ABB
ASEA BROWN BOVERI

April 21, 1993
LD-93-069

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: System 80+™ Information for Issue Closure

Dear Sirs:

The attachments to this letter provide markups OF CESSAR-DC for closure of DSER issues. These markups, and others to be discussed with NRC staff in forthcoming meetings, will be printed as part of Amendment O in May, 1993.

Attachment 1 provides minor technical revisions to seven DSER issues being resolved by the Plant Systems Branch, as the result of the 3% increase in core power level. Since the changes are not large, it is expected that any staff review performed to date remains valid. Attachment 2 provides revisions to seven DSER issues related to the Initial Test Program (Chapter 14) as a result of recent telephone calls with NRC staff. Attachment 3 provides a listing of computer run microfische provided to NRC staff as a result of the April 6-8, 1993, meeting on piping analysis with the Engineering and Geosciences Branch and Mechanical Engineering Branch.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for
C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)
P. Lang (DOE)

D032

260023

ABB Combustion Engineering Nuclear Power

Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500
Telephone (203) 688-1911
Fax (203) 285-9512
Telex 99297 COMBEN WSOR

ATTACHMENT 1

DSER RESPONSES ENCLOSED - 4/20/93

OPEN ITEM NUMBER	REV	DATE	NRC BRANCH
11.1-1	B	4/20/93	SPLB
11.1-2	B	4/20/93	SPLB
11.2-1	B	4/20/93	SPLB
11.2-2	B	4/20/93	SPLB
11.3-4	B	4/20/93	SPLB
11.3-5	B	4/20/93	SPLB

COL ITEM NUMBER	REV	DATE	NRC BRANCH
15.3.10-1	B	4/20/93	SPLB

K. J. Montagu
4/20/93

SPLB

DSER Open Item 11.1-1

CESSAR Section 11.1 requires more information to enable staff to evaluate the radioactive waste system.

Proposed Open Item 11.1-1 Resolution

The effluent analyses for LWMS and GWMS have been revised to evaluate compliance with 10 CFR 20, Appendix B, Table II based on a 1% failed fuel rate. Proposed revisions to CESSAR-DC Sections 11.1, 11.2 and 11.3 are attached.

The decontamination factors, shown in Table 11.2-3, are consistent with those provided in NUREG-0017, Revision 1. Radwaste ion-exchange system decontamination factors are based on recent industry experience with such systems, including the use of cesium-selective zeolite media which are added to organic ion-exchange materials to improve the performance of liquid waste treatment ion-exchange systems (and that additional processing can be performed, as necessary, to achieve the stated performance levels). The effective decontamination factor for each flow stream is calculated based on the decontamination factors shown in Table 11.2-3 and the simplified liquid release pathway shown in Figure 11.2-2.

The computer code PWR-GALE, based on NUREG-0017, Revision 1 methodology, accounts for operator error and adds an additional 0.16 Ci/yr to the release from the LWMS. Table 11.2-1, Note 3 states, "The total is adjusted to include 0.16 Curies attributable to operational occurrences that results from unplanned releases."

The carrier gas flow rate (1 scfm) in Section 11.3 is based on the EPRI Utility Requirements Document, Chapter 12 specifications.

The gaseous release points are graphically represented in Figure 11.3-2.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

The average quantity of radioactive material released to the environment during normal operation including anticipated operational occurrences is calculated using PWR-GALE Code (Reference 1) and is based on guidance provided in NUREG-0017 (Reference 2). The adequacy of radioactive waste management systems is demonstrated by verifying compliance with 10 CFR 50, Appendix I offsite radiological release objectives using the NUREG-0017 "expected" source term basis.

Design basis "maximum" source terms are used in plant radiation shielding design and accidental offsite release evaluations. The use of design basis "maximum" source terms in shielding and J accident calculations allows for short-term increases in reactor coolant concentration above the NUREG-0017 "expected" average concentrations. Design basis "maximum" source terms are addressed in Chapter 12. based on a 1% failed fuel rate assumption

S

The adequacy of radioactive waste management systems is also demonstrated by verifying compliance with the ~~instantaneous~~ offsite release rate and concentration limits of 10 CFR 20. ~~Both "expected" and "maximum"~~ source terms are used to demonstrate compliance with 10 CFR 20 radiological protection criteria. ~~The "expected" source terms are used to demonstrate the long-term operational adequacy of radioactive waste management systems. The ability to maintain liquid and airborne radionuclide concentrations below 10 CFR 20 instantaneous limits under short-term "maximum" source term conditions is also addressed.~~

11.1.1 ANTICIPATED PRIMARY COOLANT CONCENTRATIONS

Reactor Coolant System (RCS) radionuclide activity concentration source terms for normal reactor operating conditions, including anticipated operational occurrences, are developed as a basis for a) calculating routine radioactive releases in station effluents, b) calculating radionuclide concentrations in radioactive waste management and other plant systems during normal operation, and c) ensuring that occupational radiation exposures are as low as reasonably achievable (ALARA). A description of reactor coolant and plant system radiation sources used as the basis for shield I design calculations is provided in Section 12.2.

11.1.1.1 Fission Product Activities

The concentrations of radioactive fission product isotopes in primary coolant under normal reactor operating conditions are calculated by methods developed in NUREG-0017. The parameters used in the coolant fission product source term

calculations are summarized in Table 11.1.1-1. The calculated RCS fission product activity concentrations ~~are summarized~~ in Table 11.1.1-2.

INSERT A

11.1.1.2 Corrosion and Activation Products

, maximum, and 1% fuel failure rate

The concentrations of radioactive corrosion and activation products in primary coolant under normal reactor operating conditions (i.e., Na-24, Cr-51, Mn-54, Fe-55, Fe-59, Co-58, Co-60, Zn-65, W-187 and Np-239) are included in Table 11.1.1-2. Corrosion and activation product concentrations are calculated by methods developed in NUREG-0017. I

11.1.1.3 Tritium Production in Reactor Coolant

The principal sources of tritium production in a pressurized water reactor (PWR) are from ternary fission and neutron induced reactions in boron, lithium and deuterium that are present in the coolant, borated shim rods and Control Element Assemblies (CEAs). The tritium produced in the coolant contributes immediately to the overall tritium activity while the tritium produced by fission and neutron capture in the CEAs and borated shim rods contributes to the overall tritium activity via release through the cladding.

11.1.1.3.1 Activation Sources of Tritium

The activation reactions producing tritium are as shown in Table 11.1.1-3. The tritium production from reactions 5 and 6 (B-11 and N-14 sources) is insignificant due to low cross section and/or abundance and can be neglected. Reactions 1-4 (from B-10, lithium, and deuterium) are the major sources of tritium in the coolant, CEAs and borated shim rods. I

The tritium production from the above sources is determined by the following expressions:

Tritium Formation Rate = Production Rate - Decay

$$\frac{dN}{dt} = \Sigma_a \phi - \lambda N$$

$$N = \frac{\Sigma_a \phi}{\lambda} (1 - e^{-\lambda t}), \text{ atoms/cm}^3 \text{ at time (t)}$$

$$\text{activity (curies)} = V \lambda N \times 2.7 \times 10^{-11}$$
$$= \Sigma_a \phi \, (1 - e^{\lambda t}) \, V \times 2.7 \times 10^{-11}$$

1570

to surface waters and the atmosphere calculated using NUREG-0017 methodology plus the annual decay of tritium in tritiated water sources) to the maximum and average tritium activity production rates provided in Table 11.1.1-5. For both the maximum and average tritium production cases, total tritium production (1452 and 1211 Ci/yr, respectively) does not exceed the total liquid and airborne releases accounted for in the Section 11.2 and 11.3 offsite radiological impact evaluations (1560 Ci/yr) plus decay (280 Ci/yr - based on an average tritiated water concentration of 1 μCi/gm and a total tritiated water inventory of 5.0E+09 gms). The conclusion of this comparison is that RCS coolant concentrations will likely remain below both design basis maximum and average tritium concentration levels and that annual average environmental releases will likely be lower than calculated using NUREG-0017 methodology. Assuming a station tritiated water inventory of 5.0E+09 gms (see Table 11.1.1-7) and the tritium production rates summarized in Table 11.1.1-5, an average primary system bleed rate of 183 GPD to the Liquid Waste Management System (see Section 11.2.6) will be sufficient to maintain reactor coolant tritium concentrations below the assumed average and maximum levels. J I

193

Reduced levels of tritium in the RCS and environmental effluents are expected benefits of higher levels of fuel performance and the specification of enriched lithium for RCS coolant chemistry control. Both of these measures help to reduce tritium production during power operation below historical levels experienced at operating PWRs (on which NUREG-0017 methodology is based). Plant operational procedures shall include provisions for controlled discharges of RCS water (following processing in the Liquid Waste Management System) as necessary to control RCS tritium levels. J

Replace with Attached Table 11.1.1-1

TABLE 11.1.1-1

PRIMARY AND SECONDARY COOLANT ACTIVITY CONCENTRATION BASES

Core Power Level, MWt	3800
Fuel Cycle Duration, Effective Full Power Days	438
Plant Capacity Factor	0.8
Reactor Coolant Mass Including Pressurizer, lbs	5.713E+05
Letdown Purification Flow, gpm	72
Failed Fuel Fraction	(1)
Purification IX Removal Efficiencies	
Iodines	100
Cs, Rb	2
Others	50
CVCS Gas Stripper Operation	Continuous
Beginning of Cycle Boron Concentration, ppm	1200
Initial Deborating IX Service Level, ppm (2)	30
Total Steam Flow Rate, lbs/hr (3)	1.712E+07
Mass of Liquid in Each SG, lbs	2.810E+05
Total SG Blowdown Flow, lb/hr (3)	3.43E+04
Primary to Secondary Leak Rate, lbs/day	75
Steam Generator Partition Factors	
Tritium, Noble Gases	1.0
Iodines	0.01
All Other Nuclides	0.005
Condensate Demineralizer Flow Fraction (4)	1

I

NOTES:
(1) Failed fuel rate consistent with NUREG-0017, Revision 1 model.
(2) Boron concentration at which shim bleed to reduce primary coolant boron level discontinued.
(3) Values are for sum of both steam generators.
(4) Powdex demineralizers used for condensate clean-up.

Replace with Attached 11.1.1-2

TABLE 11.1.1-2

(Sheet 1 of 2)

PRIMARY AND SECONDARY ACTIVITY DURING NORMAL OPERATION

Primary Coolant (μCi/gm)		Secondary Coolant (μCi/gm)	
Nuclide	Water	Water	Steam (1)
H-3	1.00E+00	1.00E-03	1.00E-03
Sr-89	1.40E-04	3.92E-09	1.96E-11
Sr-90	1.20E-05	3.37E-10	1.69E-12
Sr-91	9.60E-04	2.01E-08	1.01E-10
Y-91	5.20E-06	1.44E-10	7.20E-13
Y-93	4.20E-03	8.59E-08	4.29E-10
Zr-95	3.90E-04	1.10E-08	5.50E-11
Nb-95	2.80E-04	7.56E-09	3.78E-11
Mo-99	6.40E-03	1.73E-07	8.65E-10
Tc-99m	4.70E-03	8.02E-08	4.01E-10
Ru-103	7.50E-03	2.13E-07	1.06E-09
Rh-103m	0.00E+00	0.00E+00	0.00E+00
Ru-106	9.00E-02	2.54E-06	1.27E-08
Rh-106	0.00E+00	0.00E+00	0.00E+00
Ag-110m	1.30E-03	3.64E-08	1.82E-10
Ag-110	0.00E+00	0.00E+00	0.00E+00
Sb-124	0.00E+00	0.00E+00	0.00E+00
Te-129m	1.90E-04	5.36E-09	2.68E-11
Te-129	2.40E-02	1.70E-07	8.50E-10
Te-131m	1.50E-03	3.77E-08	1.88E-10
Te-131	7.70E-03	2.29E-08	1.15E-10
I-131	4.50E-02	7.39E-07	7.39E-09
Te-132	1.70E-03	4.57E-08	2.29E-10
I-132	2.10E-01	1.84E-06	1.84E-08
I-133	1.40E-01	2.12E-06	2.12E-08
Cs-134	7.10E-03	2.90E-07	1.45E-09
I-135	2.60E-01	3.29E-06	3.29E-08
Cs-136	8.70E-04	3.50E-08	1.75E-10
Cs-137	9.40E-03	3.86E-07	1.93E-09
Ba-137m	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.30E-02	3.58E-07	1.79E-09
La-140	2.50E-02	6.47E-07	3.24E-09
Ce-141	1.50E-04	4.19E-09	2.09E-11
Ce-143	2.80E-03	6.98E-08	3.49E-10
Pr-143	0.00E+00	0.00E+00	0.00E+00
Ce-144	3.90E-03	1.10E-07	5.50E-10
Pr-144	0.00E+00	0.00E+00	0.00E+00
Na-24	4.70E-02	1.06E-06	5.30E-09
P-32	0.00E+00	0.00E+00	0.00E+00

I

Replace with Attached 11.1.1-2

TABLE 11.1.1-2 (Cont'd)

(Sheet 2 of 2)

PRIMARY AND SECONDARY ACTIVITY DURING NORMAL OPERATION

Primary Coolant (μCi/gm)		Secondary Coolant (μCi/gm)	
Nuclide	Water	Water	Steam (1)
Cr-51	3.10E-03	8.94E-08	4.47E-10
Mn-54	1.60E-03	4.47E-08	2.24E-10
Fe-55	1.20E-03	3.37E-08	1.68E-10
Fe-59	3.00E-04	8.25E-09	4.12E-11
Co-58	4.60E-03	1.31E-07	6.55E-10
Co-60	5.30E-04	1.51E-08	7.55E-11
Ni-63	0.00E+00	0.00E+00	0.00E+00
Zn-65	5.10E-04	1.44E-08	7.20E-11
W-187	2.50E-03	6.10E-08	3.05E-10
Np-239	2.20E-03	5.83E-08	2.92E-10
Kr-85m	1.24E-01	---	2.30E-08
Kr-85	6.65E-03	---	1.21E-09
Kr-87	1.45E-01	---	2.54E-08
Kr-88	2.41E-01	---	4.45E-08
Xe-131m	4.01E-02	---	7.23E-09
Xe-133m	1.36E-02	---	2.55E-09
Xe-133	2.60E-01	---	4.73E-08
Xe-135m	1.37E-01	---	2.49E-08
Xe-135	5.03E-01	---	9.34E-08
Xe-137	3.64E-02	---	6.65E-09
Xe-138	1.26E-01	---	2.31E-08
All Others	5.47E-01	2.14E-06	1.07E-08

NOTES: (1) Calculated assuming 0.5% moisture carryover for particulate species and 1% iodine carryover with the steam.

I

TABLE 11.1.1-7

IN-PLANT TRITIATED WATER SOURCES

Component	Volume (1) (Gallons)
Reactor Coolant System	107,000
Spent Fuel Pool	231,000
Reactor Makeup Water Tank	178,000 ~~168,000~~ (2)
Holdup Tank	190,000 ~~174,000~~ (2)
In-Containment Refueling Water Storage Tank	546,000
TOTAL:	~~1,226,000~~ 1,252,000

I

NOTES: (1) Volumes used in the calculation of in-plant tritium concentrations. The calculation uses minimum tritiated water volume assumptions to result in maximum tritium concentration estimates.

(2) Value represents 40% of component/system capacity.

Add Attached

Table 11.1.1-9

INSERT A:

For the purposes of demonstrating compliance with 10 CFR 20, Appendix B, Table II limiting concentrations for radioactive materials in unrestricted areas, concentrations of radioactive fission product isotopes in the primary coolant under "limiting" 1% fuel failure rate conditions are calculated by the Combustion Engineering DAMSAM computer code. The calculated RCS fission product activity concentrations for 1% fuel failure rate operation are summarized in Table 11.1.1-9.

TABLE 11.1.1-1

PRIMARY AND SECONDARY COOLANT ACTIVITY CONCENTRATION BASES

Core Power Level (MWt)	3931
Fuel Cycle Duration (Effective Full Power Days)	438
Plant Capacity Factor	0.8
Reactor Coolant Mass Including Pressurizer (lbs)	6.64E+05
Letdown Purification Flow (gpm)	72
Failed Fuel Fraction	[1]
Purification IX Removal Efficiencies	
Iodines	100
Cs, Rb	2
Others	50
CVCS Gas Stripper Operation	Continuous
Beginning of Cycle Boron Concentration (ppm)	1200
Initial Deborating IX Service Level (ppm) [2]	30
Total Steam Flow Rate (lbs/hr) [3]	1.76E+07
Mass of Liquid in Each SG (lbs)	2.13E+05
Total SG Blowdown Flow (lb/hr) [3]	3.53E+04
Primary to Secondary Leak Rate (lbs/day)	75
Steam Generator Partition Factors	
Tritium, Noble Gases	1.0
Iodines	0.01
All Other Nuclides	0.005
Condensate Demineralizer Flow Fraction [4]	0.65

Notes:

[1] Failed fuel rate consistent with NUREG-0017, Revision 1 model.
[2] Boron concentration at which shim bleed to reduce primary coolant boron level discontinued .
[3] Values are for sum of both steam generators.
[4] Deep bed demineralizers used for condensate clean-up.

