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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

DOCKET NO. 50-206

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. DPR-13

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

   A. The applications for amendment by Southern California Edison Company (SCE or the licensee) and San Diego Gas and Electric Company dated February 26, 1988, and December 12, 1989, as clarified by letters dated July 27, 1990, and September 23, 1991, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Provisional Operating License No. DPR-13 is hereby amended to read as follows:

2. Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 145, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell

for Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 15, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NO. DPR-13

DOCKET NO. 50-206

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
i	i
ii	ii
iii	iii
iv	iv
vi	vi
vii	vii
viii	viii
ix	ix
x	x
xii	xii
xiii	xiii
xiv	xiv
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5
3.5-16	3.5-16
3.5-26 through 3.5-34	3.5-26 through 3.5-30
3.15-1	3.15-1
3.15-2	--
3.15-3	--
3.16-1 through 3.16-7	3.16-1 through 3.16-6
3.17-1	3.17-1
3.17-2	--
3.18-1 through 3.18-9	3.18-1
3.19-1	3.19-1
4.1-13 through 4.1-35	4.1-13 through 4.1-31
4.5-1 through 4.5-6	4.5-1
4.6-1 through 4.6-9	4.6-1 through 4.6-6
4.17-1	4.17-1
4.18-1 through 4.18-6	4.18-1
4.19-1	4.19-1
6.5-4	6.5-4
6.8-3	6.8-3
--	6.8-4
--	6.8-5
6.9-2	6.9-2
6.9-3	6.9-3
6.9-4	--
6.9-5	--
6.10-2	6.10-2
6.13-1	6.13-1
6.14-1	6.14-1
6.15-1	6.15-1

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
i	143, 145
ii	143, 145
iii	143, 145
iv	139, 145
v	90, 130, 131
vi	90, 130, 131, 145
vii	90, 102, 130, 131 145
viii	90, 130, 131, 145
ix	90, 91, 102, 130 131, 145
x	90, 91, 130, 131 145
xi	55, 92, 102, 110, 111, 130, 131, 143
xii	56, 58, 71, 79, 83, 104, 117, 130, 131, 143, 145
xiii	31, 56, 58, 79, 83, 84, 91, 117, 130, 131, 143, 145
xiv	131, 143, 145
xv	143
1.0-1	31, 56, 59, 83, 117, 130
1.0-2	31, 56, 59, 83, 104, 117, 130
1.0-3	31, 56, 59, 79, 83, 104, 117, 130, 145
1.0-4	31, 56, 59, 79, 83, 117, 130, 145
1.0-5	77, 79, 83, 117, 130, 145
1.0-6	79, 83, 96, 117, 130
1.0-7	58, 83, 117, 130
1.0-8	56, 83, 117, 130
2.1-1	43, 55, 97, 117, 130
2.1-2	43, 97, 117, 121, 130
2.1-3	43, 117, 121, 122, 130
2.1-4	43, 117, 121, 122, 130
2.1-5	43, 97, 117, 121, 122, 130
2.1-6	55, 130
3.0-1	43, 56, 64, 83 130
3.0-2	56, 64, 83, 130, 135
3.0-3	135
3.0-4	135
3.1-1	29, 38, 70, 83, 91, 96, 130
3.1-2	29, 83, 96, 130
3.1-3	43, 77, 103, 130
3.1-4	77, 130
3.1-5	77, 125, 130
3.1-6	77, 102, 130
3.1-7	77, 102, 103, 130
3.1-8	43, 102, 103, 130
3.1-9	77, 102, 130
3.1-10	Change No. 14, 38, 102, 130
3.1-11	Change No. 14, 38, 102, 130
3.1-12	Change No. 14, 92, 130
3.1-13	Change No. 14, 92, 130
3.1-14	Change No. 14, 102, 130
3.1-15	Change No. 14, 102, 130
3.1-16	Change No. 14, 102, 130
3.1-17	Change No. 14, 102, 130
3.1-18	Change No. 7, 37, 55, 91, 119, 130
3.1-19	Change No. 7, 55, 119, 130
3.1-20	37, 55, 119, 130
3.1-21	58, 59, 83, 130
3.1-22	58, 130
3.1-23	83, 130
3.2-1	102, 130
3.2-2	25, 102, 130
3.3-1	25, 37, 86, 124 130
3.3-2	25, 130
3.3-3	25, 38, 86, 124, 130, 142
3.3-4	25, 37, 124, 130
3.3-5	25, 130
3.3-6	25, 102, 120, 130
3.3-7	25, 102, 130
3.3-8	25, 38, 122, 130
3.3-9	25, 38, 122, 130
3.3-10	25, 130
3.3-11	NRC Order 4/20/81, 130
3.3-12	NRC Order 4/20/81, 130
3.4-1	29, 82, 125, 130
3.4-2	29, 130
3.4-3	82, 125, 130, 138
3.4-4	82, 125, 130, 138
3.5-1	83, 117, 130, 143
3.5-2	43, 56, 58, 83, 117, 128, 130
3.5-3	43, 56, 58, 83, 117, 121, 122, 130, 143
3.5-4	55, 58, 83, 117, 118, 121, 128, 130
3.5-5	83, 117, 130
3.5-5a	143
3.5-5b	143
3.5-5c	143
3.5-6	7, 11, 25, 35, 55, 56, 111, 130
3.5-7	7, 11, 25, 35, 49, 55, 56, 111, 122, 130
3.5-8	11, 49, 111, 122, 130
3.5-9	11, 25, 56, 111, 130

