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

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

DOSE

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. (This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141.)*

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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*This sentence is applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river sited plants this is 3 miles downstream only.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liqud radwaste treatment system shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-3) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each reactor unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be demonstrated OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to ____ curies, excluding tritium and dissolved or entrained noble gases.

a. ________
b. ________
c. ________
d. Outside temporary tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

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*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE or VENT	P Each PURGE[c] Grab Sample	P Each PURGE[c]	Principal Gamma Emitters[b]	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
C.1 Plant Vent	M[c,d] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.2 Fuel Storage Area Ventilation	M[e] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.3 Auxiliary Bldg, Radwaste Area, SGB Vent, Others	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
D. All Release Types as listed in A, B, C above.	Continuous[f]	W[g] Charcoal Sample	I-131	1×10^{-12}
	Continuous[f]	W[g] Particulate Sample	Principal Gamma Emitters[b]	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

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

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

E is the counting efficiency, as counts per disintegration,

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

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

Y is the fractional radiochemical yield, when applicable,

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

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

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

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATION

[b]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

[c]Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within a one hour period.

[d]Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

[e]Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[f]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 3.11.2.1, 3.11.2.2 and 3.11.2.3.

[g]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed either:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each reactor unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.2.4.2 The installed Gaseous Radwaste Treatment System shall be demonstrated OPERABLE by meeting Specification 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume without delay then take the ACTION in a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ___ curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transporation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4.a.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

RADIOLOGICAL ENVIRONMENTAL MONITORING

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations (DR1-DR40) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY (DR1-DR16); an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR17-DR32); the balance of the stations (DR33-DR40) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sampling program. The code letters in parentheses, e.g. DR1, A1, provide one way of defining generic sample locations in this specification that can be used to identify the specific locations in the map(s) and table in the ODCM.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q. 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[c] Gamma isotopic analysis[d] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[e]	1 sample upstream (Wa1) 1 sample downstream (Wa2)	Composite sample over 1-month period[f]	Gamma isotopic analysis[d] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources (Wb1, Wb2), only if likely to be affected[g].	Quarterly	Gamma isotopic[d] and tritium analysis quarterly.
c. Drinking	1 sample of each of 1 to 3 (Wc1 - Wc3) of the nearest water supplies that could be affected by its discharge. 1 sample from a control location (Wc4).	Composite sample over 2-week period[f] when I-131 analysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[h] Composite for gross beta and gamma isotopic analyses[d] monthly. Composite for tritium analysis quarterly.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
d. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wd1).	Semiannually	Gamma isotopic analysis[d] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations (Ia1 - Ia3) within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas (Ia1 - Ia3) between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.[h] 1 sample from milking animals at a control location (Ia4), 15-30 km distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic[d] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
b. Fish and Inverte-brates	1 sample of each commercially and recreationally important species in vicinity of plant discharge area. (Ib1 - Ib__). 1 sample of same species in areas not influenced by plant dis-charge (Ib10 - Ib__).	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[d] on edible portions.
c. Food Products	1 sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged (Ic1 - Ic__).	At time of harvest[i]	Gamma isotopic analyses[d] on edible portion.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products (cont'd)	Samples of 3 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 (Ic10 - Ic13).	Monthly during growing season	Gamma isotopic[d] and I-131 analysis.
	1 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 (Ic20 - Ic23).	Monthly during growing season	Gamma isotopic[d] and I-131 analysis.

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TABLE 3.12-1 (Continued)

TABLE NOTATION

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. (The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The freqency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.)

[c]Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

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TABLE 3.12-1 (Continued)

TABLE NOTATION

[d]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[e]The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

[f]A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

[g]Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[h]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[i]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuborous and root food products.

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TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

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

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TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a,b]

LOWER LIMIT OF DETECTION (LLD)[c]

Analysis	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	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.

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TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 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 picocuries per unit mass or volume,

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

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

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

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[d]LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.)

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.12.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4c shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50. to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of radioactive materials in liquid effluents from all reactor units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in

Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of radioactive materials in liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

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

This specification applies to the release of radioactive materials in liquid effluents from each reactor unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this Specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

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

This specification applies to the release of radioactive materials in gaseous effluents from all reactor units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

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

The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM 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 the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

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

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, *HASL-300* (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," *Anal. Chem. 40*, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report *ARH-SA-215* (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 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 made: 1) 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/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

5.0 DESIGN FEATURES

5.1 SITE

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

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

The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3 (a)(17). The UNRESTRICTED AREA boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

This figure shall consist of a map of the site area showing the SITE BOUNDARY and locating points within the SITE BOUNDARY where radioactive gaseous and liquid effluents are released, as well as where radioactive liquid effluents leave the site. If onsite areas subject to radioactive materials in gaseous or liquid effluents are utilized by the public for recreational or other purposes, these areas shall be outlined on the map and identified by occupancy factors and the licensee's method of occupancy control (if any). The figure shall be sufficiently detailed to allow identification of structures and release point locations and elevations, as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC. The map scale shall be on the order of 2-3"/mile. See NUREG-0133 for additional guidance.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

FIGURE 5.1-3

6.0 ADMINISTRATIVE CONTROLS

6.5.1 UNIT REVIEW GROUP (URG)

RESPONSIBILITIES

6.5.1.6 The URG shall be responsible for:

k. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the (Vice President, Nuclear Operations) and to the (Company Nuclear Review and Audit Group).

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

6.5.2 COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNPAG)

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the (CNRAG). These audits shall encompass:

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

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

m. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

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

6.8 PROCEDURES AND PROGRAMS

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

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

THIRTY DAY WRITTEN REPORTS

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

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

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.11 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual 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.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the site boundary; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.12 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

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

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall also include once a year an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

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

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

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

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

Special 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.

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

6.10 RECORD RETENTION

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

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the (URG).

2. Shall become effective upon review and acceptance by the (URG).

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

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

   a. 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 ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the (URG).

2. Shall become effective upon review and acceptance by the (URG).

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

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

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the (Unit Review Group). The discussion of each change shall contain:

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

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

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

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

    e. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

    f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

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

    h. Documentation of the fact that the change was reviewed and found acceptable by the (URG).

2. Shall become effective upon review and acceptance by the (URG).

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*Licensees may chose to submit the information called for in this Specification as part of the annual FSAR update.