TABLE 11.1.1-2
(SHEET 1 OF 2)

PRIMARY AND SECONDARY ACTIVITY DURING NORMAL OPERATION

	Primary Coolant (uCi/gm)	Secondary Coolant (uCi/gm)	
Nuclide	Water	Water	Steam
H -3	1.00E+00	1.00E-03	1.00E-03
Sr-89	1.66E-04	1.22E-08	6.10E-11
Sr-90	1.43E-05	1.06E-09	5.30E-12
Sr-91	1.01E-03	4.42E-08	2.21E-10
Y -91 M	4.47E-04	3.53E-09	1.77E-11
Y -91	6.17E-06	4.50E-10	2.25E-12
Y -93	4.45E-03	1.91E-07	9.55E-10
Zr-95	4.63E-04	3.43E-08	1.72E-10
Nb-95	3.32E-04	2.35E-08	1.18E-10
Mo-99	7.36E-03	5.01E-07	2.51E-09
Tc-99 M	4.84E-03	1.59E-07	7.95E-10
Ru-103	8.89E-03	6.64E-07	3.32E-09
Rh-103M	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.07E-01	7.96E-06	3.98E-08
Rh-106	0.00E+00	0.00E+00	0.00E+00
Ag-110M	1.54E-03	1.14E-07	5.70E-10
Ag-110	0.00E+00	0.00E+00	0.00E+00
Sb-124	0.00E+00	0.00E+00	0.00E+00
Te-129M	2.25E-04	1.67E-08	8.35E-11
Te-129	2.34E-02	2.49E-07	1.25E-09
Te-131M	1.68E-03	1.01E-07	5.05E-10
Te-131	7.43E-03	3.07E-08	1.54E-10
I -131	5.31E-02	2.45E-06	2.45E-08
Te-132	1.96E-03	1.33E-07	6.65E-10
I -132	2.09E-01	3.36E-06	3.36E-08
I -133	1.55E-01	6.04E-06	6.04E-08
Cs-134	9.10E-03	7.71E-07	3.86E-09
I -135	2.70E-01	7.65E-06	7.65E-08
Cs-136	1.09E-03	9.08E-08	4.54E-10
Cs-137	1.20E-02	1.03E-06	5.15E-09
Ba-137M	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.53E-02	1.10E-06	5.50E-09
La-140	2.83E-02	1.79E-06	8.95E-09
Ce-141	1.78E-04	1.30E-08	6.50E-11
Ce-143	3.14E-03	1.89E-07	9.45E-10
Pr-143	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.63E-03	3.44E-07	1.72E-09
Pr-144	0.00E+00	0.00E+00	0.00E+00
Na-24	5.08E-02	2.55E-06	1.28E-08
P -32	0.00E+00	0.00E+00	0.00E+00
Cr-51	3.67E-03	2.78E-07	1.39E-09
Mn-54	1.90E-03	1.40E-07	7.00E-10
Fe-55	1.43E-03	1.05E-07	5.25E-10

TABLE 11.1.1-2
(SHEET 2 OF 2)

PRIMARY AND SECONDARY ACTIVITY DURING NORMAL OPERATION

	Primary Coolant (uCi/gm)	Secondary Coolant (uCi/gm)	
Nuclide	Water	Water	Steam
Fe-59	3.56E-04	2.57E-08	1.29E-10
Co-58	5.46E-03	4.08E-07	2.04E-09
Co-60	6.30E-04	4.74E-08	2.37E-10
Ni-63	0.00E+00	0.00E+00	0.00E+00
Zn-65	6.06E-04	4.52E-08	2.26E-10
W -187	2.77E-03	1.58E-07	7.90E-10
Np-239	2.52E-03	1.66E-07	8.30E-10
Kr-85 M	1.15E-01	---	2.09E-08
Kr-85	6.88E-03	---	1.26E-09
Kr-87	1.31E-01	---	2.39E-08
Kr-88	2.21E-01	---	4.03E-08
Xe-131M	4.13E-02	---	7.53E-09
Xe-133M	1.37E-02	---	2.50E-09
Xe-133	2.66E-01	---	4.85E-08
Xe-135M	1.22E-01	---	2.23E-08
Xe-135	4.78E-01	---	8.73E-08
Xe-137	3.24E-02	---	5.91E-09
Xe-138	1.13E-01	---	2.06E-08
All Others	5.29E-01	4.21E-05	2.11E-07

NOTES:

[1] Calculated assuming 0.5% moisture carryover for particulate species and 1% iodine carryover with the steam.

TABLE 11.1.1-9
(Sheet 1 of 2)

PRIMARY ACTIVITY DURING LIMITING
1% FUEL FAILURE RATE OPERATION (1)

Nuclide	DAMSAM 1% Failed Fuel Concentration (uCi/gm)
Sr89	3.92E-03
Sr90	1.37E-04
Sr91	5.80E-03
Y91	5.62E-04
Y93	1.39E-04
Zr95	6.13E-04
Nb95	6.07E-04
Mo99	3.38E-01
Tc99m	1.95E-01
Ru103	2.09E-04
Rh103m	---
Ru106	7.64E-05
Rh106	---
Ag110m	---
Ag110	---
Sb124	---
Te129m	7.15E-03
Te129	7.61E-03
Te131m	3.38E-02
Te131	1.32E-02
I131	2.96E+00
Te132	2.35E-01
I132	8.01E-01
I133	4.25E+00
Cs134	2.08E-01
I135	2.39E+00
Cs136	5.64E-02
Cs137	3.66E-01
Ba137m	3.45E-01
Ba140	4.80E-03
La140	1.62E-03
Ce141	1.80E-04
Ce143	5.03E-04
Pr143	6.46E-04
Ce144	4.59E-04
Pr144	4.57E-04

TABLE 11.1.1-9
(Sheet 2 of 2)

PRIMARY ACTIVITY DURING LIMITING
1% FUEL FAILURE RATE OPERATION (1)

Nuclide	DAMSAM 1% Failed Fuel Concentration (uCi/gm)
Na24	5.08E-02 (2)
Cr51	3.67E-03 (2)
Mn54	1.90E-03 (2)
Fe55	1.43E-03 (2)
Fe59	3.56E-04 (2)
Co58	5.46E-03 (2)
Co60	6.30E-04 (2)
Zn65	6.06E-04 (2)
W187	2.77E-03 (2)
Np239	2.52E-03 (2)
Kr85m	9.03E-01
Kr85	2.10E-02
Kr87	8.81E-01
Kr88	2.20E+00
Xe131m	2.19E-01
Xe133m	5.75E-02
Xe133	2.87E+01
Xe135m	6.96E-01
Xe135	4.08E+00
Xe137	1.65E-01
Xe138	5.94E-01
H3	1.00E+00 (3)

NOTES:
(1) Continuous gas stripping assumed.
(2) NUREG-0017 basis corrosion product source terms.
(3) Assumed maximum based on in-plant tritium calculations.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

The design objectives of the Liquid Waste Management System (LWMS) is to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor radioactive liquid waste. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The processed liquid radioactive waste is sampled prior to release from Waste Monitor Tanks and radiation monitors are provided in the discharge line to provide for a controlled monitored release. The concentration of the liquid effluent at the ~~potable water source~~ released during normal operation, including anticipated operational occurrences, is below concentrations specified in 10 CFR 20 and meet the As Low As Reasonably Achievable (ALARA) criteria of 10 CFR 50, Appendix I.

site discharge point

11.2.1 DESIGN BASES

11.2.1.1 Criteria and Evaluation

The Liquid Waste Management System (LWMS) is designed in accordance with the acceptance criteria defined in the Standard Review Plan, Section 11.2. The design criteria are the following:

A. Effluents normally released to unrestricted areas must meet the limiting requirements of 10 CFR 20 and meet the ALARA objectives of 10 CFR 50, Appendix I. J

The LWMS intermittently discharges liquid effluent in batches to the environment. Table 11.2-1 provides an estimate of the annual liquid effluent releases (Ci/yr) based on results from PWR-GALE using NUREG-0017 methodology. Assumptions used to calculate the annual release rate are discussed in Section 11.2.6. This estimated annual release rate is used to calculate the estimated annual dose to the maximum individual. These results are listed in Table 11.2-4. This analysis assures that effluents during normal operation and anticipated operational occurrences meet 10 CFR 50, Appendix I objectives.

The LWMS is designed to ensure that normal releases to unrestricted areas are within 10 CFR 20, Appendix B, Table II, Column 2 maximum permissible concentrations based on the design basis source term. Section 11.2.7 provides a detailed discussion regarding the methodology used to calculate the concentration of the effluent at the ~~potable water source~~. The results of this analysis assure that the

site discharge point

Maximum in unrestricted area

concentration of the liquid effluent at the potable water source are well within 10 CFR 20, Appendix B, Table II, Column 2 maximum permissible concentrations.

B. The system must contribute to meeting the performance design objectives in that it must never interfere with the normal station operation including anticipated operational occurrences.

The LWMS is a non-nuclear safety related system. It has no accident mitigation functions. The LWMS is designed in accordance with requirements in ANSI/ANS 55.2-1976 and Regulatory Guide 1.143. This includes the following features:

1. The LWMS is designed with sufficient redundancy to tolerate a single major component failure and process radioactive liquid waste during normal operation, including anticipated occurrences.

2. The LWMS is designed with sufficient storage capacity and redundancy to accommodate an increase in demand during normal operation of the plant. J

C. Releases of radioactive materials to the environment must be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61 and 64).

The release of liquid waste requires an operator action. Prior to release through the plant discharge, radioactive liquid waste is sampled. The LWMS is also provided with a radiation monitor which monitors in the discharge line downstream from the Waste Monitor Tanks. In the event that the concentration of the discharge may exceed 10 CFR 20 limits, the radiation monitor would terminate the discharge. Section 11.5, Radiation Monitoring System, provides a detailed discussion regarding the radiation monitoring for the LWMS.

D. Accidental releases of radioactive materials from a single component of the LWMS must not result in offsite doses which exceed the guidelines of 10 CFR 20.

The LWMS is housed in a structure designed in accordance with requirements specified in Regulatory Guide 1.143. The Radwaste Building provides a seismic containment facility or bathtub to contain the maximum inventory of liquid in the building. It is assumed that the Radwaste Building is physically connected to the Nuclear Annex. This assumption will be confirmed once a site is specified. Therefore,

Access is provided to manually load vessels if appropriate. The normal disposition of a fully expended (high differential pressure, high radiation or loss of desired isotopic removal capability) media is sluicing to the Low Activity Spent Resin Tank in the Solid Waste Management System (SWMS) or directly to a disposal container for processing and shipment offsite.

11.2.2.2.6 Provisions for Mobile Equipment

In addition to the simplicity and versatility provided by a bag filter/ion-exchange based process system, the modular approach of movable skid mounted process equipment further promotes process flexibility within a space limited process area. This is especially true since it is anticipated that it may be advantageous to use additional mobile treatment or direct solidification equipment at times. This may be true because of changing waste streams or changing economics of processing, shipping and burial. Piping provisions are made to permit connection of mobile process equipment while using the installed Waste Collection Tanks, process pumps, and Waste Monitor Tanks. Rapid re-alignment of a process flow path can be accomplished using remote operated valves (outside of skid shielding), quick-connect fittings and flexible high pressure industrial hoses.

11.2.2.2.7 Steam Generator Drain Tank

One Steam Generator Drain Tank is optionally provided so that in the event of a significant steam generator tube leak, the affected steam generator can be drained expeditiously after isolation. This water will generally be unsuitable for recycling because it will be chemically unsuitable for the Reactor Coolant System and radioactively undesirable for condensate makeup. The Steam Generator Drain Tank is sized for three Steam Generator volumes of feedwater to allow for a rinse. The tank is made of stainless steel to permit use as general liquid radwaste surge capacity; however, it is normally kept empty.

11.2.2.2.8 Dilution Pumps

A dedicated source of dilution water is necessary to maintain liquid waste effluent concentrations in the environment below 10 CFR 20 concentration limits and 10 CFR 50 Appendix I as low as reasonably achievable offsite dose objectives. The dilution flow is provided by four centrifugal pumps. The pumps are sized such that any two pumps can provide a minimum of 100 CFS dilution flow to facilitate LWMS discharges.

Although the cost-benefit analysis is deferred to site-specific environmental reports, it is fully expected that the LWMS, as described in this design certification, will pass the as low as reasonably achievable test for most proposed sites without modification. In any case, LWMS modifications resulting from site-specific cost-benefit analyses will reduce the maximum individual doses presented in Table 11.2-4. I

11.2.7 CONCENTRATION OF NORMAL EFFLUENTS

The Liquid Waste Management System (LWMS) processes liquid waste prior to release to the environment. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The process liquid radioactive waste is sampled prior to release from Waste Monitor Tanks and radiation monitors are provided in the discharge line to provide for a controlled monitored release. The concentration at the potable water source resulting from releases during normal operation, including anticipated operational occurrences was analyzed to verify that it is less than 10 CFR 20, Appendix B, Table II, Column 2 Maximum Permissible Concentration.

11.2.7.1 Analysis of Effects and Consequences J

A. Bases

For the purpose of this analysis, the following assumptions were made to estimate the concentration of the liquid effluent at the potable water source for the design basis source term and the normal operating source term:

1. The system discharges intermittently at an average of approximately 11200 gallons per day shown below.

Type of Waste	Discharge Flow Rate (gpd)	Reference
Shim Bleed	183	Table 11.2-2
Clean Waste	70	Table 11.2-2
Dirty Waste	3200	Table 11.2-2
Detergent Waste	540	NUREG-0017
Turbine Building Drains	7200	NUREG-0017

2. The source term is based on the concentration of the liquid in the Waste Monitor Tank. All effluent is assumed to be at this concentration for conservatism.

Replace with attached INSERT A

Replace with Insert A

a. The LWMS may be occasionally operated at design basis source term conditions. During these conditions, the concentration of the liquid in the Waste Monitor Tank is based on the design basis source term (0.25% failed fuel rate).

b. Typically, the LWMS is operated at the average normal operating source term calculated using

NUREG-0017 methodology. For conservatism, the concentration in the Waste Monitor Tank is calculated based on the shim bleed flow stream processed via the Chemical Volume and Control System (CVCS).

The initial concentration of the shim bleed flow stream, prior to processing via the CVCS, is assumed to be equal to the Primary Coolant Activity (PCA) shown in Table 11.1.1-2. No credit is taken for processing of the flow stream by the LWMS (i.e., the shim bleed is assumed to bypass the LWMS process ion exchangers and filters and be transferred directly from the Equipment Waste Tank to the Waste Monitor Tank for discharge).

The concentration of the shim bleed in the Waste Monitor Tank is calculated as follows: J

$$C_t(i) = 1.0\ PCA * \rho/DF(i)_{CVCS}$$

Where: $C_t(i)$ = Concentration in the Waste Monitor Tank ($\mu Ci/ml$)

PCA = Primary Coolant Activity ($\mu Ci/gm$) (See Table 11.1.1-2)

ρ = Density (gm/ml) = 1.0 gm/ml

$DF(i)_{CVCS}$ = Chemical Volume and Control System (CVCS) Total Process Decontamination Factor (See Table 11.2.3 and Figure 11.2-2)

$DF(i)_{CVCS}$ = 5.0E+5 (Iodine)
= 4.0E+3 (Cs, Rb)
= 5.0E+6 (Other)

3. The dilution flow rate is assumed to be 100 scfs, which is consistent with that assumed in Section 11.2.6.

(Replace with INSERT A)

4. In the absence of site specific information for dilution flow. The dilution factor is 1.74×10^{-4}, based on a dilution flow rate of 100 scfs and discharge rate of 11200 gpd.