LIST OF EFFECTIVE PAGES
(Cont.)

Page	Amendment No.
3.5-10	56, 130
3.5-11	56, 130
3.5-12	56, 130
3.5-13	56, 130
3.5-14	56, 130
3.5-15	58, 83, 130
3.5-16	58, 83, 130, 145
3.5-17	58, 83, 130
3.5-18	58, 72, 130
3.5-19	58, 83, 125, 130
3.5-20	125, 130
3.5-21	58, 83, 117, 124, 125, 130
3.5-22	64, 82, 125, 130
3.5-23	58, 82, 125, 130
3.5-24	58, 82, 125, 130
3.5-25	58, 82, 125, 129, 130
3.5-26	79, 90, 105, 130, 145
3.5-27	79, 90, 105, 130, 145
3.5-28	83, 130, 145
3.5-29	83, 130, 145
3.5-30	83, 91, 130, 145
3.6-1	25, 56, 58, 73, 130
3.6-2	Change No. 7, 56, 130
3.6-3	58, 99, 130
3.6-4	58, 71, 99, 130
3.6-5	58, 59, 71, 99, 130
3.6-6	58, 59, 83, 130
3.7-1	25, 52, 68, 84, 130, 134, 136
3.7-2	25, 84, 106, 130, 134, 136
3.7-3	25, 84, 130, 134, 136
3.7-4	25, 52, 68, 84, 106, 130, 134, 136
3.7-5	25, 52, 68, 84, 106, 120, 130, 134, 136, 143
3.7-6	134, 136, 143
3.7-7	134, 136
3.7-8	136
3.8-1	25, 36, 37, 73, 77, 130
3.8-2	36, 77, 130
3.8-3	25, 36, 73, 77, 130
3.8-4	36, 73, 77, 130
3.8-5	36, 73, 77, 130
3.9-1	3, 10, 112, 130
3.9-2	3, 10, 112, 130
3.10-1	7, 8, 112, 122, 130
3.10-2	7, 112, 122, 130
3.11-1	7, 8, 11, 35, 55, 112, 117, 122, 130
3.11-2	7, 11, 112, 117, 122, 130
3.12-1	14, 130
3.13-1	21, 63, 81, 130
3.13-2	21, 63, 81, 130
3.14-1	31, 44, 93, 130, 131
3.14-2	31, 130, 131
3.14-3	31, 130, 131
3.14-4	31, 93, 130, 131
3.14-5	31, 130, 131
3.14-6	130, 131, 134
3.14-7	130, 131
3.14-8	130, 131
3.14-9	130, 131
3.14-10	130, 131
3.14-11	31, 93, 130, 131
3.14-12	31, 130, 131
3.14-13	31, 130, 131
3.14-14	31, 130, 131
3.14-15	31, 93, 130, 131
3.14-16	31, 130, 131
3.14-17	31, 130, 131, 134
3.14-18	31, 130, 131
3.14-19	130, 131
3.14-20	130, 131
3.14-21	130, 131
3.14-22	130, 131
3.14-23	130, 131
3.14-24	130, 131
3.14-25	130, 131
3.14-26	130, 131
3.15-1	79, 130, 145
3.16-1	79, 130, 145
3.16-2	79, 90, 91, 105, 130, 145
3.16-3	79, 90, 91, 105, 130, 145
3.16-4	79, 90, 91, 105, 130, 145
3.16-5	79, 90, 130, 145
3.16-6	79, 90, 130, 145
3.17-1	79, 90, 91, 105, 130, 145
3.18-1	79, 91, 105, 130, 145
3.19-1	79, 90, 105, 130, 145
3.20-1	102, 130
3.20-2	102, 130
4.0-1	83, 130, 137
4.0-2	83, 130, 137
4.0-3	135
4.0-4	135
4.1-1	29, 56, 83, 117, 130
4.1-1a	143
4.1-2	7, 22, 83, 117, 122, 130, 143
4.1-3	117, 130
4.1-4	25, 29, 70, 96, 117, 130
4.1-5	25, 29, 70, 96, 130
4.1-6	25, 29, 56, 70, 100, 122, 130, 134
4.1-7	25, 29, 36 77, 103, 130, 142
4.1-8	77, 125, 130
4.1-9	77, 122, 125, 130