B. Methodology

The methodology used to calculate the concentration of the effluent at the potable water source is as follows:

$$C_p(i) = C_t(i) * D_f$$

Where:

$C_p(i)$ = Concentration of the i^{th} isotope at the potable water source (μCi/ml)

Ct(i) = Concentration of the i^{th} isotope in the Waste Monitor Tank (for the design basis failed fuel rate, 0.25%) (μCi/ml)

Df = Dilution Factor

D_f = F_{dis}/F_{dil}

D_f = 1.74×10^{-4}

F_{dis} = Discharge Flow Rate (gpd)
= 11200 gpd

Fdil = Dilution Flow Rate (scfs)
= 100 scfs

C. Results and Conclusions

The estimated concentration of the liquid effluent at the potable water source is shown in Table 11.2-5 and Table 11.2-6. The total average daily concentration of the liquid effluent at the potable water source for the design basis source term and the normal operating source term is 2.76E-1 MPC and 5.81E-2 MPC, respectively. The concentration of isotopes at the potable water source, for both the design basis and normal operating source terms conditions, is well within 10 CFR 20 guidelines.

The rate of radioactive liquid discharges will be based on the available dilution flow and the concentrations of 10 CFR 20, Appendix B, Table II, Column 2. For a dilution flow of 100 scfs, the maximum allowable discharge rate of liquid effluent from the LWMS, during design basis source term conditions and normal operating source term conditions, is approximately 4.05E+4 gpd and 1.92E+5 gpd, respectively. J

The Owner Operator will develop procedures for the operation of the LWMS and release of radioactive liquid effluents from the LWMS to ensure that the concentration of the liquid effluents at the potable water source are within 10 CFR 20 guidelines. J

Replace with INSERT A

Replaced with Attached Table 11.2-1

TABLE 11.2-1

(Sheet 1 of 2)

ANNUAL AVERAGE LIQUID RELEASE SOURCE TERMS(1)
(Curies/yr)

Nuclide	Primary Bleed Waste	Liquid Waste System	Turbine Building Drains	SG Drain Tank(2)	Detergent Waste	Total(3)
Sr-89	0.0	1.00E-05	0.0	7.42E-07	9.00E-05	1.71E-04
Sr-90	0.0	0.0	0.0	6.38E-08	1.00E-05	2.01E-05
Sr-91	0.0	0.0	0.0	3.81E-06	0.0	2.38E-05
Y-91	0.0	0.0	0.0	2.73E-08	8.00E-05	9.00E-05
Y-93	0.0	1.00E-05	0.0	1.63E-05	0.0	8.63E-05
Zr-95	0.0	4.00E-05	0.0	2.08E-06	1.10E-03	1.32E-03
Nb-95	0.0	3.00E-05	0.0	1.43E-06	1.90E-03	2.07E-03
Mo-99	0.0	2.50E-04	1.00E-05	3.28E-05	6.00E-05	1.48E-03
Tc-99m	0.0	2.40E-04	1.00E-05	1.52E-05	0.0	1.33E-03
Ru-103	0.0	7.60E-04	1.00E-05	4.03E-05	2.90E-04	4.42E-03
Rh-103m	0.0	7.60E-04	1.00E-05	0.0	0.0	4.09E-03
Ru-106	0.0	9.98E-03	1.30E-04	4.81E-04	6.90E-03	6.30E-02
Rh-106	0.0	9.98E-03	1.30E-04	0.0	0.0	5.36E-02
Ag-110m	0.0	1.40E-04	0.0	6.89E-06	1.20E-03	1.98E-03
Ag-110	0.0	2.00E-05	0.0	0.0	0.0	1.00E-04
Sb-124	0.0	0.0	0.0	0.0	4.30E-04	4.30E-04
Te-129m	0.0	2.00E-05	0.0	1.02E-06	0.0	1.01E-04
Te-129	0.0	1.00E-05	0.0	3.22E-05	0.0	1.02E-04
Te-131m	0.0	3.00E-05	0.0	7.14E-06	0.0	1.47E-04
Te-131	0.0	0.0	0.0	4.34E-06	0.0	3.43E-05
I-131	0.0	3.28E-03	7.00E-05	1.40E-04	1.60E-03	1.95E-02
Te-132	0.0	8.00E-05	0.0	8.65E-06	0.0	4.19E-04
I-132	0.0	8.00E-05	3.00E-05	3.48E-04	0.0	9.38E-04
I-133	0.0	1.40E-03	1.70E-04	4.01E-04	0.0	6.73E-03
Cs-134	2.10E-04	7.90E-04	1.00E-05	5.49E-05	1.10E-02	1.65E-02
I-135	0.0	2.30E-04	1.80E-04	6.23E-04	0.0	2.78E-03
Cs-136	1.00E-05	7.00E-05	0.0	6.63E-06	3.70E-04	8.07E-04
Cs-137	3.00E-04	1.05E-03	2.00E-05	7.31E-05	1.60E-02	2.33E-02
Ba-137m	2.80E-04	9.90E-04	2.00E-05	0.0	0.0	6.79E-03
Ba-140	0.0	1.09E-03	2.00E-05	6.78E-05	9.10E-04	6.86E-03
La-140	0.0	1.50E-03	3.00E-05	1.23E-04	0.0	8.24E-03
Ce-141	0.0	1.00E-05	0.0	7.93E-07	2.30E-04	3.11E-04
Ce-143	0.0	5.00E-05	0.0	1.32E-05	0.0	3.13E-04
Pr-143	0.0	2.00E-05	0.0	0.0	0.0	1.10E-04
Ce-144	0.0	4.30E-04	1.00E-05	2.08E-05	3.90E-03	6.24E-03
Pr-144	0.0	4.30E-04	1.00E-05	0.0	0.0	2.32E-03
Na-24	0.0	2.60E-04	4.00E-05	2.01E-04	0.0	1.82E-03
P-32	0.0	0.0	0.0	0.0	1.80E-04	1.80E-04

I

Replace with Attach 1 Table 11.2-1

TABLE 11.2-1 (Cont'd)

(Sheet 2 of 2)

ANNUAL AVERAGE LIQUID RELEASE SOURCE TERMS(1)
(Curies/yr)

Nuclide	Primary Bleed Water	Liquid Waste System	Turbine Building Drains	SG Drain Tank(2)	Detergent Waste	Total(3)
Cr-51	0.0	3.00E-04	0.0	1.69E-05	4.70E-03	6.34E-03
Mn-54	0.0	1.80E-04	0.0	8.46E-06	3.80E-03	4.76E-03
Fe-55	0.0	1.30E-04	0.0	6.38E-06	7.20E-03	7.93E-03
Fe-59	0.0	3.00E-05	0.0	1.56E-06	2.20E-03	2.37E-03
Co-58	0.0	4.90E-04	1.00E-05	2.48E-05	7.90E-03	1.05E-02
Co-60	0.0	6.00E-05	0.0	2.86E-06	1.40E-02	1.43E-02
Ni-63	0.0	0.0	0.0	0.0	1.70E-03	1.70E-03
Zn-65	0.0	6.00E-05	0.0	2.73E-06	0.0	3.03E-04
W-187	0.0	3.00E-05	0.0	1.16E-05	0.0	1.82E-04
Np-239	0.0	8.00E-05	0.0	1.10E-05	0.0	4.21E-04
Total:	8.00E-4	3.54E-02	9.20E-04	2.81E-03	8.97E-02	2.90E-01

Tritium Release is 360 Curies/yr

NOTES:
1. 0.0 appearing in this table indicates release is less than 1.0E-05 Curies/yr.
2. One Steam Drain Tank Volume (50,000 gallons at secondary coolant concentration) is assumed to be released per year with no processing.
3. Total is adjusted to include 0.16 Curies attributable to operational occurrences that result in unplanned releases.

Replace with Attached Table 11.2-2

TABLE 11.2-2

(Sheet 1 of 2)

SOURCES, ESTIMATED VOLUMES AND ACTIVITIES OF LIQUID WASTE MANAGEMENT SYSTEM WASTE INPUTS

Liquid Waste Source	Flow Rate (gpd)	Activity[(1)] (pca)	LWMS Collection Tank	Collection Time (Days)	Processing Time (Days)	Discharge Fraction
SHIM BLEED	1830	1.0	Equipment Waste	95 (1)	0.76	0.1
EQUIPMENT DRAINS	250	1.0	(2)	(2)	(2)	(2)
- Reactor Drain Tank - Equipment Drain Tank						
CLEAN WASTE	700	0.2	Equipment Waste	30	0.76	0.1
- Reactor Grade Lab Drains - Aerated Equipment Drains						
DIRTY WASTE	3200	0.021	Floor Drain Waste	6.7	0.76	1.0
- Containment Sump - Plant Floor Drains - Fuel Pool Liner Leakage - Containment Cooling Condensate - Equipment and Area Non-detergent Decon						
STEAM GENERATOR BLOWDOWN	1.0(3)	---	(3)	---	---	0.0
DETERGENT WASTE	(4)	(4)	Laundry and Hot Shower	(4)	(4)	1.0

I

Replace with Attached Table 11.2-2

TABLE 11.2-2 (Cont'd)

(Sheet 2 of 2)

SOURCES, ESTIMATED VOLUMES AND ACTIVITIES OF LIQUID WASTE MANAGEMENT SYSTEM WASTE INPUTS

Liquid Waste Source	Flow Rate (gpd)	Activity(1) (pca)	LWMS Collection Tank	Collection Time (Days)	Processing Time (Days)	Discharge Fraction

I

NOTES:
1. Shim bleed collection time based on 40% of Holdup Tank capacity collection volume.
2. Hydrogenated primary system equipment drain fluids (i.e., Reactor Drain Tank and Equipment Drain Tank inputs) normally recycled directly to the Volume Control Tank. J
3. Full blowdown flow processed by Blowdown System and recycled to the condensate system demineralizers.
4. Detergent wastes collected and discharged without treatment consistent with NUREG-0017 method. I

Replace with Attached Table 11.2-4

TABLE 11.2-4

ESTIMATED DOSES FROM RADIOACTIVE LIQUID EFFLUENTS RELEASED FROM THE STATION

	Annual Dose (mrem/yr)	Appendix I Objective(2) (mrem/yr)
Maximum Whole Body Dose From All Exposure Pathways(1)	2.11 (Adult)	3
Maximum Organ Dose From All Exposure Pathways	2.97 (Child-Bone)	10

I

NOTES:

1. Liquid effluent exposure pathways considered include fish ingestion, drinking water, and external exposure from shoreline sediments.
2. 10 CFR 50, Appendix I numerical design objectives to meet the criterion "As Low As Reasonably Achievable".

Replace with Attached Table 11.2-5

TABLE 11.2-5

(Sheet 1 of 2)

DESIGN BASIS AVERAGE DAILY LIQUID EFFLUENT CONCENTRATION [a]

Nuclide	$C_p(i)$ (µCi/ml)	MPC(i) (µCi/ml)	FMPC(i)
Br-83	6.77E-12	3.00E-06	2.26E-06
Br-84	3.30E-10	3.00E-06	1.10E-04
Br-85	3.82E-11	3.00E-06	1.27E-05
Rb-88	6.43E-10	3.00E-06	2.14E-04
Rb-89	7.12E-10	3.00E-06	2.37E-04
Sr-89	2.08E-11	3.00E-06	6.95E-06
Sr-90	1.30E-12	3.00E-07	4.34E-06
Sr-91	1.29E-11	7.00E-05	1.84E-07
Sr-92	6.08E-12	6.00E-05	1.01E-07
Y-90	1.91E-13	2.00E-05	9.55E-09
Y-91	4.17E-12	3.00E-05	1.39E-07
Y-91m	1.91E-13	3.00E-03	6.37E-11
Y-92	1.20E-12	6.00E-05	2.00E-08
Y-93	2.61E-12	3.00E-05	8.68E-08
Zr-95	5.04E-12	6.00E-05	8.40E-08
Nb-95	5.04E-12	1.00E-04	5.04E-08
Mo-99	1.91E-12	4.00E-05	4.78E-08
Tc-99m	5.73E-09	3.00E-03	1.91E-06
Ru-103	3.82E-12	8.00E-05	4.78E-08
Ru-106	9.55E-13	1.00E-05	9.55E-08
Rh-103m	3.30E-13	1.00E-02	3.30E-11
Te-129m	7.64E-11	2.00E-05	3.82E-06
Te-129	5.91E-11	8.00E-04	7.38E-08
Te-131m	1.91E-10	4.00E-05	4.78E-06
Te-131	6.95E-11	3.00E-06	2.32E-05
Te-132	2.26E-09	2.00E-05	1.13E-04
Te-134	2.26E-10	3.00E-06	7.53E-05
I-131	2.08E-08	3.00E-07	6.95E-02
I-132	6.77E-09	8.00E-06	8.47E-04
I-133	3.30E-08	1.00E-06	3.30E-02
I-134	4.69E-09	2.00E-05	2.34E-04
I-135	1.91E-08	4.00E-06	4.78E-03
Cs-134	3.65E-09	9.00E-06	4.05E-04
Cs-136	1.32E-09	6.00E-05	2.20E-05
Cs-137	3.47E-09	2.00E-05	1.74E-04
Cs-138	2.61E-09	3.00E-06	8.69E-04
Ba-137m	7.12E-15	3.00E-06	2.37E-09
Ba-139	4.52E-12	3.00E-06	1.51E-06
Ba-140	3.13E-11	2.00E-05	1.56E-06

(a) Based on concentration in Waste Monitor Tank for design basis source term during normal operating conditions (See Table 12.2-17).

J

Replace with Attached Table 11.2-5

TABLE 11.2-5 (Cont'd)

(Sheet 2 of 2)

DESIGN BASIS AVERAGE DAILY LIQUID EFFLUENT CONCENTRATION [a]

Nuclide	$C_p(i)$ (µCi/ml)	MPC(i) (µCi/ml)	FMPC(i)
La-140	4.34E-12	2.00E-05	2.17E-07
Ce-141	4.86E-12	9.00E-05	5.40E-08
Ce-143	3.65E-12	4.00E-05	9.12E-08
Ce-144	3.65E-12	1.00E-05	3.65E-07
Pr-144	1.18E-13	3.00E-06	3.94E-08
Mn-54	5.56E-11	1.00E-04	5.56E-07
Co-58	1.60E-10	9.00E-05	1.78E-06
Co-60	1.91E-11	3.00E-05	6.37E-07
Fe-59	1.04E-11	5.00E-05	2.08E-07
Cr-51	1.08E-10	2.00E-03	5.38E-08
H-3	4.97E-04	3.00E-03	1.66E-01
		Total:	2.76E-01 MPC

J

(a) Based on the concentration in Waste Monitor Tank for the design basis source term during normal operating conditions (See Table 12.2-17).