LIST OF EFFECTIVE PAGES
(Cont.)

Page	Amendment No.
4.1-10	117, 130
4.1-11	25, 130
4.1-12	Change No. 5, 25, 130
4.1-13	79, 130, 145
4.1-14	79, 130, 145
4.1-15	58, 130, 145
4.1-16	58, 130, 145
4.1-17	58, 130, 145
4.1-18	58, 83, 130, 145
4.1-19	58, 83, 117, 125, 130, 145
4.1-20	58, 109, 130, 145
4.1-21	58, 130, 145
4.1-22	58, 82, 125, 130, 145
4.1-23	58, 82, 125, 130, 145
4.1-24	65, 82, 125, 130, 145
4.1-25	65, 125, 130, Revised by NRC letter 2/12/92, 145
4.1-26	82, 130, 145
4.1-27	83, 130, 145
4.1-28	83, 130, 145
4.1-29	83, 130, 145
4.1-30	119, 130, 145
4.1-31	119, 130, 145
4.2-1	25, 37, 54, 114, 130
4.2-2	25, 37, 114, 130
4.2-3	25, 37, 114, 130
4.2-4	25, 37, 114, 124, 130
4.2-5	25, 37, 114, 124, 130
4.2-6	37, 114, 130
4.2-7	NRC Order 4/20/81, 54, 130
4.3-1	24, 58, 75, 130
4.3-2	5, 24, 58, 75, 87, 118, 130
4.3-3	24, 58, 87, 118, 130
4.3-4	24, 87, 130
4.3-5	24, 75, 87, 130

Page	Amendment No.
4.3-6	75, 87, 130
4.3-7	58, 130
4.3-8	58, 59, 83, 130
4.3-9	58, 83, 130
4.4-1	Change No. 12, 25, 56, 82, 84, 95, 104, 123, 130, 136
4.4-2	25, 84, 130, 136
4.4-3	25, 56, 82, 84, 117, 130, 134, 136
4.4-4	25, 82, 84, 95, 104, 123, 124, 130, 134, 136, 143
4.4-5	84, 130, 136
4.4-6	25, 56, 84, 104, 123, 130, 136
4.4-7	25, 56, 84, 130, 134, 136
4.4-8	136
4.4-9	136
4.4-10	136
4.5-1	Change No. 10, 38, 79, 130, 145
4.6-1	38, 79, 130, 145
4.6-2	79, 130, 145
4.6-3	79, 130, 145
4.6-4	79, 130, 145
4.6-5	79, 130, 145
4.6-6	79, 130, 145
4.7-1	Change No. 14, 46, 130
4.8-1	91, 130
4.9-1	Change No. 14, 130
4.9-2	130
4.10-1	13, 94, 130
4.10-2	13, 94, 130
4.11-1	14, 109, 130
4.11-2	14, 109, 130
4.12-1	91, 130, Revised by NRC letter 2/12/92
4.13-1	18, 113, 130
4.14-1	21, 33, 63, 81, 130
4.14-2	21, 63, 81, 130
4.14-3	21, 63, 81, 130

Page	Amendment No.
4.14-4	21, 63, 81, 105, 130
4.15-1	31, 37, 38, 44, 130, 131
4.15-2	31, 44, 130, 131
4.15-3	31, 130, 131
4.15-4	130, 131
4.15-5	130, 131
4.15-6	31, 130, 131
4.15-7	31, 130, 131
4.15-8	130, 131
4.15-9	130, 131
4.15-10	130, 131
4.15-11	130, 131
4.15-12	130, 131
4.16-1	31, 101, 115, 130, 133
4.16-2	37, 101, 115, 130, 133
4.16-3	37, 101, 115, 130, 133
4.16-4	37, 55, 101, 115, 130, 133
4.16-5	37, 91, 101, 105, 115, 130, 133
4.17-1	79, 130, 145
4.18-1	79, 130, 145
4.19-1	79, 105, 130, 145
4.20-1	102, 130
5.1-1	25, 72, 130
5.1.2	72, 79, 130
5.2-1	72, 130
5.2-2	72, 130
5.2-3	72, 130
5.3-1	3, 72, 130
5.3.2	3, 37, 72, 130
5.4-1	130
6.1-1	12, 39, 66, 91, 130
6.2-1	44, 66, 91 110, 130
6.2-2	88, 91, 130
6.2-3	66, 88, 91, 105, 110, 126, 130
6.2-4	12, 58, 91, 130

LIST OF EFFECTIVE PAGES
(Cont.)

Page	Amendment No.
6.3-1	12, 27, 39 54, 58, 66, 91, 130
6.4-1	31, 39, 54, 91, 130, 139
6.5-1	Change No. 69, 12, 39, 42, 54, 66, 91, 130
6.5-2	39, 54, 66, 91, 130
6.5-3	12, 16, 39, 50, 54, 66, 91, 130
6.5-4	39, 66, 69, 79, 91, 130, 145
6.5-5	12, 50, 90, 91, 130
6.5-6	12, 16, 69, 91, 105, 130
6.5-7	12, 31, 91, 110, 130
6.6-1	12, 91, 130
6.7-1	39, 54, 66, 91, 130
6.8-1	39, 54, 66, 79, 91, 105, 130
6.8-2	39, 91, 130
6.8-3	91, 130, 145
6.8-4	145
6.8-5	145
6.9-1	12, 16, 30, 91, 130
6.9-2	12, 16, 30, 79, 91, 96, 130 145
6.9-3	12, 15, 16, 30, 79, 91, 96, 105 130, 145
6.10-1	12, 15, 16, 91, 130
6.10-2	12, 63, 81, 91, 130, 145
6.11-1	58, 61, 91, 130
6.12-1	23, 38, 58, 91, 130
6.12-2	23, 38, 91, 130
6.13-1	58, 79, 91, 130, 145
6.14-1	58, 79, 91, 130, 145
6.15-1	58, 79, 91, 130, 145
6.16-1	90, 91, 130
6.16-2	90, 91, 130
6.16-3	90, 91, 105, 130
6.16-4	90, 91, 130

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

TABLES

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLES

TABLES

E - AVERAGE DISINTEGRATION ENERGY

E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPPRESSION WATER SYSTEM shall consist of a water source(s), pump(s), and distribution piping with associated isolation valves (i.e., system header, hose standpipe and spray header isolation valves).

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from 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 Effluent Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

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

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 1.2.

PROCESS CONTROL PROGRAM

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE-PURGING

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

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 1347 Mwt.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESIDUAL HEAT REMOVAL (RHR) TRAIN

An RHR TRAIN shall be a train of components that includes: one RHR pump aligned with one RHR heat exchanger; one component cooling water pump aligned with the same RHR heat exchanger and with one component cooling water heat exchanger; and one salt water pump aligned with the same component cooling water heat exchanger.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

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

SOLIDIFICATION

[Deleted]

SOURCE CHECK

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

STAGGERED TEST BASIS

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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

TABLE 3.5.5-1

CONTAINMENT ISOLATION INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Containment Isolation (Valves listed In Table 3.6.2-1)					
a) Manual	2	1	2	1, 2, 3, 4	11
b) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	9
c) Sequencer Subchannels	2/sequencer	2/sequencer	2/sequencer	1, 2, 3, 4	8
d) Safety Injection					
1) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	9*
2) Pressurizer Pressure-Low	3/train	2/train	2/train	1, 2, 3	9*
Purge and Exhaust Isolation (POV-9, POV-10, CV-10, CV-40, CV-116)					
a) Manual	1/valve	1/valve	1/valve	1, 2, 3, 4	10
b) Containment Radioactivity-High	1	1	1	1, 2, 3, 4	10*

3.5.8 [Deleted]

3.5.9 [Deleted]

3.5.10 RADIATION MONITORING INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.10-1.

OBJECTIVE: To ensure reliability of the radiation monitoring instrumentation.