Delete this Table

TABLE 11.2-6

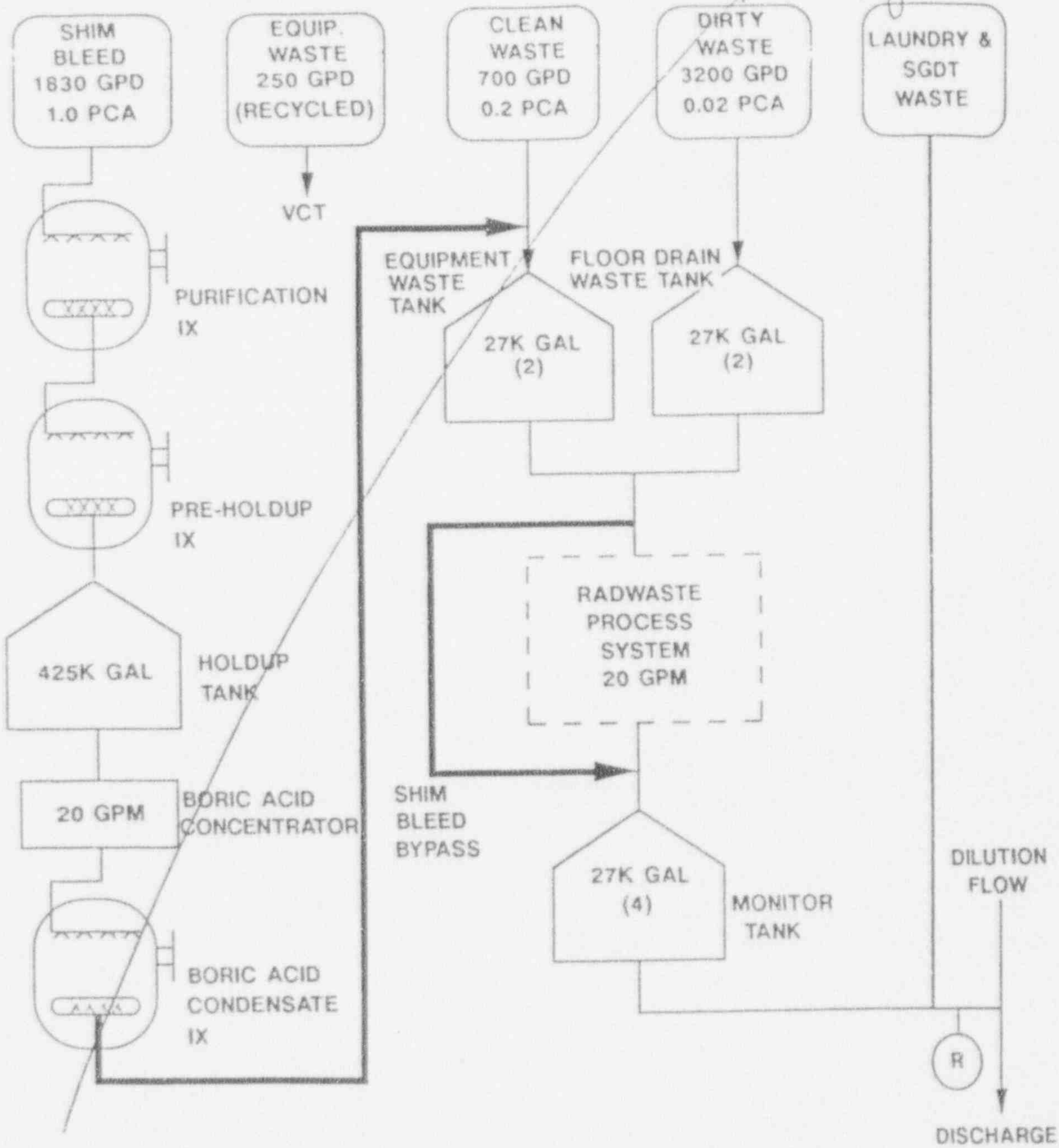
NORMAL OPERATING AVERAGE DAILY LIQUID EFFLUENT CONCENTRATION [a]

Nuclide	$C_p(i)$ (μCi/ml)	MPC(i) (μCi/ml)	FMPC(i)
Sr-89	4.86E-15	3.00E-06	1.62E-09
Sr-90	4.17E-16	3.00E-07	1.39E-09
Sr-91	3.34E-14	7.00E-05	4.76E-10
Y-91	1.81E-16	3.00E-05	6.02E-12
Y-93	1.46E-13	3.00E-05	4.86E-09
Zr-95	1.35E-14	6.00E-05	2.26E-10
Nb-95	9.73E-15	1.00E-04	9.73E-11
Mo-99	2.22E-13	4.00E-05	5.56E-09
Tc-99m	1.63E-13	3.00E-03	5.44E-11
Ru-103	2.61E-13	8.00E-05	3.26E-09
Ru-106	3.13E-12	1.00E-05	3.13E-07
Ag-110m	4.52E-14	3.00E-05	1.51E-09
Te-129m	6.60E-15	2.00E-05	3.30E-10
Te-129	8.34E-13	8.00E-04	1.04E-09
Te-131m	5.21E-14	4.00E-05	1.30E-09
Te-131	2.67E-13	3.00E-06	8.92E-08
Te-132	5.91E-14	2.00E-05	2.95E-09
I-131	1.56E-11	3.00E-07	5.21E-05
I-132	7.30E-11	8.00E-06	9.12E-06
I-133	4.86E-11	1.00E-06	4.86E-05
I-135	9.03E-11	4.00E-06	2.26E-05
Cs-134	3.08E-10	9.00E-06	3.43E-05
Cs-136	3.78E-11	6.00E-05	6.30E-07
Cs-137	4.08E-10	2.00E-05	2.04E-05
Ba-140	4.52E-13	2.00E-05	2.26E-08
La-140	8.68E-13	2.00E-05	4.34E-08
Ce-141	5.21E-15	9.00E-05	5.79E-11
Ce-143	9.73E-14	4.00E-05	2.43E-09
Ce-144	1.35E-13	1.00E-05	1.35E-08
Na-24	1.63E-12	3.00E-05	5.44E-08
Cr-51	1.08E-13	2.00E-03	5.38E-11
Mn-54	5.56E-14	1.00E-04	5.56E-10
Fe-55	4.17E-14	8.00E-04	5.21E-11
Fe-59	1.04E-14	5.00E-05	2.08E-10
Co-58	1.60E-13	9.00E-05	1.78E-09
Co-60	1.84E-14	3.00E-05	6.14E-10
Zn-65	1.77E-14	1.00E-04	1.77E-10
W-187	8.69E-14	6.00E-05	1.45E-09
Np-239	7.64E-14	1.00E-04	7.64E-10
H-3	1.74E-04	3.00E-03	5.79E-02
		Total:	5.81E-02 MPC

J

(a) Based on the concentration in Waste Monitor Tank for the average normal operating source term (calculated utilizing NUREG-0017 methodology) during normal operating conditions.

Replace with Attached Figure 11.2-2



Amendment J
April 30, 1992

SYSTEM 80+™	SIMPLIFIED LIQUID PATHWAY RELEASE ASSESSMENT PROCESS MODEL	Figure 11.2-2

INSERT A:

11.2.7 CONCENTRATION OF NORMAL EFFLUENTS

The Liquid Waste Management System (LWMS) processes liquid waste prior to release to the environment. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The processed liquid radioactive waste is sampled prior to release from Waste Monitor Tanks and radiation monitors are provided in the discharge line to provide for a controlled monitored release. The concentration at the plant discharge, resulting from releases during normal operation, including anticipated operational occurrences, was analyzed to verify that it is less than 10 CFR 20, Appendix B, Table II, Column 2 Maximum Permissible Concentrations for the specified source term (1% failed fuel).

11.2.7.1 <u>Analysis of Effects and Consequences</u>

A. Bases

The concentration of the liquid effluent at the plant discharge is compared to 10 CFR 20, Appendix B, Table II, Column 2 Maximum Permissible Concentrations (MPC) to verify the effluent concentrations are within 10CFR20 limits. The bases for estimating the plant liquid effluent discharge concentration are as follows:

1. The LWMS releases effluent periodically in batches after processing waste liquids with radionuclide concentrations based on the specified 1% failed fuel rate reactor coolant source term.

2. Annual average concentration is calculated based on the design basis conditions (1% failed fuel). The concentration at the plant liquid effluent discharge is averaged over a period not to exceed one year in accordance with 10 CFR 20.103 guidance.

3. It is assumed that the Reactor Coolant System (RCS) is continuously degassed by the CVCS during normal operating conditions. The 1% failed fuel rate reactor coolant equilibrium concentration is calculated using the Combustion Engineering DAMSAM computer code and is presented in Table 9.1.1-9.

4. The average dilution flow rate at the plant discharge is 100 scfs which is consistent with Section 11.2.6. The annual dilution volume is calculated by multiplying the average dilution flow rate (100 scfs) by the number of seconds in one year.

5. The total estimated annual liquid effluent releases, shown in Table 11.2-2, are multiplied by an isotope specific multiplication factor. This multiplication factor is calculated by the division of the 1% failed fuel RCS equilibrium concentration, calculated by the Combustion Engineering DAMSAM computer code, by the RCS equilibrium concentration calculated using PWR-GALE, presented in Table 11.1.1-2, for each isotope.

   For isotopes with a 1% failed fuel rate calculated concentration which is less than PWR-GALE results, the PWR-GALE concentration is used for conservatism. It is assumed that differences in the methodology used to calculate the reactor coolant concentrations are responsible for any differences observed in isotopic concentrations.

6. Since DAMSAM does not calculate the concentration of tritium, the maximum calculated concentration of 1.00 μCi/gm is assumed for 1% failed fuel condition for conservatism.

7. Since DAMSAM does not calculate the concentration of corrosion products, the PWR-GALE numbers are used. The concentration of these radionuclides should not be affected by the fraction of fuel defects.

B. Methodology

To calculate the concentration at the plant discharge, the source term, based on the output from the computer code DAMSAM, is used. This code is used to calculate the reactor coolant equilibrium concentration with continuous degassing based on 1% failed fuel fraction in accordance with Standard Review Plan Section 11.3. The resulting reactor coolant radionuclide equilibrium concentrations are divided by the reactor coolant radionuclide concentration determined by PWR-GALE, using NUREG-0017, Revision 1 methodology, to yield a multiplication factor for each isotope. The total annual radionuclide release rates (Ci/yr) of liquid effluent, calculated by PWR-GALE, are multiplied by the multiplication factor and divided by the annual dilution flow rate to calculate the liquid effluent concentrations at the plant discharge. These concentrations are then compared to the Maximum Permissible Concentrations (MPC) to verify compliance with 10 CFR 20, Appendix B, Table II, Column 1 limits.

The methodology used to calculate the concentration of the effluent, averaged over a period of one year at the plant discharge, is as follows:

$$C_{dis}(i) = \frac{R(i) \times MF(i)}{F_{dil} \times CF}$$

Where:

$C_{dis}(i)$ = Concentration at the plant discharge of the ith isotope (μCi/ml)

R(i) = Total annual release rate of the ith isotope (Ci/yr) (Table 11.2-1)

MF(i) = Multiplication Factor for the ith isotope

$$MF(i) = \frac{RCS(i)_{DAMSAM}}{RCS(i)_{GALE}}$$

F_{dil} = Dilution Flow Rate
= 100 cfs (Section 11.2.6)

CF = Conversion Factor

$$CF = 8.94E+5 \, \frac{cm^3\text{-sec-}Ci}{ft^3\text{-}yr\text{-}\mu Ci}$$

The fraction of the MPC for each isotope is calculated as follows:

$$FMPC(i) = \frac{C_{dis}(i)}{MPC(i)}$$

Where: FMPC(i) = Fraction of MPC for the ith isotope

MPC(i) = Maximum Permissible Concentration of the ith isotope (μCi/ml) (10 CFR 20, Appendix B, Table 2, Column 2)

C. Results and Conclusions

The concentration of the liquid effluents at the plant discharge is shown in Table 11.2-5. The resultant concentration at the plant discharge is less than the Maximum Permissible Concentration specified in 10 CFR 20, Appendix B, Table II, Column 2 guidelines.

TABLE 11.2-1
(Sheet 1 of 2)

ANNUAL AVERAGE LIQUID RELEASE SOURCE TERMS
(Curies/yr) [1]

Nuclide	Primary Bleed Waste	Liquid Waste System	Turbine Bldg Drains	Adjusted Total [3]	SG Drain Tank [2]	Detergent Waste	Total [3,4]
Sr-89	0.0	2.00E-05	0.0	8.00E-05	2.31E-06	9.00E-05	1.72E-04
Sr-90	0.0	0.0	0.0	1.00E-05	2.01E-07	1.00E-05	2.02E-05
Sr-91	0.0	0.0	0.0	2.00E-05	8.37E-06	0.0	2.84E-05
Y -91	0.0	0.0	0.0	1.00E-05	8.52E-08	8.00E-05	9.01E-05
Y -91 M	0.0	0.0	0.0	1.00E-05	6.68E-07	0.0	1.07E-05
Y -93	0.0	1.00E-05	1.00E-05	8.00E-05	3.62E-05	0.0	1.16E-04
Zr-95	0.0	5.00E-05	0.0	2.30E-04	6.50E-06	1.10E-03	1.34E-03
Nb-95	0.0	4.00E-05	0.0	1.80E-04	4.45E-06	1.90E-03	2.08E-03
Mo-99	0.0	2.90E-04	2.00E-05	1.42E-03	9.49E-05	6.00E-05	1.57E-03
Tc-99 M	0.0	2.80E-04	1.00E-05	1.32E-03	3.01E-05	0.0	1.35E-03
Ru-103	0.0	9.00E-04	3.00E-05	4.20E-03	1.26E-04	2.90E-04	4.62E-03
Rh-103M	0.0	9.00E-04	3.00E-05	4.20E-03	0.0	0.0	4.20E-03
Ru-106	0.0	1.19E-02	4.00E-04	5.51E-02	1.51E-03	8.90E-03	6.55E-02
Rh-106	0.0	1.19E-02	4.00E-04	5.51E-02	0.0	0.0	5.51E-02
Ag-110M	0.0	1.70E-04	1.00E-05	7.90E-04	2.16E-05	1.20E-03	2.01E-03
Ag-110	0.0	2.00E-05	0.0	1.00E-04	0.0	0.0	1.00E-04
Sb-124	0.0	0.0	0.0	0.0	0.0	4.30E-04	4.30E-04
Te-129M	0.0	2.00E-05	0.0	1.00E-04	3.16E-06	0.0	1.03E-04
Te-129	0.0	1.00E-05	0.0	7.00E-05	4.72E-05	0.0	1.17E-04
Te-131M	0.0	3.00E-05	0.0	1.50E-04	1.91E-05	0.0	1.69E-04
Te-131	0.0	1.00E-05	0.0	3.00E-05	5.81E-06	0.0	3.58E-05
I -131	0.0	3.87E-03	2.40E-04	1.85E-02	4.64E-04	1.60E-03	2.05E-02
Te-132	0.0	9.00E-05	1.00E-05	4.20E-04	2.52E-05	0.0	4.45E-04
I -132	0.0	9.00E-05	6.00E-05	6.80E-04	6.36E-04	0.0	1.32E-03
I -133	0.0	1.54E-03	4.90E-04	9.16E-03	1.14E-03	0.0	1.03E-02
Cs-134	2.90E-04	1.02E-03	4.00E-05	6.05E-03	1.46E-04	1.10E-02	1.72E-02
I -135	0.0	2.40E-04	4.10E-04	2.92E-03	1.45E-03	0.0	4.37E-03
Cs-136	1.00E-05	9.00E-05	0.0	4.70E-04	1.72E-05	3.70E-04	8.57E-04
Cs-137	4.00E-04	1.35E-03	5.00E-05	8.11E-03	1.95E-04	1.60E-02	2.43E-02
Ba-137M	3.70E-04	1.26E-03	5.00E-05	7.58E-03	0.0	0.0	7.58E-03
Ba-140	0.0	1.29E-03	5.00E-05	6.03E-03	2.08E-04	9.10E-04	7.15E-03
La-140	0.0	1.74E-03	9.00E-05	8.21E-03	3.39E-04	0.0	8.55E-03
Ce-141	0.0	2.00E-05	0.0	8.00E-05	2.46E-06	2.30E-04	3.12E-04
Ce-143	0.0	6.00E-05	1.00E-05	3.10E-04	3.58E-05	0.0	3.46E-04
Pr-143	0.0	2.00E-05	0.0	1.10E-04	0.0	0.0	1.10E-04
Ce-144	0.0	5.10E-04	2.00E-05	2.38E-03	6.51E-05	3.90E-03	6.35E-03
Pr-144	0.0	5.10E-04	2.00E-05	2.38E-03	0.0	0.0	2.38E-03
Na-24	0.0	2.90E-04	1.00E-04	1.72E-03	4.83E-04	0.0	2.20E-03
P -32	0.0	0.0	0.0	0.0	0.0	1.80E-04	1.80E-04

TABLE 11.2-1
(Sheet 2 of 2)

ANNUAL AVERAGE LIQUID RELEASE SOURCE TERMS
(Curies/yr) [1]

Nuclide	Primary Bleed Waste	Liquid Waste System	Turbine Bldg Drains	Adjusted Total [3]	SG Drain Tank [2]	Detergent Waste	Total [3,4]
Cr-51	0.0	3.60E-04	1.00E-05	1.67E-03	5.26E-05	4.70E-03	6.42E-03
Mn-54	0.0	2.10E-04	1.00E-05	9.80E-04	2.65E-05	3.80E-03	4.81E-03
Fe-55	0.0	1.60E-04	1.00E-05	7.40E-04	1.99E-05	7.20E-03	7.96E-03
Fe-59	0.0	4.00E-05	0.0	1.70E-04	4.87E-06	2.20E-03	2.37E-03
Co-58	0.0	5.80E-04	2.00E-05	2.69E-03	7.73E-05	7.90E-03	1.07E-02
Co-60	0.0	7.00E-05	0.0	3.30E-04	8.98E-06	1.40E-02	1.43E-02
Ni-63	0.0	0.0	0.0	0.0	0.0	1.70E-03	1.70E-03
Zn-65	0.0	7.00E-05	0.0	3.10E-04	8.56E-06	0.0	3.19E-04
W -187	0.0	3.00E-05	1.00E-05	1.80E-04	2.99E-05	0.0	2.10E-04
Np-239	0.0	9.00E-05	1.00E-05	4.20E-04	3.14E-05	0.0	4.51E-04
Total	1.07E-03	4.21E-02	2.62E-03	2.06E-01	7.38E-03	8.98E-02	3.03E-01

Tritium Release is 370 Curies/yr

Notes:
[1] 0.0 appearing in this table indicates release is less than 1.0E-05 Curies/yr.
[2] One Steam Drain Tank Volume (50,000 gallons at secondary coolant concentration) is assumed to be released per year with no processing.
[3] Total is adjusted to include 0.16 Curies attributable to operational occurrences that result in unplanned releases. See PWRGAS output, Microfiche Attachment 1.
[4] Total includes sum of "Total Adjusted", "SG Drain Tank", and "Detergent Waste" columns.