SPECIFICATION: The radiation monitoring instrumentation shown in Table 3.5.10-1 shall be OPERABLE with their alarm setpoints within the specified limits.

ACTION:

A. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.5.10-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

B. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.5.10-1.

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

BASIS: The OPERABILITY of the radiation monitoring channels ensures that (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm is initiated when the radiation level trip setpoint is exceeded.

REFERENCES: (1) NRC letter dated November 1, 1983, from D. G. Eisenhut to to all Pressurized Water Reactors Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 3.5.10-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Control Room Area (R-1231)	1	All	1 mR/hr	10^{-2} - 10^{2} mR/hr	25
b. Spent Fuel Pool Area (R-1236)	1	*	25 mR/hr	10^{-2} - 10^{2} mR/hr	26
c. Containment Radiation Monitor-High Range (R-1255, R-1257)	2	1, 2, 3 & 4	10 R/hr	1 - 10^{8} R/hr	27
2. PROCESS MONITORS					
a. Wide Range Gas Monitor (R-1254)	1	1, 2, 3 & 4	per ODCM	10^{-7} - 10^{5} mCi/cc	27
b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B)	1/steamline	1, 2, 3 & 4	1 mR/hr (low) 1 R/hr (high)	10^{-1} - 10^{4} mR/hr 10^{-1} - 10^{4} R/hr	27

* With fuel in the spent fuel pool or building

TABLE 3.5.10-1 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement; within 1 hour: (1) either initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation, or (2) initiate the preplanned alternate method of monitoring and alarming the area radiation.

ACTION 26 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 27 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement of Table 3.5.10-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

(1) either restore the inoperable channel(s) to OPERABLE status within 7 days of initiating the preplanned alternate method, or

(2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, within 14 days following initiating the preplanned alternate method, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3.15 [Deleted]

3.16 RADIOACTIVE GASEOUS EFFLUENTS

3.16.1 [Deleted]

3.16.2 [Deleted]

3.16.3 [Deleted]

3.16.4 [Deleted]

3.16.5 GAS STORAGE TANK

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of radioactivity contained in gas storage tanks.

SPECIFICATION:

A. The quantity of radioactivity contained in each gas storage tank shall be limited to $\leq$ 56,000 curies noble gases (considered as Xe-133).

B. ACTION:

1. 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 within 48 hours reduce the tank contents to within the limit.

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

BASIS: The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

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 an individual at the nearest exclusion area boundary will not exceed 0.5 mrem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG 0800, July 1982.

3.16.6 EXPLOSIVE GAS MIXTURE

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of explosive gases contained in the gas storage tanks.

SPECIFICATION:

A. 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.

B. ACTION:

1. 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, restore the concentration of oxygen to within the limit within 48 hours.

2. With the concentration of oxygen in the waste gas holdup system 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 2% by volume without delay.

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

BASIS: 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. 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.

3.17 [Deleted]

3.18 [Deleted]

3.19 [Deleted]

4.1.2 [Deleted]

4.1.3 [Deleted]

4.1.4 CONTAINMENT ISOLATION INSTRUMENTATION

APPLICABILITY: Applies to instrumentation which actuates the containment sphere isolation valves, containment sphere purge and exhaust valves, and containment sphere instrumentation vent header valves.

OBJECTIVE: To ensure reliability of the containment sphere isolation provisions.

SPECIFICATION:

A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.4-1.

B. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

BASIS: The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standard. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.4-1

CONTAINMENT ISOLATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
<u>Containment Isolation</u> (Valves listed in Table 3.6.2-1)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
c) Sequencer Subchannels	N.A.	N.A.	M	1, 2, 3, 4
d) Safety Injection				
1) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
2) Pressurizer Pressure-Low	N.A.	R	M	1, 2, 3, 4
<u>Purge and Exhaust Isolation</u> (POV-9, POV-10, CV-10, CV-40, CV-116)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Radioactivity-High	S	R	M	1, 2, 3, 4

TABLE 4.1.4-1 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL TEST at least once per 31 days.

(2) The CHANNEL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

4.1.5 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3.

OBJECTIVES: To ensure the reliability of the accident monitoring instrumentation shown in Table 4.1.5-1.

SPECIFICATION: Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.1.5-1.