TABLE 11.2-2

SOURCES, ESTIMATED VOLUMES AND ACTIVITIES
OF LIQUID WASTE MANAGEMENT SYSTEM INPUTS

Liquid Waste Source	Flow Rate (GPD)	Activity (PCA) [1]	LWMS Collection Tank	Collection Time (Days)	Processing Time (Days)
SHIM BLEED	193	1.0	Equipment Waste	90 [1]	0.76
EQUIPMENT DRAINS - Reactor Drain Tank - Equipment Drain Tank	[2]	1.0	[2]	[2]	[2]
CLEAN WASTE - Reactor Grade Lab Drains - Aerated Equipment Drains	70	0.2	Equipment Waste	30	0.76
DIRTY WASTE - Containment Sump - Plant Floor Drains - Fuel Pool Liner Leakage - Containment Cooling Condensate - Equipment and Area Non-detergent Decon	3200	0.021	Floor Drain Waste	6.7	0.76
SG BLOWDOWN	1.0 [3]	---	[3]	---	---
DETERGENT WASTE	[4]	[4]	Laundry and Hot Shower	[4]	[4]

Notes:

[1] Shim bleed collection time based on 40% of Holdup Tank capacity collection volume, and 1931 GPD average Holdup Tank input rate.
[2] Hydrogenated primary system equipment drain fluids (i.e., 250 GPD Reactor Drain Tank and Equipment Drain Tank inputs) normally recycled to RCS.
[3] Full blowdown flow processed by Blowdown System and recycled to condensate system.
[4] Detergent wastes collected and discharged without treatment consistent with NUREG-0017 method.

TABLE 11.2-4

ESTIMATED DOSES FROM RADIOACTIVE LIQUID EFFLUENTS
RELEASED FROM THE STATION

	Annual Dose (mrem/yr)	Appendix I Objective [2] (mrem/yr)
Maximum Total Body Dose From All Exposure Pathways [1]	2.20 (Adult)	3
Maximum Organ Dose From All Exposure Pathways	3.05 (Child-Bone)	10

Notes:

[1] Liquid effluent exposure pathways considered include fish ingestion, drinking water, and external exposure from shoreline sediments.
[2] 10 CFR 50, Appendix I numerical design objectives to meet the criterion "As Low As Reasonably Achievable".

Table 11.2-5

(Sheet 1 of 2)

Design Basis Average Annual Liquid Effluent Concentration

Nuclide	$C_r(i)$ (μCi/ml)	MPC(i) (μCi/ml)	FMPC(i)
Sr89	4.54E-11	3.00E-06	1.51E-05
Sr90	2.17E-12	3.00E-07	7.23E-06
Sr91	1.83E-12	7.00E-05	2.61E-08
Y91	9.18E-11	3.00E-05	3.06E-06
Y93	1.30E-12	3.00E-05	4.33E-08
Zr95	1.99E-11	2.00E-05	9.93E-07
Nb95	4.26E-11	1.00E-04	4.26E-07
Mo99	8.07E-10	4.00E-05	2.02E-05
Tc99m	6.07E-10	3.00E-03	2.02E-07
Ru103	5.17E-11	8.00E-05	6.46E-07
Rh103m	4.70E-11	1.00E-02	4.70E-09
Ru106	7.33E-10	1.00E-05	7.33E-05
Ag110m	2.25E-11	3.00E-05	7.50E-07
Sb124	4.81E-12	3.00E-06	1.60E-06
Te129m	3.66E-11	2.00E-05	1.83E-06
Te129	1.31E-12	8.00E-04	1.64E-09
Te131m	3.80E-11	4.00E-05	9.51E-07
Te131	7.14E-13	3.00E-06	2.38E-07
I131	1.28E-08	3.00E-07	4.27E-02
Te132	5.98E-10	2.00E-05	2.99E-05
I132	5.66E-11	8.00E-06	7.07E-06
I133	3.16E-09	1.00E-06	3.16E-03
Cs134	4.39E-09	9.00E-06	4.88E-04
I135	4.33E-10	4.00E-06	1.08E-04
Cs136	4.96E-10	6.00E-05	8.27E-06

Table 11.2-6

Normal Operating Average Annual Liquid Effluent Concentration

(Delete this table)

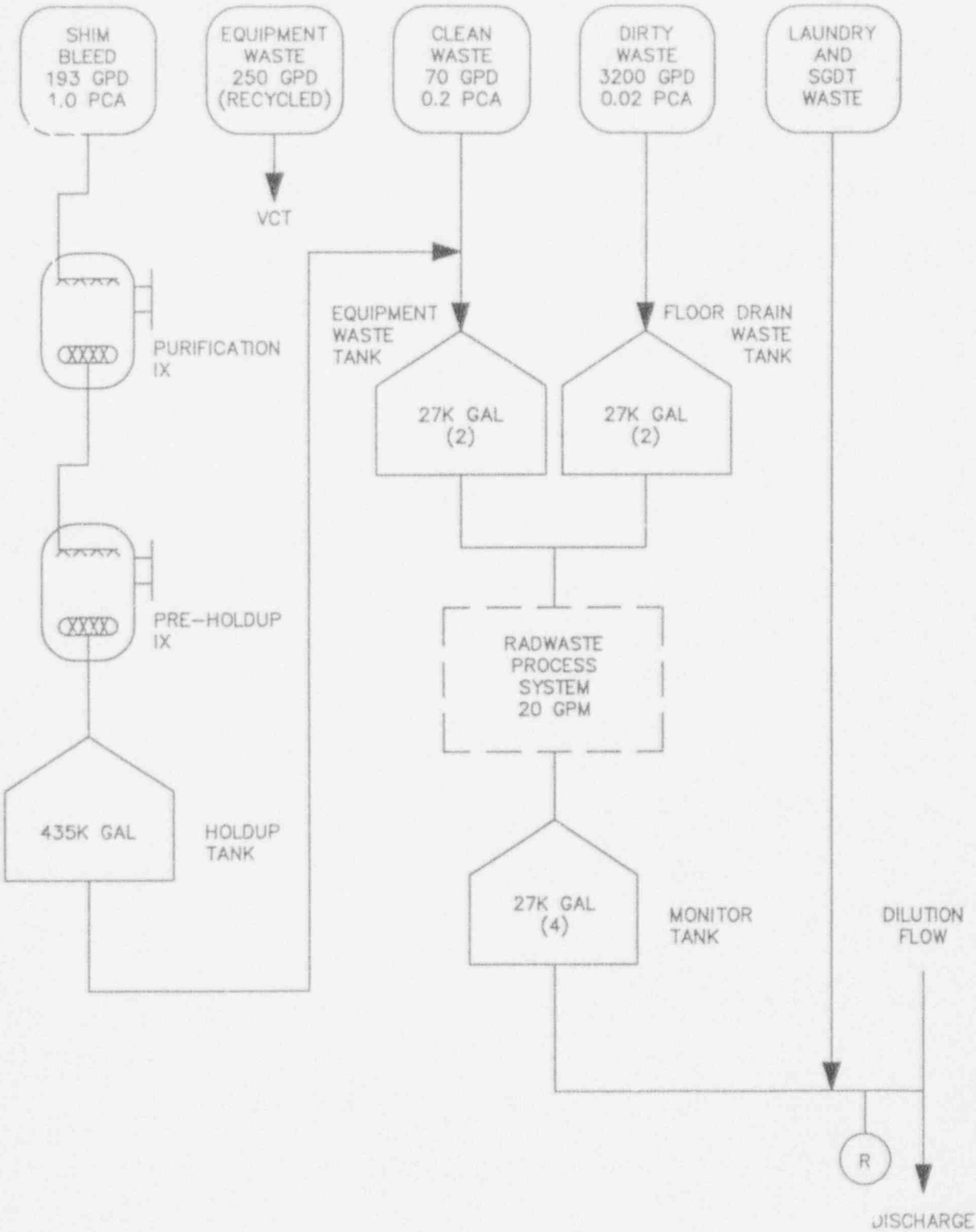

SHIM
BLEED
193 GPD
1.0 PCA
EQUIPMENT
WASTE
250 GPD
(RECYCLED)
CLEAN
WASTE
70 GPD
0.2 PCA
DIRTY
WASTE
3200 GPD
0.02 PCA
LAUNDRY
AND
SGDT
WASTE
VCT
PURIFICATION
IX
EQUIPMENT
WASTE
TANK
FLOOR DRAIN
WASTE
TANK
27K GAL
(2)
27K GAL
(2)
PRE-HOLDUP
IX
RADWASTE
PROCESS
SYSTEM
20 GPM
435K GAL
HOLDUP
TANK
27K GAL
(4)
MONITOR
TANK
DILUTION
FLOW
R
DISCHARGE

Figure 11.2-2

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

11.3.1 DESIGN BASES

11.3.1.1 Criteria and Evaluation

The GWMS is designed in accordance with the acceptance criteria defined in the Standard Review Plan, Section 11.3. The design criteria are the following:

A. Effluents normally released to unrestricted areas must meet the limiting requirements of 10 CFR 20 and meet the ALARA objectives of 10 CFR 50, Appendix I.

   The GWMS continuously discharges effluent. Table 11.3-4 provides an estimate of the annual airborne effluent releases (Ci/yr) based on results from PWR-GALE. Assumptions used to calculate the annual release rate are discussed in Section 11.3.6. This estimated annual release rate is used to calculate the estimated annual dose to the maximum individual. These results are listed in Table 11.3-5. This analysis assures that effluents during normal operation and anticipated operational occurrences meet 10 CFR 50, Appendix I objectives.

   The GWMS is designed to ensure that normal releases to unrestricted areas are within 10 CFR 20, Appendix B maximum permissible concentrations based on the design basis source term. Section 11.3.8 provides a detailed discussion regarding the methodology used to calculate the concentration of the effluent at the Exclusion Area Boundary. The results of this analysis assure that the concentration of the effluent are well within 10 CFR 20, Appendix B, Table II, Column 1 maximum permissible concentrations.

   a

B. The system must contribute to meeting the performance design objectives in that it must never interfere with normal station operation including anticipated operational occurrences.

The GWMS is a non-nuclear safety related system. It has no accident mitigation functions. The GWMS is designed in accordance with requirements in ANSI/ANS 55.4-1979, Regulatory Guide 1.143 and 1.140. This includes the following features:

1. The GWMS is designed to preclude a buildup of an explosive mixture of hydrogen and oxygen which could impact the operation of the plant. J

2. The GWMS is designed with sufficient storage capacity and redundancy to accommodate an increase in demand during normal operation of the plant.

C. Releases of radioactive materials to the environment must be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61 and 64). I

The GWMS is provided with radiation monitors which monitor the discharge from the charcoal adsorber beds upstream of the filter packages in the Radwaste Ventilation System. The GWMS discharge is automatically isolated if the discharge limit will be exceeded. Section 11.5, Radiation Monitoring System, provides a detailed discussion regarding the radiation monitoring for the GWMS. J

D. Accidental releases of radioactive materials from a single component of the GWMS must not result in offsite doses which exceed the guidelines of 10 CFR 20. I

Section 11.3.7 provides a discussion of the analysis of a single component failure of the GWMS. The methodology used in this analysis is in accordance with Branch Technical ESTB-11-5 for the design basis source term. The results of this analysis confirm that the dose consequence of a single failure of component in the GWMS is within the guidelines of 10 CFR 20. J

a component

E. The system must also contribute to meeting the occupational exposure design objective by keeping operation and maintenance exposure ALARA. I

The GWMS is designed in accordance with guidance provided in Regulatory Guide 8.8, ANSI/ANS-55.4-1979, and Regulatory Guide 1.143 and 1.140. This ensures that the GWMS will meet ALARA objectives. J

11.3.7 GASEOUS WASTE MANAGEMENT SYSTEM LEAK OR FAILURE

11.3.7.1 Identification of Causes and Accident Description

The Gaseous Waste Management System (GWMS), as discussed in Section 11.3 is designed to collect, monitor, and store radioactive waste gases which originate in the reactor coolant system and require processing by holdup for decay prior to release. The GWMS utilizes ambient temperature charcoal adsorption beds to provide sufficient decay of noble gases.

The accident is described as an unexpected and uncontrolled release of radioactive Xenon and Krypton gases from the GWMS resulting from an inadvertent bypass of the main decay portion of the charcoal adsorber beds. It is assumed to take as long as 2 hours to isolate or terminate the release.

11.3.7.2 Analysis of Effects and Consequences

A. Bases

1. The assumptions and methodology are consistent with guidance provided in Branch Technical Position ESTB 11-5. J

2. An effective holdup time of 30 minutes is assumed for the bypass flow to account for transport time of the gases through the GWMS components via the release point to the nearest exclusion area boundary.

3. In accordance with ESTB 11-5, the Waste Gas System maximum design capacity source term (at sustained power) is assumed to seven times the source term considered for normal operation, including anticipated operational occurrences. PWR-GALE is run for a 30 minute decay case and the results are multiplied by seven to calculate the maximum design capacity source term.

4. The total source term is equal to the maximum design basis source term plus the normal operations source term shown in Table 11.3-4.

5. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. Therefore, only the whole body dose is calculated in this analysis.

Replace with Attachment Insert A

Replace with Attached Insert A

6. In the absence of site specific meteorological data and exclusion area boundary information, an atmospheric dispersion factor (X/Q) of 1.00×10^{-3} is assumed for the exclusion area boundary as described in Chapter 15, Appendix A. J K

B. Methodology

To calculate the dose consequences for a Waste Gas System failure methodology consistent with Branch Technical ESTB 11-5 is used.

$$D = \Sigma K(i) * Q(i) * X/Q * 7.25$$

J

Where:

D = Dose (mrem)

K(i) = the total-body dose factor given in Table B-1 of Regulatory Guide 1.109 for the i^{th} isotope (mrem-m^3/pCi/yr)

Q(i) = the noble gas nuclide release rate for the i^{th} isotope (Ci/yr)

X/Q = atmospheric dispersion factor at the exclusion area boundary

X/Q = 1.00×10^{-3} s/m^3 K

7.25 = conversion factor for 2 hour release (p/Ci-yr^2/Ci-event-sec)

C. Results and Conclusions

The resulting Exclusion Area Boundary noble gas dose to the whole body is 14.7 mrem. This meets the guidelines specified in the Standard Review Plan Section 11.3. J

11.3.8 CONCENTRATION OF NORMAL EFFLUENTS

The Gaseous Waste Management System (GWMS) processes gaseous waste through a charcoal delay system which holds up noble gases and allows them to decay prior to release. The concentration at the exclusion area boundary during normal operation, including anticipated operating occurrences, was analyzed to verify it is less than 10 CFR 20, Appendix B, Table II, Column 1.

Replace with Attached Insert B

11.3.8.1 Analysis of Effects and Consequences

A. Bases

The bases for the estimated concentration of effluent are as follows:

1. This system continuously discharges at a uniform rate.

2. The concentration of the effluent is based on the design basis source term. J

3. The total gaseous effluent calculated using NUREG-0017 methodology shown in Table 11.3-4 is multiplied by seven to yield a conservative approximation of the design basis source term. This methodology is consistent with the suggested methodology in Branch Technical Position ESTB 11-5 for a Waste Gas System Leak of Failure consequence analysis.