BASIS: The surveillance requirements specified for these systems ensure that the overall functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
Pressurizer Water Level	M	R
Auxiliary Feedwater Flow Indication*	M	R
° Auxiliary Feedwater Flow Rate	M	R
° Steam Generator Water Level (Wide Range)	M	R
° Reactor Coolant System Loop Delta-T Indication	M	R
Reactor Coolant System Subcooling Margin Monitor	M	R
PORV Position Indicator	M	R
PORV Block Valve Position Indicator	M	R
Safety Valve Position Indicator	M	R
Containment Pressure (Wide Range)	M	R
Refueling Water Storage Tank Water Level	M	R
Containment Sump Water Level (Narrow Range)	M	R
Containment Water Level (Wide Range)	M	R
Neutron Flux (Wide Range)	M	R**

* See footnote of Table 3.5.6-1.
** Neutron detectors may be excluded from CHANNEL CALIBRATION.

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) and their associated block valves for MODES 1, 2, and 3.

OBJECTIVE To ensure the reliability of the PORVs and block valves.

SPECIFICATION:

A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and

2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is being maintained closed in order to meet the requirements of Specification 3.1.5.A.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the getting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.7 PRESSURIZER

APPLICABILITY: Applies to pressurizer heaters and pressurizer water level for MODES 1, 2, and 3.

OBJECTIVE: To ensure proper pressurizer water volume and to ensure the capability to energize the pressurizer heaters from the emergency diesel generator.

SPECIFICATION:

A. The pressurizer water level shall be determined to be between 5% and 70% at least once per 12 hours.

B. The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal supply to the emergency diesel generator and energizing the heaters.

BASIS: The requirement that the pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency diesel generator provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

REFERENCES:

(1) NRC Letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.8 AUXILIARY FEEDWATER INSTRUMENTATION

APPLICABILITY: Applies to the auxiliary feedwater instrumentation and interlocks in MODES 1, 2, and 3.

OBJECTIVE: To ensure reliability of automatic initiation of the auxiliary feedwater system.

SPECIFICATION: A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the surveillance requirements specified in Table 4.1.8-1.

BASIS: The surveillance requirements specified for this instrumentation ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.8-1

AUXILIARY FEEDWATER INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
a.	Manual	N/A	N/A	N/A	R	1, 2, 3
b.	Automatic Actuation Logic	N/A	N/A	M	N/A	1, 2, 3
c.	Steam Generator Water Level-Low	S	R	M	N/A	1, 2, 3
d.	AFW Train Interlocks	N/A	R	M	N/A	1, 2, 3

4.1.9 AUXILIARY FEEDWATER SYSTEM SURVEILLANCE

APPLICABILITY: Applies to the auxiliary feedwater pumps and valves for MODES 1, 2, and 3.

OBJECTIVE: To ensure the reliability of the auxiliary feedwater system.

SPECIFICATION:

A. Each auxiliary feedwater pump shall be demonstrated OPERABLE by testing each pump in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

B. At least once per 31 days an inspection shall be made to verify that each non-automatic valve in the emergency flow path is not locked, sealed, or otherwise secured in position is in its correct position.

C. Each auxiliary feedwater Train shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying that the AFW Train B pump starts as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that AFW Train A motor driven pump starts as designed automatically upon receipt of auxiliary feedwater actuation AND Train B low flow test signals. Subsequently, verify the pump stops upon receipt of a Train B positive flow test signal.

3. Within 72 hours after entering MODE 3, verifying that the AFW Train A steam driven pump enters warm-up mode upon receipt of an auxiliary feedwater actuation test signal. Subsequently, verify pump starts upon receipt of a Train B low flow test signal, and returns to warm-up mode upon receipt of Train B positive flow test signal.

4. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of actuation test signals.

D. When the reactor coolant system pressure remains less than 500 psig for a period longer than thirty (30) days, flow tests shall be performed to verify the emergency flow paths from the auxiliary feedwater storage tank to each steam generator, using each motor driven auxiliary feedwater pump prior to increasing reactor coolant system pressure above

500 psig. The flow tests shall be conducted with the auxiliary feedwater system valves in their emergency alignment. Within 72 hours after entering MODE 3, the steam driven auxiliary feedwater pump shall be similarly tested.

E. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the steam driven auxiliary feedwater pump. However, the steam driven AFW pump must be OPERABLE in all other respects.