4. In the absence of site specific meteorological data and site Exclusion Area Boundary (EAB) information, an atmospheric dispersion factor of 1.00×10^{-3} s/m^3 was assumed for the EAB (500 meters) based on Chapter 15, Appendix A. K

B. Methodology

The methodology used to calculate the concentration of the effluent at the Exclusion Area Boundary is as follows: J

$$C(i) = CF * 7R(i) * X/Q_{EAB}$$

Where:

$C(i)$ = Concentration of the i^{th} isotope at the EAB ($\mu Ci/ml$) K

CF = Conversion Factor
= 3.17×10^{-11} ($s\text{-}\mu Ci\text{-}m^3/yr\text{-}Ci\text{-}ml$)

$R(i)$ = Release Rate of i^{th} isotope (Ci/yr) J

X/Q_{EAB} = Atmospheric dispersion factor at EAB (s/m^3)
= 1.00×10^{-3} (s/m^3) K

Replace with Attached Insert B

C. Results and Conclusions

The concentration of the effluent at the Exclusion Area Boundary is shown in Table 11.3-6. The concentration at the Exclusion Area Boundary is well within 10 CFR 20 guidelines. Although there are periodic purges of containment during normal operation, these purges will be controlled by procedures developed by the Owner Operator to ensure compliance with 10 CFR 20 limits. J

1/A

Replace with Attached Table 11.3-4

TABLE 11.3-4

ESTIMATED ANNUAL AIRBORNE EFFLUENT RELEASES (1)
(Curies/yr)

Nuclide	Waste Gas System	Fuel Handling	Reactor Bldg Purge	Nuclear Annex Vent (2)	Turbine Bldg Vent	Air Ejector Exhaust (3)	Total
I-131	0.0	3.4E-04	1.3E-03	1.6E-02	0.0	0.0	1.8E-02
I-133	0.0	1.0E-03	2.7E-03	5.0E-02	1.0E-04	0.0	5.4E-02
Kr-85m	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Kr-85	7.5E+02	---	1.5E+01	0.0	0.0	0.0	7.7E+02
Kr-87	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Kr-88	0.0	---	1.0E+00	5.0E+00	0.0	2.0E+00	8.0E+00
Xe-131m	1.4E+02	---	1.6E+01	0.0	0.0	0.0	1.6E+02
Xe-133m	0.0	---	1.0E+00	0.0	0.0	0.0	1.0E+00
Xe-133	1.1E+01	---	5.1E+01	6.0E+00	0.0	3.0E+00	7.1E+01
Xe-135m	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Xe-135	0.0	---	8.0E+00	1.1E+01	0.0	5.0E+00	2.4E+01
Xe-137	0.0	---	0.0	0.0	0.0	0.0	0.0
Xe-138	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Cr-51	1.4E-06	1.8E-06	2.7E-05	3.2E-06	---	---	3.3E-05
Mn-54	2.1E-07	3.0E-06	1.6E-05	7.8E-07	---	---	2.0E-05
Co-57	0.0	0.0	2.4E-06	0.0	---	---	2.4E-06
Co-58	8.7E-07	2.1E-04	7.3E-05	1.9E-05	---	---	3.0E-04
Co-60	1.4E-06	8.3E-05	7.6E-06	5.1E-06	---	---	9.7E-05
Fe-59	1.8E-07	0.0	7.9E-06	5.0E-07	---	---	8.6E-06
Sr-89	4.4E-06	2.1E-05	3.8E-05	7.5E-06	---	---	7.1E-05
Sr-90	1.7E-06	8.0E-06	1.5E-05	2.9E-06	---	---	2.8E-05
Zr-95	4.8E-07	3.6E-08	0.0	1.0E-05	---	---	1.1E-05
Nb-95	3.7E-07	2.4E-05	5.3E-06	3.0E-07	---	---	3.0E-05
Ru-103	3.2E-07	3.8E-07	4.7E-06	2.3E-07	---	---	5.6E-06
Ru-106	2.7E-07	6.9E-07	0.0	6.0E-08	---	---	1.0E-06
Sb-125	0.0	5.7E-07	0.0	3.9E-08	---	---	6.1E-07
Cs-134	3.3E-06	1.7E-05	7.3E-06	5.4E-06	---	---	3.3E-05
Cs-136	5.3E-07	0.0	9.4E-06	4.8E-07	---	---	1.0E-05
Cs-137	7.7E-06	2.7E-05	1.6E-05	7.2E-06	---	---	5.8E-05
Ba-140	2.3E-06	0.0	0.0	4.0E-06	---	---	6.3E-06
Ce-141	2.2E-07	4.4E-09	3.8E-06	2.6E-07	---	---	4.3E-06

Total H-3 Released Via Airborne Pathway = 1200 Curies/yr
C-14 Released Via Airborne Pathway = 7.3 Curies/yr
Ar-41 Released Via Containment Vent = 34 Curies/yr

NOTES:
(1) 0.0 appearing in this table indicates release is less than 1.0 Ci/yr for noble gases, and 0.0001 Ci/yr for iodines/particulates.
(2) Includes Annex, Subsphere, and Radwaste Building release contributions.
(3) Includes Blowdown System Flash Tank vent release contributions.

Replace with Attached Table 11.3-5

TABLE 11.3-5 I

ESTIMATED ANNUAL DOSES FROM GASEOUS EFFLUENT FROM A SINGLE UNIT J

	Dose	Appendix I Objective
Maximum Beta Air Dose (mrad/yr)	4.6	20
Maximum Gamma Air Dose (mrad/yr)	1.4	10
MAXIMUM INDIVIDUAL ANNUAL DOSE (mrem/yr)		
Skin Dose (1)	4.2	15
Total Body Dose (1)	0.9	5
Maximum Organ Dose (2)	(Infant-Thyroid)	15

I

NOTES:

(1) Exposure from noble gas plume immersion pathway.

(2) Maximum exposure from iodine, particulate, tritium and C-14 via the terrestrial exposure pathways (i.e., ground plane, vegetable, meat and milk) and the inhalation exposure pathway.

Replace with Attached Table 11.3-6

TABLE 11.3-6

AVERAGE ANNUAL CONCENTRATION OF GASEOUS EFFLUENTS AT THE EXCLUSION AREA BOUNDARY (a)

J

Nuclide	C(i) (μCi/ml)	MPC(i) (μCi/ml)	FMPC(i)
I-131	3.99E-15	1.00E-10	3.99E-05
I-133	1.20E-14	4.00E-10	2.99E-05
Kr-85M	8.87E-13	1.00E-07	8.87E-06
Kr-85	1.71E-10	3.00E-07	5.69E-04
Kr-87	8.87E-13	2.00E-08	4.44E-05
Kr-88	1.77E-12	2.00E-08	8.87E-05
Xe-131M	3.55E-11	4.00E-07	8.87E-05
Xe-133M	2.22E-13	3.00E-07	7.39E-07
Xe-133	1.57E-11	3.00E-07	5.25E-05
Xe-135M	8.87E-13	3.00E-08	2.96E-05
Xe-135	5.32E-12	1.00E-07	5.32E-05
Xe-138	8.87E-13	3.00E-08	2.96E-05
Cr-51	7.32E-18	8.00E-08	9.15E-11
Mn-54	4.44E-18	1.00E-09	4.44E-09
Co-57	5.32E-19	6.00E-09	8.87E-11
Co-58	6.65E-17	2.00E-09	3.33E-08
Co-60	2.13E-17	3.00E-10	7.10E-08
Fe-59	1.91E-18	2.00E-09	9.54E-10
Sr-89	1.57E-17	3.00E-10	5.25E-08
Sr-90	6.21E-18	3.00E-11	2.07E-07
Zr-95	2.44E-18	1.00E-09	2.44E-09
Nb-95	6.65E-18	3.00E-09	2.22E-09
Ru-103	1.24E-18	3.00E-09	4.14E-10
Ru-106	2.22E-19	2.00E-10	1.11E-09
Sb-125	1.35E-19	9.00E-10	1.50E-10
Cs-134	7.32E-18	4.00E-10	1.83E-08
Cs-136	2.22E-18	6.00E-09	3.70E-10
Cs-137	1.29E-17	5.00E-10	2.57E-08
Ba-140	1.40E-18	1.00E-09	1.40E-09
Ce-141	9.54E-19	5.00E-09	1.91E-10
H-3	2.66E-10	4.00E-05	1.33E-03
C-14	1.62E-12	1.00E-07	1.62E-05
Ar-41	7.54E-12	4.00E-08	1.89E-04
		Total:	2.57E-03 MPC

K

J

(a) Based on the design basis source term.

INSERT A:

11.3.7.2 Analysis of Effects and Consequences

A. Bases

The bases for the estimated maximum offsite concentration of the gaseous effluent resulting from a leak or failure of the GWMS are as follows:

1. The design basis airborne effluent source term is based on 1% failed fuel rate in accordance with the Standard Review Plan Branch Technical Position (BTP) ESTB 11-5. The BTP ESTB 11-5 method adds the accident induced charcoal unit bypass leakage to the source term for normal operation; both accident source contributions are calculated based on a 1% failed fuel rate assumption.

2. In the absence of site specific meteorological data and site Exclusion Area Boundary (EAB) information, the short-term 2-hour accident atmospheric dispersion factor, corresponding to a distance of approximately 0.5 miles from the station vent, is assumed to be 1.0×10^{-3} s/m^3. This is consistent with the dilution factors provided in Section 2.3.

3. The sum of total estimated annual airborne effluent releases and the expected airborne effluent releases associated with the 30 minute decay case are calculated by PWR-GALE and are multiplied by an isotope specific multiplication factor. This multiplication factor is calculated by the division of the 1% failed fuel RCS equilibrium concentration, calculated using the Combustion Engineering DAMSAM computer code and presented in Table 11.1.1-9, by the RCS equilibrium concentration calculated using PWR-GALE presented in Table 11.1.1-2, for each isotope.

4. For isotopes with a 1% failed fuel rate calculated concentration which is less than PWR-GALE results, the PWR-GALE concentration is used for conservatism. It is assumed that differences in the methodology used to calculate the reactor coolant concentrations are responsible for any differences observed in isotopic concentrations.

5. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. Therefore, only the whole body dose is calculated in this analysis.

B. Methodology

To calculate the release of noble gases from the GWMS, the source term is based on the output from the computer code DAMSAM computer code. This code is used to calculate the reactor coolant equilibrium concentration with continuous degassing based on 1% failed fuel fraction in accordance with Standard Review Plan Section 11.3. The resulting reactor coolant equilibrium concentration is divided by the reactor coolant concentration determined by PWR-GALE, using NUREG-0017, Revision 1 methodology, to yield a multiplication factor for each isotope. The total release of gaseous effluent for the 30 minute decay case is calculated using PWR-GALE with BTP ESTB 1 5 alterations. The 30 minute decay case releases are added to the normal operation source term and the sum for each radionuclide is multiplied by the multiplication factor, the 2-hour accident atmospheric dispersion factor, the total body dose factor, and a conversion factor to calculate whole body dose.

The methodology used to calculate the dose consequences for a GWMS failure, which is consistent with BTP ESTB 11-5, is as follows:

$$D=\sum K(i)\times Q(i)\times \frac{X}{Q}\times 7.25$$

Where: D = whole body dose (mrem)

K(i) = the total-body dose factor given in Table B-1 of Regulatory Guide 1.109 for the ith isotope (mrem-m^3/pCi/yr)

Q(i) = the noble gas nuclide accident release rate for the ith isotope (Ci/yr for 2 hours)

$$Q(i)=[R(i)_{Norm}+R(i)_{30}]\times MF(i)$$

$R(i)_{norm}$ = annual estimated airborne release rate for normal operation (Ci/yr) (Table 11.3-4)

$R(i)_{30}$ = annual estimate airborne release rate for 30 minute decay case (Ci/yr)

MF = Multiplication Factor

$$MF=\frac{RCS(i)_{DAMSAM}}{RCS(i)_{GALE}}$$

X/Q = short-term 2-hour accident atmospheric dispersion factor at EAB (sec/m^3)
= 1.0×10^{-3} (Section 2.3)

7.25 = conversion factor for 2 hour release (pCi-yr^2/Ci-event-sec)

C. Results and Conclusions

The calculated whole body dose at the exclusion area boundary is 41.3 mrem which is within the 500 mrem acceptance criterion specified in Standard Review Plan Section 11.3.

INSERT B:

11.3.8.1 Analysis of Effects and Consequences

A. Bases

The bases for the estimated concentration of effluent are as follows:

1. The GWMS continuously discharges at a uniform rate at the design basis source term.

2. The design basis airborne effluent source term is based on 1% failed fuel rate in accordance with the Standard Review Plan Section 11.3. It is assumed that the Reactor Coolant System (RCS) is continuously degassed by the CVCS during normal operating conditions. The reactor coolant equilibrium concentration is calculated using the Combustion Engineering DAMSAM computer code and is presented in Table 9.1.1-9.

3. In the absence of site specific meteorological data and site Exclusion Area Boundary (EAB) information, the long-term annual average atmospheric dispersion factor, corresponding to a distance of approximately 0.5 miles from the station vent, is assumed to be 7.2×10^{-5} s/m^3. This is consistent with the dilution factors assumed in Section 11.3.6.3.

4. The total estimated annual airborne effluent releases are multiplied by an isotope specific multiplication factor. This multiplication factor is calculated by the division of the 1% failed fuel RCS equilibrium concentration, calculated by the Combustion Engineering DAMSAM computer code, by the RCS equilibrium concentration, calculated using PWR-GALE, presented in Table 11.1.1-2, for each isotope.

   For isotopes with a 1% failed fuel rate calculated concentration which is less than PWR-GALE results, the PWR-GALE concentration is used for conservatism. It is assumed that differences in the methodology used to calculate the reactor coolant concentrations are responsible for any differences observed in isotopic concentrations.

5. Since DAMSAM does not calculate the concentration of tritium, the maximum calculated concentration of 1.00 μCi/gm is assumed for the 1% failed fuel source term for conservatism.

6. Since DAMSAM does not calculate the concentration of corrosion products, the PWR-GALE numbers are used. The concentration of these radionuclides should not be affected by

the fraction of fuel defects.

B. **Methodology**

To calculate the concentration at the exclusion area boundary, the source term is based on the output from the computer code DAMSAM computer code. This code is used to calculate the reactor coolant equilibrium concentration with continuous degassing based on 1% failed fuel fraction in accordance with Standard Review Plan Section 11.3. The resulting reactor coolant equilibrium concentration is divided by the reactor coolant concentration determined by PWR-GALE, using NUREG-0017, Revision 1 methodology, to yield a multiplication factor for each isotope. The total annual release rate of gaseous effluent is multiplied by the multiplication factor and the average atmospheric dispersion factor to calculate the annual average concentration of the gaseous effluent at the exclusion area boundary. This concentration is then compared to the Maximum Permissible Concentration (MPC(i)) for each isotope to verify compliance with 10 CFR 20, Appendix B, Table II, Column 1 limits.

The methodology used to calculate the concentration of the effluent, averaged over a period of one year at the EAB, is as follows:

$$C_{EAB}(i) = R(i) \times MF(i) \times \frac{X}{Q} \times CF$$

Where: $C(i)_{EAB}$ = Concentration of ith isotope at the EAB (μCi/ml)

X/Q = Average atmospheric dispersion factor at EAB(sec/m^3)
= 7.2×10^{-5} (Section 11.3.6.3)

R(i) = Release Rate (Ci/yr) (Table 11.3-4)

MF = Multiplication Factor

$$MF = \frac{RCS(i)_{DAMSAM}}{RCS(i)_{GALE}}$$

CF = Conversion Factor

$$CF = 3.17E\text{-}8 \frac{\mu Ci\text{-}yr\text{-}m^3}{Ci\text{-}sec\text{-}ml}$$

The fraction of the MPC for each isotope is calculated as follows:

$$FMPC(i) = \frac{C_{EAB}(i)}{MPC(i)}$$

Where: FMPC(i) = Fraction of MPC for the ith isotope

MPC(i) = Maximum Permissible Concentration of the ith isotope (μCi/ml) (10 CFR 20, Appendix B, Table 2, Column 1)

C. **Results and Conclusions**

The concentration of the gaseous effluents at the EAB is shown in Table 11.3-5. The resultant concentration at the EAB is within the Maximum Permissible Concentration specified in 10 CFR 20, Appendix B, Table II, Column 1 guidelines.

TABLE 11.3-4

ESTIMATED ANNUAL AIRBORNE EFFLUENT RELEASES (1)
(Curies/yr)

Nuclide	Waste Gas System	Fuel Handling	Reactor Bldg Purge	Aux Bldg Vent (2)	Turbine Bldg Vent	Air Eject Exhaust (3)	Total
I -131	0.0	3.5E-04	1.4E-03	1.7E-02	1.0E-04	0.0	1.9E-02
I -133	0.0	1.0E-03	2.6E-03	4.9E-02	2.5E-04	1.0E-04	5.3E-02
Kr-85 M	0.0	---	0.0	2.0E+00	0.0	1.0E+00	3.0E+00
Kr-85	7.8E+02	---	1.8E+01	0.0	0.0	0.0	8.0E+02
Kr-87	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Kr-88	0.0	---	1.0E+00	5.0E+00	0.0	2.0E+00	8.0E+00
Xe-131M	1.4E+02	---	1.9E+01	0.0	0.0	0.0	1.6E+02
Xe-133M	0.0	---	1.0E+00	0.0	0.0	0.0	1.0E+00
Xe-133	1.2E+01	---	6.0E+01	6.0E+00	0.0	3.0E+00	8.1E+01
Xe-135M	0.0	---	0.0	3.0E+00	0.0	1.0E+00	4.0E+00
Xe-135	0.0	---	8.0E+00	1.0E+01	0.0	5.0E+00	2.3E+01
Xe-137	0.0	---	0.0	0.0	0.0	0.0	0.0
Xe-138	0.0	---	0.0	2.0E+00	0.0	1.0E+00	3.0E+00
Cr-51	1.4E-06	1.8E-06	2.7E-05	3.2E-06	---	---	3.3E-05
Mn-54	2.1E-07	3.0E-06	1.6E-05	7.8E-07	---	---	2.0E-05
Co-57	0.0E+00	0.0E+00	2.4E-06	0.0E+00	---	---	2.4E-06
Co-58	8.7E-07	2.1E-04	7.3E-05	1.9E-05	---	---	3.0E-04
Co-60	1.4E-06	8.2E-05	7.6E-06	5.1E-06	---	---	9.6E-05
Fe-59	1.8E-07	0.0E+00	7.9E-06	5.0E-07	---	---	8.6E-06
Sr-89	4.4E-06	2.1E-05	3.8E-05	7.5E-06	---	---	7.1E-05
Sr-90	1.7E-06	8.0E-06	1.5E-05	2.9E-06	---	---	2.8E-05
Zr-95	4.8E-07	3.6E-08	0.0E+00	1.0E-05	---	---	1.1E-05
Nb-95	3.7E-07	2.4E-05	5.3E-06	3.0E-07	---	---	3.0E-05
Ru-103	3.2E-07	3.8E-07	4.7E-06	2.3E-07	---	---	5.6E-06
Ru-106	2.7E-07	6.9E-07	0.0E+00	6.0E-08	---	---	1.0E-06
Sb-125	0.0E+00	5.7E-07	0.0E+00	3.9E-08	---	---	6.1E-07
Cs-134	3.3E-06	1.7E-05	7.3E-06	5.4E-06	---	---	3.3E-05
Cs-136	5.3E-07	0.0E+00	9.4E-06	4.8E-07	---	---	1.0E-05
Cs-137	7.7E-06	2.7E-05	1.6E-05	7.2E-06	---	---	5.8E-05
Ba-140	2.3E-06	0.0E+00	0.0E+00	4.0E-06	---	---	6.3E-06
Ce-141	2.2E-07	4.4E-09	3.8E-06	2.6E-07	---	---	4.3E-06

Total H-3 Released Via Airborne Pathway = 1200 Curies/yr
Ar-41 Released Via Containment Vent = 34 Curies/yr

Notes:

(1) 0.0 appearing in this table indicates release is less than 1.0 Ci/yr for noble gases, and 0.0001 curies/yr for iodines/particulates.
(2) Includes Annex, Subsphere, and Radwaste Building release contributions.
(3) Includes Blowdown System Flash Tank vent release contributions.

TABLE 11.3-5

ESTIMATED ANNUAL DOSES FROM GASEOUS EFFLUENT
FROM A SINGLE UNIT

	Dose	Appendix I Objective
Maximum Beta Air Dose (mrad/yr)	4.8	20
Maximum Gamma Air Dose (mrad/yr)	1.4	10
MAXIMUM INDIVIDUAL ANNUAL DOSE (mrem/yr)		
Skin Dose (1)	4.3	15
Total Body Dose (1)	0.9	5
Maximum Organ Dose (2)	13.9 (Infant-Thyroid)	15

Notes:

(1) Exposure from noble gas plume immersion pathway.
(2) Maximum exposure from tritium, iodine and particulate airborne effluent releases via the terrestrial exposure pathways (i.e., ground plane, vegetable, meat, and milk) and the inhalation exposure pathway.

Table 11.3-6

(Sheet 1 of 2)

Design Basis Annual Average Gaseous Effluent Concentration at the Exclusion Area Boundary

Nuclide	$C_r(i)$ (μCi/ml)	MPC(i) (μCi/ml)	FMPC(i)
Sr89	3.82E-15	3.00E-10	1.27E-05
Sr90	6.13E-16	3.00E-11	2.04E-05
Zr95	3.33E-17	1.00E-09	3.33E-08
Nb95	1.25E-16	3.00E-09	4.18E-08
Ru103	1.28E-17	3.00E-09	4.26E-09
Ru106	2.28E-18	2.00E-10	1.14E-08
I131	2.42E-12	1.00E-10	2.42E-02
I133	3.32E-12	4.00E-10	8.29E-03
Cs134	1.72E-15	4.00E-10	4.30E-06
Cs136	1.18E-05	6.00E-09	1.97E-07
Cs137	4.04E-15	5.00E-10	8.08E-06
Ba140	4.51E-18	1.00E-09	4.51E-09
Ce141	9.93E-18	5.00E-09	1.99E-09
Cr51	7.53E-17	8.00E-08	9.41E-10
Mn54	4.56E-17	1.00E-09	4.56E-08
Fe59	1.96E-17	2.00E-09	9.81E-09
Co57	5.48E-18	6.00E-09	9.13E-10
Co58	6.85E-16	2.00E-09	3.42E-07
Co60	2.19E-16	3.00E-10	7.30E-07
Kr85m	5.37E-11	1.00E-07	5.37E-04
Kr85	5.58E-09	3.00E-07	1.86E-02
Kr87	6.14E-11	2.00E-08	3.07E-03
Kr88	1.82E-10	2.00E-08	9.08E-03
Xe131m	1.94E-09	4.00E-07	4.85E-03
Xe133m	9.57E-12	3.00E-07	3.19E-05
Xe133	1.99E-08	3.00E-07	6.65E-02
Xe135m	5.21E-11	3.00E-08	1.74E-03
Xe135	4.48E-10	1.00E-07	4.48E-03

Table 11.3-6

(Sheet 2 of 2)

Design Basis Annual Average Gaseous Effluent Concentration at the Exclusion Area Boundary

Nuclide	$C_r(i)$ (μCi/ml)	MPC(i) (μCi/ml)	FMPC(i)
Xe137	0.00E+00	0.00E+00	0.00E+00
Xe138	3.60E-11	3.00E-08	1.20E-03
H3	2.74E-09	4.00E-05	6.85E-05
C14	1.67E-11	1.00E-07	1.67E-04
Ar41	7.76E-11	4.00E-08	1.94E-03
		Total:	1.43E-01

K. J. Mortensen
4/20/93

DSER Open Item 11.1-2

SRP Sections 11.2 and 11.3 state that the radwaste system should have the capability to process waste based on 1-percent failed fuel.

Proposed Open Item 11.1-2 Resolution

Analysis to evaluate the concentration of effluent from the LWMS and the GWMS during normal operating conditions for 1-percent failed fuel rate condition has been performed. Proposed revisions to CESSAR-DC Sections 11.1, 11.2 and 11.3 are included as an attachment to the DSER Open Item 11.1-1 response.

K J Montan
4/20/93

DSER Open Item 11.2-1

The liquid waste management system must consider operator error. Also, the DFs for the shim bleed used for concentration determination may not be conservative.

Proposed Open Item 11.2-1 Resolution

The computer code PWR-GALE based on NUREG-0017, Revision 1 methodology accounts for operator error and anticipated operational occurances by applying an additional 0.16 Ci/yr to the estimated annual release source term for the LWMS.

The decontamination factors (DFs) used for shim bleed are consistent with those provided in Section 11.2 based on NUREG-0017, Revision 1 guidance. The Reactor Coolant System (RCS) is purified during letdown to the Chemical Volume and Control System (CVCS) by purification filters and ion exchangers. The CVCS is a kidney cleanup system for the RCS. Only a small fraction of the total volume of the RCS is processed during letdown; therefore, crediting processing through the CVCS purification filters and ion exchangers for shim bleed flow stream is considered a reasonable assumption. The full flow of the shim bleed is processed by the CVCS prior to the discharge of 10% of the total shim bleed flow rate (1931 gpd average) to the environment. The rest is recycled in the CVCS.

The effluent analysis has been completed for the LWMS using a source term based on 1-percent failed fuel considering all effluent pathways. The proposed revision to CESSAR-DC Section 11.2 is included as an attachment to the DSER Open Item 11.1-1 response.

K. J. Menton
4/20/93

DSER Open Item 11.2-2

The applicant must revise CESSAR Section 11.2 tables and demonstrate that normal releases to unrestricted areas will be within 10 CFR 20, Appendix B acceptable concentrations.

Proposed Open Item 11.2-2 Resolution

CESSAR-DC Section 11.2 has been updated to include the results and methodology used to calculate the concentration of the effluents from the LWMS at the plant discharge point to demonstrate compliance with 10 CFR 20, Appendix B acceptable concentrations assuming the specified 1% failed fuel rate. The proposed revision to CESSAR-DC Section 11.2 is included as an attachment to the DSER Open Item 11.1-1 response.

K. J. Montage
4/20/93

DSER Open Item 11.3-4

The four values identified by the DSER Section 11.3 must be corrected.

Proposed Open Item 11.3-4 Resolution

These values have been corrected with the completion of the revised airborne effluent concentration analysis associated with 1-percent failed fuel operation. CESSAR-DC Section 11.3 revisions, which include the results and methodology used in this analysis, are included as an attachment to the DSER Open Item 11.1-1 response.

K. J. Montague
4/20/93

DSER Open Item 11.3-5

The source term used in the charcoal delay bed system design should correspond to 1-percent failed fuel.

Proposed Open Item 11.3-5 Resolution

The effluent analysis for the GWMS has been revised to calculate the site-boundary concentration of effluents from the GWMS associated with 1-percent failed fuel operation. Also, the GWMS failure analysis provided in Section 11.3.7.2 has been revised as necessary to account for the maximum design basis airborne effluent source term (1% failed fuel) to ensure that the whole body dose is less than 500 mrem. Section 11.3 has been updated to include the results and methodology used in these analyses. The proposed revision to CESSAR-DC Section 11.3 is included as an attachment to the DSER Open Item 11.1-1 response.

K. J. Mortas
4/20/93

COL Action Item 15.3.10-1:

Compliance with 10 CFR 20 is demonstrated, provided that the site-specific application, which uses the ABB-CE System 80+ design verifies that (a) the failure of the Boric Acid Storage Tank is limiting, among liquid waste tanks outside containment, and (b) the site provides a minimum equivalent dilution factor of 1.438E+08.

Response to COL Action Item 15.3.10-1:

It is agreed that to demonstrate compliance with 10 CFR 20, it must be verified that the failure of the Boric Acid Storage Tank (BAST) is limiting among liquid waste tanks outside containment. However, there is a question regarding the requirement of a minimum equivalent dilution factor to be 1.438E+08 rather than 1.438E+05 (or, 1 / 6.95E-06) as calculated in the analysis presented in CESSAR-DC Section 15.7.3. ABB-CE believes the dilution factor in the COL Action Item is a typo.

In previous revisions of Section 15.7.3, 6.95E-06 was specified as the maximum allowable dilution factor. (This value has since been changed to 6.22E-06 as a result of revised reactor coolant source terms associated with a 3% rated power increase - the proposed revision to CESSAR-DC Section 15.7 is attached.) "Dilution factor" is defined as 80% BAST volume divided by the dilution volume available at the potable water source. Therefore, based on a BAST volume of 250,000 gallons, the minimum dilution volume required to ensure 10 CFR 20 limits are met is 4.30E+09 cubic feet (under the revised analysis).

ABB-CE agrees with this COL Action Item provided that the Staff accepts a minimum dilution factor of 1.608E+05.

dilution factor is ratioed to the MPC(i) and summed. This process is repeated with additional values of dilution factor until the summation of the fractions is less than or equal to 1.

B. Input Parameters and Initial Conditions

1. The concentration at the nearest potable water supply is equal to the Maximum Permissible Concentration (MPC) for each isotope.

2. The concentration in the BAST is calculated for the batch processing mode of operation described in Section 9.3.4 and shown in Figure 9.3.4-1.

3. Credit is taken for dilution by only the main flow path from the letdown through the purification process in the CVCS to the recycle evaporator. The concentrate is sent to the BAST.

4. The concentration of the flow streams are specified as a fraction of the Primary Coolant Concentration (PCC). The PCC is obtained from the output of the computer code PWR-GALE (See Section 11.1) and is consistent with NUREG-0017.

5. No additional credit is taken for radioactive decay during the purification process or during the transport of the liquid effluent to the nearest potable water source. H

6. The Decontamination Factors (DF_i) for the i-th isotope for each component were obtained from NUREG-0017 and Section 9.3.4.

7. System parameters such as flow rates and tank volumes were obtained from Section 9.3.4.

8. 80% of the volume is assumed to be released per SRP Section 15.7.3.

9. For conservatism, all radionuclides are assumed to be in the insoluble form.

C. Results

The results of the iterative process are shown in Table 15.7.3-2. The maximum allowable dilution factor was determined to be ~~6.95×10^{-6}~~ 6.22×10^{-6}.

| J

TABLE 15.7.3-1

(Sheet 1 of 2)

CONCENTRATION OF ISOTOPES IN BAST

Isotope(i)	Primary Coolant Concentration (uCi/gm)	Concentration in BAST (uCi/ml)	Concentration in BAST (uCi/ml)
Na 24	5.08E-02	3.60E-01 PCC	1.83E-02
P 32	0.00E+00	3.60E-01 PCC	0.00E+00
Cr 51	3.67E-03	3.60E-01 PCC	1.32E-03
Mn 54	1.90E-03	3.60E-01 PCC	6.84E-04
Fe 55	1.43E-03	3.60E-01 PCC	5.15E-04
Fe 59	3.56E-04	3.60E-01 PCC	1.28E-04
Co 58	5.46E-03	3.60E-01 PCC	1.97E-03
Co 60	6.30E-04	3.60E-01 PCC	2.27E-04
Ni 63	0.00E+00	3.60E-01 PCC	0.00E+00
Zn 65	6.06E-04	3.60E-01 PCC	2.18E-04
W 187	2.77E-03	3.60E-01 PCC	9.97E-04
Np 239	2.52E-03	3.60E-01 PCC	9.07E-04
Sr 89	1.66E-04	3.60E-01 PCC	5.98E-05
Sr 90	1.43E-05	3.60E-01 PCC	5.15E-06
Y 91	6.17E-06	3.60E-01 PCC	2.22E-06
Y 93	4.45E-03	3.60E-01 PCC	1.60E-03
Zr 95	4.63E-04	3.60E-01 PCC	1.67E-04
Nb 95	3.32E-04	3.60E-01 PCC	1.20E-04
Mo 99	7.36E-03	3.60E-01 PCC	2.65E-03
Tc 99M	4.84E-03	3.60E-01 PCC	1.74E-03
Ru 103	8.89E-03	3.60E-01 PCC	3.20E-03
Ru 106	1.07E-01	3.60E-01 PCC	3.85E-02
Rh 103M	0.00E+00	3.60E-01 PCC	0.00E+00
Rh 106	0.00E+00	3.60E-01 PCC	0.00E+00
Ag 110M	1.54E-03	3.60E-01 PCC	5.54E-04
Ag 110	0.00E+00	3.60E-01 PCC	0.00E+00
Te 129M	2.25E-04	3.60E-01 PCC	8.10E-05
Te 129	2.34E-02	3.60E-01 PCC	8.42E-03
Te 131M	1.68E-03	3.60E-01 PCC	6.05E-04
Te 131	7.43E-03	3.60E-01 PCC	2.67E-03
Te 132	1.96E-03	3.60E-01 PCC	7.06E-04
I 131	5.31E-02	3.51E-01 PCC	1.87E-02
I 132	2.09E-01	3.51E-01 PCC	7.35E-02
I 133	1.55E-01	3.51E-01 PCC	5.45E-02
I 135	2.70E-01	3.51E-01 PCC	9.49E-02
Cs 134	9.10E-03	3.00E+00 PCC	2.73E-02
Cs 136	1.09E-03	3.00E+00 PCC	3.27E-03
Cs 137	1.20E-02	3.00E+00 PCC	3.60E-02

TABLE 15.7.3-1 (Cont'd)

(Sheet 2 of 2)

CONCENTRATION OF ISOTOPES IN BAST

Isotope(i)	Primary Coolant Concentration (uCi/gm)	Concentration in BAST (uCi/ml)	Concentration in BAST (uCi/ml)
Ba 137M	0.00E+00	3.60E-01 PCC	0.00E+00
Ba 140	1.53E-02	3.60E-01 PCC	5.51E-03
La 140	2.83E-02	3.60E-01 PCC	1.02E-02
Ce 141	1.78E-04	3.60E-01 PCC	6.41E-05
Ce 143	3.14E-03	3.60E-01 PCC	1.13E-03
Ce 144	4.63E-03	3.60E-01 PCC	1.67E-03
Pr 143	0.00E+00	3.60E-01 PCC	0.00E+00
Pr 144	0.00E+00	3.60E-01 PCC	0.00E+00
H 3	1.00E+00	3.60E-01 PCC	3.60E-01

TABLE 15.7.3-2

(Sheet 1 of 2)