BASIS:

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

The design of the auxiliary feedwater system further ensures sufficient AFW flow into the intact feedwater lines without exceeding pump run-out or water hammer limits for any applicable design basis event with or without concurrent loss of offsite power and a single active failure.(2,3)

Specification 4.1.9.A demonstrates the operability of the AFW pumps by testing requirements of the Section XI IST program.(4) During normal power operation, pump G-10W is substantial flow tested and pump G-10 and G-10S are tested on minimum flow. During this test differential pressure, pump speeds, and bearing vibration are measured for all 3 pumps. Pump discharge flow is measured for AFW pump G-10W. At each return from a MODE 3, 4, or 5 shutdown following power operations, pumps G-10 and G-10S are full flow tested unless tested in the previous 92 days. G-10S is tested prior to MODE 3. G-10 is tested in MODE 1 at about 20% power.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

(2) SCE letter dated November 6, 1987, from M. O. Medford to NRC Document Control Desk.

(3) SCE letter dated November 20, 1987, from M. O. Medford to NRC Document Control Desk.

(4) SCE letter dated January 3, 1991, from F. R. to NRC Document Control Desk.

4.1.10 AUXILIARY FEEDWATER STORAGE TANK SURVEILLANCE

APPLICABILITY: Applies to the auxiliary feedwater storage tank for MODES 1, 2, and 3.

OBJECTIVE: To ensure the availability of an adequate auxiliary feedwater supply.

SPECIFICATION:

A. The auxiliary feedwater storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

BASIS: See Basis for 3.4.4.

4.1.11 RADIATION MONITORING INSTRUMENTATION

APPLICABILITY: As shown in Table 4.1.11-1

OBJECTIVE: To ensure the reliability of the radiation monitoring instrumentation.

SPECIFICATION: A. Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.11-1.

BASIS: The surveillance requirements specified for these systems ensure that the overall functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES: (1) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 4.1.11-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	APPLICABLE MODES
1. AREA MONITORS				
a. Spent Fuel Pool Area (R-1236)	S	R	M	*
b. Control Room Area (R-1231)	S	R	M	All
c. Containment Radiation Monitor-High Range (R-1255, R-1257)	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Wide Range Gas Monitor (R-1254)	**	**	**	1, 2, 3 & 4
b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B)	S	R	M	1, 2, 3 & 4

* See footnote of Table 3.5.10-1

**In accordance with Table 4.1.3.1 surveillance requirements for this instrument channel.

4.1.12 REACTOR COOLANT SYSTEM VENTS

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure the reliability of the reactor coolant vent system.

SPECIFICATION: Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.
3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN.

BASIS: See basis for 3.1.7, Reactor Coolant System Vents.

REFERENCES: NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specifications (Generic Letter No. 83-37).

4.1.13 LEAKAGE AND LEAKAGE DETECTION SYSTEMS

APPLICABILITY: Applies to the reactor coolant leakage and detection systems delineated in Specification 3.1.4.

OBJECTIVE: To ensure the reactor coolant system leakage limits are maintained and to ensure the OPERABILITY of those systems that are used to detect leakage from the reactor coolant system.

SPECIFICATION:

A. Reactor Coolant System leakage shall be demonstrated to be within limits by:

1. Monitoring the containment atmosphere radioactivity at least once per 12 hours.

2. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

3. Monitoring the steam generator blowdown effluent radioactivity at least once per 12 hours.

4. Monitoring the containment sump level indicator (LIS 2001 or 3001) at least once per 12 hours.

B. The leakage detection systems shall be demonstrated OPERABLE by the performance of CHANNEL CHECK, SOURCE CHECK, CHANNEL TEST, AND CHANNEL CALIBRATION at the frequencies specified in Table 4.1.13-1;

BASIS: The monitoring of reactor coolant system leakage and maintenance of OPERABILITY of the reactor coolant leakage detection systems will assure that the sources of leakage are monitored and/or identified. The methods described above provide an acceptable means of verifying the OPERABILITY required by Specification 3.1.4.

REFERENCES:

1. SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage, NUREG-0829, December 1986

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973

3. Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 4, NUREG-0452

TABLE 4.1.13-1

LEAKAGE DETECTION SYSTEMS

	INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL TEST	CHANNEL CALIBRATION
1.	Containment Atmosphere Particulate Monitor (R1211)	D	M	N/A	R
2.	Containment Atmosphere Gaseous Monitor (R1212)	*	*	*	*
3.	Sphere Sump Level Control System (LS80 and 82)	N/A	N/A	N/A	R
4.	Containment Sphere Sump Level Monitor (LIS 2001 and 3001)	**	N/A	N/A	**
5.	Steam Generator Blowdown Effluent Monitor (R1216)	***	***	***	***

*In accordance with Table 4.1.3.1, surveillance requirements for this instrument channel.
**In accordance with Table 4.1.5-1, surveillance requirements for these instrument channels.
***In accordance with Table 4.1.2.1, surveillance requirements for this instrument channel.

4.5 [Deleted]

4.6 RADIOACTIVE GASEOUS EFFLUENTS

4.6.1 [Deleted]

4.6.2 [Deleted]

4.6.3 [Deleted]

4.6.4 [Deleted]

4.6.5 GAS STORAGE TANK

APPLICABILITY: At all times.

OBJECTIVE: To verify the quantity of radioactive material contained within the gas storage tanks.

SPECIFICATION: The quantity of radioactivity material contained in each gas storage tank shall be determined to be within the limit specified in Specification 3.16.5 at least once per 24 hours when radioactive materials are being added to the tank.

BASIS: Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

4.6.6 EXPLOSIVE GAS MIXTURE

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of explosive gases contained in the gas storage tanks.

SPECIFICATION: The concentrations of hydrogen and/or oxygen in the waste gas holdup system shall be determined to be within the limits specified in Specification 3.16.6 by analyzing grab samples of the waste gas holdup system contents at the waste gas decay tank in service daily and every 4 hours during degassing. Degassing is defined as the process to reduce reactor coolant system (RCS) dissolved H_2 gas concentration in preparation for refueling or for opening the reactor coolant system.

BASIS: 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. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the release of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

4.17 [Deleted]

4.18 [Deleted]

4.19 [Deleted]

6.5.2.5 Proposes tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager, or members of the site/station management staff previously designated by the Vice President and Site Manager, Nuclear Generation Site. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and the NSG.

6.5.2.6 Recommended changes to the station security plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.

6.5.2.7 Recommended changes to the station emergency plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.

6.5.2.8 The Station Manager shall assure the performance of a review by a qualified individual/organization of every uncontrolled or unplanned release of radioactivity to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President and Site Manager, Nuclear Generation Site, to the NSG, and to the Station Manager.

6.5.2.9 The Station Manager shall assure the performance of a review by a qualified individual/organization and may designate the approval of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, to the NSG, and to the Station Manager.

6.5.2.10 Documentation of each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained.

(ii) Identification of the procedures used to measure the values of the critical parameters,

(iii) Identification of process sampling points,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off control point chemistry conditions, and

(iv) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiated corrective action.

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include:

(i) Training of personnel,

(ii) Procedures for sampling and analysis, and

(iii) Provisions for maintenance of sampling and analysis equipment.

f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

(1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

(2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2;

(3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

(4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50;

(5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year, respectively, in accordance with the methodology and parameters in the ODCM at least every 31 days;

(6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50;

(7) Limitations of the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1;

(8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50;

(9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radioactive nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50;

(10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

g. <u>Radiological Environmental Monitoring Program</u>

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

g. <u>Radiological Environmental Monitorint Program</u> (Cont.)

(1) Monitoring, sampling, analysis, and reporting of radiation and radionuclices in the environment in accordance with the methodology and parameters in the ODCM.

(2) A Land Uses Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

(3) Participation in an Interlaboratory Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.1.1. The following information shall be included in these reports: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis whle limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretation, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.7 [Deleted]

*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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted withn 60 days after January 1 and July 1 of each year.

6.9.1.9 [Deleted]

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to pressurizer safety and relief valves, shall be submitted to the Nuclear Regulatory Commission, on a monthly basis, no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission within the time period specified for each report.

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* A single submittal may be made for 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 radioactive material from each unit.

** [Deleted]

f. Records of reactor tests and experiments.

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

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities not included in 6.10.1 that are required by the QA Manual.

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

k. Records of OSRC meetings and NSG reports.

l. Records of the service lives of all safety related hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of review performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee-initiated changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by specification 6.10.2.

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and approval pursuant to 6.5.2. by the Station Manager.

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. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s);

    b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations; and

    c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.

2. Shall become effective upon review and approval pursuant to 6.5.2. by the Station Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 [Deleted]