RESULTS OF ITERATIVE PROCESS TO DETERMINE DILUTION FACTOR

D_f =6.22E-06

Isotope(i)	MPC(i) (uCi/ml)	Conc. in BAST (uCi/ml)	Conc. at Potable Water (uCi/ml)	Fraction of MPC(i) FMPC(i)
Na 24	3.00E-05	1.83E-02	1.14E-07	3.79E-03
P 32	2.00E-05	0.00E+00	0.00E+00	0.00E+00
Cr 51	2.00E-03	1.32E-03	8.22E-09	4.11E-06
Mn 54	1.00E-04	6.84E-04	4.25E-09	4.25E-05
Fe 55	8.00E-04	5.15E-04	3.20E-09	4.00E-06
Fe 59	5.50E-04	1.28E-04	7.97E-10	1.45E-06
Co 58	9.00E-05	1.97E-03	1.22E-08	1.36E-04
Co 60	3.00E-05	2.27E-04	1.41E-09	4.70E-05
Ni 63	3.00E-05	0.00E+00	0.00E+00	0.00E+00
Zn 65	1.00E-04	2.18E-04	1.36E-09	1.36E-05
W 187	6.00E-05	9.97E-04	6.20E-09	1.03E-04
Np 239	1.00E-04	9.07E-04	5.64E-09	5.64E-05
Sr 89	3.00E-06	5.98E-05	3.72E-10	1.24E-04
Sr 90	3.00E-07	5.15E-06	3.20E-11	1.07E-04
Y 91	3.00E-05	2.22E-06	1.38E-11	4.61E-07
Y 93	3.00E-05	1.60E-03	9.96E-09	3.32E-04
Zr 95	6.00E-05	1.67E-04	1.04E-09	1.73E-05
Nb 95	1.00E-04	1.20E-04	7.46E-10	7.46E-06
Mo 99	4.00E-05	2.65E-03	1.65E-08	4.12E-04
Tc 99M	3.00E-03	1.74E-03	1.08E-08	3.61E-06
Ru 103	8.00E-05	3.20E-03	1.99E-08	2.49E-04
Ru 106	1.00E-05	3.85E-02	2.40E-07	2.40E-02
Rh 103M	1.00E-02	0.00E+00	0.00E+00	0.00E+00
Rh 106	3.00E-06	0.00E+00	0.00E+00	0.00E+00
Ag 110M	2.00E-05	5.54E-04	3.44E-09	1.72E-04
Ag 110	3.00E-06	0.00E+00	0.00E+00	0.00E+00
Sb 124	2.00E-05	0.00E+00	0.00E+00	0.00E+00
Te 129M	2.00E-05	8.10E-05	5.04E-10	2.52E-05
Te 129	8.00E-04	8.42E-03	5.24E-08	6.55E-05
Te 131M	4.00E-05	6.05E-04	3.76E-09	9.40E-05
Te 131	3.00E-06	2.67E-03	1.66E-08	5.55E-03
Te 132	2.00E-05	7.06E-04	4.39E-09	2.19E-04
I 131	3.00E-07	1.87E-02	1.16E-07	3.87E-01
I 132	8.00E-06	7.35E-02	4.57E-07	5.71E-02
I 133	1.00E-06	5.45E-02	3.39E-07	3.39E-01

TABLE 15.7.3-2

(Sheet 2 of 2)

RESULTS OF ITERATIVE PROCESS TO DETERMINE DILUTION FACTOR

D_f =6.22E-06

Isotope(i)	MPC(i) (uCi/ml)	Conc. in BAST (uCi/ml)	Conc. at Potable Water (uCi/ml)	Fraction of MPC(i) FMPC(i)
I 135	4.00E-06	9.49E-02	5.90E-07	1.48E-01
Cs 134	9.00E-06	2.73E-02	1.70E-07	1.89E-02
Cs 136	6.00E-05	3.27E-03	2.03E-08	3.39E-04
Cs 137	2.00E-05	3.60E-02	2.24E-07	1.12E-02
Ba 137M		0.00E+00	0.00E+00	
Ba 140	2.00E-05	5.51E-03	3.43E-08	1.71E-03
La 140	2.00E-05	1.02E-02	6.34E-08	3.17E-03
Ce 141	9.00E-05	6.41E-05	3.99E-10	4.43E-06
Ce 143	4.00E-05	1.13E-03	7.03E-09	1.76E-04
Ce 144	1.00E-05	1.67E-03	1.04E-08	1.04E-03
Pr 143	5.00E-05	0.00E+00	0.00E+00	0.00E+00
Pr 144		0.00E+00	0.00E+00	
H 3	3.00E-03	3.60E-01	2.24E-06	7.46E-04
			Total =	1.00E+00

ATTACHMENT 2

CESSAR-DC PROPOSED MARKUPS

Markup 1: Response to Confirmatory Items 14.02.12.02-1 through 14.02.12.02-4.

Markup 2: Response to Open (now Confirmatory) Item 14.02.12.01-1.

Markup 3: Response to Open Item 14.02.12.01-2 for the following tests:

1) 14.2.12.1.6	6) 14.2.12.1.91
2) 14.2.12.1.9	7) 14.2.12.1.137
3) 14.2.12.1.50	8) 14.2.12.3.1
4) 14.2.12.1.58	9) 14.2.12.4.5
5) 14.2.12.1.75	10) 14.2.12.4.6

Markup 4: Response to Open Item 14.02.13-1

temperature results in only minor changes to RCS temperatures and pressure and reactor power. In addition, the performance of this test will result in unnecessary thermal cycling of the steam generator economizer valves. The performance of load rejection test and turbine trip test from full power provides sufficient information to verify design adequacy. Thus, the plant response to reduction in feedwater temperatures will not be demonstrated. E

14.2.7.1.8 Reference Appendix A, Section 1.k.(2)

Personnel monitors and radiation survey instruments are site-specific items to be addressed by the site operator. The site operator will define the appropriate testing to demonstrate proper operation of personnel monitors and radiation survey instruments. L

Insert A

14.2.7.2 Regulatory Guide 1.79, Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

The intent of Section C.1.c(2), Isolation Valve Test, is satisfied by opening the valves under maximum differential pressure (RCS at ambient pressure) using normal electrical power only. Conditions at the valve motor are independent of the power source for this test. The breaker response and the response of the valves to the "confirmatory open" signal is verified during the Integrated Safety Injection Actuation System Test.

14.2.7.3 Regulatory Guide 1.68.2, Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants

Shutdown outside the control room will be demonstrated to Hot Standby condition. Plant cooldown to entry into Shutdown Cooling conditions will be demonstrated during Hot Functional testing. E

14.2.7.4 Regulatory Guide 1.68.3 "Preoperational Testings of Instrument and Control Air Systems

Position C.9 requires that testing demonstrate that the plant equipment designated by design to be supplied by the Instrument Air System is not degraded when supplied by the Station Air System which may have less restrictive air quality requirements. The SYSTEM 80+ Instrument Air System has no interconnection to any other compressed Air System, and, therefore, ingress of less quality air is not possible. Consequently, Position C.9 does not apply. L

Insert A

14.2.7.1.9 Reference Appendix A, Section 1.h.(5)

Cold Water Interlocks are not applicable to the System 80+ design. This testing will not be performed as it is not applicable.

14.2.7.1.10 Reference Appendix A, Section 1.i.(21)

A containment penetration cooling system is not a design requirement for the System 80+. This testing will not be performed as it is not applicable.

14.2.7.1.11 Reference Appendix A, Section 1.n.(15)

A shield cooling system is not a design requirement for the System 80+. This testing will not be performed as it is not applicable.

14.2.7.1.12 Reference Appendix A, Section 4.i

Demonstration of the operability of control rod withdrawal and insertion sequences and control rod inhibit or block functions is performed during pre-critical functional testing. The reactor power level range during which such features must be operable is performed using simulated signals, as required.

LIST OF TABLES

CHAPTER 14

E

14.2.12 INDIVIDUAL TEST DESCRIPTIONS

14.2.12.1 Preoperational Tests

14.2.12.1.1 Reactor Coolant Pump (RCP) Motor Initial Operation

1.0 OBJECTIVE

1.1 To verify the proper operation of each RCP motor.

1.2 To collect base data for each RCP motor.

2.0 PREREQUISITES

2.1 RCP motor instrumentation has been calibrated.

2.2 Each RCP motor and its respective pump are uncoupled.

2.3 Support systems required for operation of each RCP motor are operational.

3.0 TEST METHOD

3.1 Start Component Cooling Water (CCW) flow to the RCP motor and observe indicating lights and alarms.

3.2 Using a torque wrench and phase rotation meter, rotate RCP motor and verify proper wiring of motor leads and torque required to rotate the motor.

3.3 Jog RCP motor and verify proper rotation.

3.4 Start RCP motor and verify proper operation. Record motor operating data.

3.5 Determine oil level setpoints of oil reservoirs by draining oil from motor reservoirs and subsequently refilling.

3.6 Simulate oil lift pumps and CCW system starting interlocks preventing RCP motor operation and observe effects.

4.0 DATA REQUIRED

4.1 Motor operating data.

4.2 Torque needed to rotate the RCP motors.

Note: Refer to Table 14.2-7 for additional details on the major support systems that are recommended to be available for the preoperational tests.

Table 14.2-7

MATRIX OF SUPPORT SYSTEMS ~~REQUIRED~~ Recommended FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-1E POWER	1-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER
RCP Initial Operation			X	X	X	X	X					X			X		
Reactor Coolant System (RCS) Test			X	X	X	X	X					X			X		
Pressurizer Safety Valve Test *																	
Pressurizer Pressure and Level Control Systems			X	X	X	X	X										X
Chemical and Volume Control System (CVCS) Letdown Subsystem Test			X	X	X	X	X						X				
CVCS Purification Subsystem Test			X	X	X	X	X						X				
Volume Control Tank (VCT) Subsystem Test			X	X	X	X	X						X				
CVCS Charging Subsystem Test			X	X	X	X	X						X				
Chemical Addition Subsystem Test			X	X	X	X	X						X				
Reactor Drain Tank (RDT) Subsystem Test			X	X	X	X	X						X				
Equipment Drain Tank (EDT) Subsystem Test			X	X	X	X	X										
Boric Acid Batching Tank (BABT) Subsystem Test			X				X										
Concentrated Boric Acid Subsystem Test			X	X	X	X	X						X				

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

IRWST - In Containment Refueling Water Storage Tank
CCS-ESF - ~~Central~~ Component Controlling System for Engineered Safety Features
CCS-Process - Component Control System-Process
DPS - Data Processing System
DIAS - Discrete Indication and Alarm System
MCR - Main Control Room
EFW - Emergency Feedwater

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Reactor Makeup (RMW) Subsystem Test			X	X	X	X	X																	
Holdup Subsystem Test			X	X	X	X	X																	
Boric Acid Concentrator Subsystem Test			X			X	X																	
Gas Stripper Subsystem Test			X	X	X	X	X																	
Boronometer Subsystem Test						X	X																	
Letdown Process Radiation Monitor Subsystem Test			X		X	X	X																	
Gas Stripper Effluent Radiation Monitor Subsystem Test			X		X	X	X																	
Shutdown Cooling System Test	X	X		X	X	X	X	X																
Safety Injection System Test	X	X	X	X	X	X	X	X	X	X		X				X								
Safety Injection Tank Subsystem Test		X		X	X	X	X	X																
Megawatt Demand Setter (MDS) System Test			X	X	X	X	X																	
Engineered Safety Features-Component Control System (ESF-CCS) Test		X		X	X	X	X	X								X								
Plant Protection System (PPS) Test		X		X	X	X	X	X								X								

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER
Ex-core Nuclear Instrumentation System Test				X	X	X	X									X	
Fixed In-Core Nuclear Signal Channel Test				X	X	X	X										
Control Element Drive Mechanism Control System (CEDMCS) Test				X	X	X	X										
Reactor Regulating System (RRS) Test			X	X	X	X	X										
Steam Bypass Control System (SBCS) Test			X	X	X	X	X										
Feedwater Control System (FWCS) Test			X	X	X	X	X										
Core Operating Limit Supervisory System (COLSS) Test				X		X											
Reactor Power Cutback System (RPCS) Test			X	X	X	X	X										
Fuel Handling and Storage System Test							X										
Emergency Feedwater (EFW) System Test		X		X	X	X	X	X		X		X			X	X	
Reactor Coolant System Hydrostatic Test *																	
Control Element Drive Mechanism (CEDM) Cooling System Test			X	X	X	X	X										
Safety Depressurization System Test	X	X		X	X	X		X									

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Containment Spray System (CSS) Test	X	X		X	X	X	X	X		X															
Integrated Engineered Safety Features/Loss of Power Test	X	X		X	X	X	X	X	X	X															
In-containment Water Storage System Test	X			X	X	X	X																		
Internals Vibration Monitoring System Test				X	X	X	X																		
Loose Part Monitoring System Test				X	X	X	X																		
Acoustic Leak Monitoring System Test				X	X	X	X																		
Data Processing System and Discrete Indication and Alarm System Test				X	X	X	X																		
Critical Function Monitoring (CFM) System Test				X	X	X	X																		
Pre-core Hot Functional Test Controlling Document **																									
Pre-core Instrument Correlation Remote Shutdown Panel **																									
Alternate Protection System Test			X	X	X	X	X																		
Pre-core Test Data Record **																									

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Pre-core Reactor Coolant System (RCS) Expansion Measurements **																								
Pre-core Reactor Coolant and Secondary Water Chemistry Data **																								
Pre-core Pressurizer Performance **																								
Pre-core Control Element Drive Mechanism (CEDM) Performance **																								
Pre-core Reactor Coolant System (RCS) Flow Measurements **																								
Pre-core Reactor Coolant System (RCS) Heat Loss **																								
Pre-core Reactor Coolant System Leak Rate Measurement **																								
Pre-core Chemical and Volume Control System (CVCS) Integrated Test **																								
Pre-core Safety Injection Check Valve Test **																								
Pre-core Boration/Dilution Measurements **																								

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	1-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Downcomer Feedwater System Water Hammer Test **																								
Main Turbine Systems Test			X	X	X	X	X							X										
Main Steam Safety Valve Test **																								
Main Steam Isolation Valves (MSIVs) and MSIV Bypass Valves Test **																								
Main Steam System Test			X	X	X	X	X																	
Steam Generator Blowdown System Test			X	X	X	X	X																	
Main Condenser and Air Removal Systems Test			X	X	X	X	X																	
Main Feedwater System Test				X	X	X	X																	
Condensate System Test			X	X	X	X	X					X		X										
Turbine Gland Sealing System Test *																								
Condenser Circulating Water System Test			X	X	X	X	X							X										
Steam Generator Hydrostatic Test **																								
Feedwater Heater and Drains System Test			X	X	X	X	X																	
Ultimate Heat Sink System Test							X				X						X							

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	OPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Chilled Water System Test						X	X	X			X						X								
Station Service Water System Test			X			X	X	X						X											
Component Cooling Water (CCW) System Test		X	X	X	X	X	X	X			X			X	X										
Spent Fuel Pool Cooling and Cleanup System Test		X	X			X	X																		
Turbine Building Cooling Water System Test							X				X						X								
Condensate Storage System Test							X																		
Turbine Building Service Water System Test							X							X											
Equipment and Floor Drainage System Test							X																		
Normal and Security Lighting Systems Test							X																		
Emergency Lighting System Test								X																	
Communications System Test							X																		
Compressed Air System Test							X					X													
Compressed Gas System Test							X																		
Process Sampling System Test							X																		

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER
Heat Tracing System Test							X										
Fire Protection Systems Test							X										
Diesel Generator Mechanical System Test							X			X							
Diesel Generator Electrical System Test		X	X							X							
Diesel Generator Auxiliary Systems Test			X							X							
Alternate AC Source System Test				X		X	X		X								
Alternate ACE Source Support Systems Test						X		X	X								
Containment Polar Crane Test							X										
Fuel Building Cranes Test							X										
Turbine Building Crane Test							X										
Containment Cooling and Ventilation System Test				X	X	X	X				X						
Containment Purge System Test				X		X	X										
Control Building Ventilation System Test						X	X				X						
Reactor Subsphere and Nuclear Annex Ventilation System Test						X	X				X						

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Turbine Building Ventilation System Test						X	X				X														
Station Service Water Pump Structure Ventilation System Test						X	X				X														
Diesel Building Ventilation System Test						X	X				X														
Fuel Building Ventilation System Test						X	X				X														
Annulus Ventilation System Test		X				X	X	X			X														
Radwaste Building Ventilation System Test						X	X				X														
Control Building Ventilation Subsystems Test						X	X				X														
Hydrogen Mitigation System (HMS) Test							X																		
Containment Hydrogen Recombiner System (CHRS) Test							X																		
Liquid Waste Management System Test			X	X		X	X																		
Solid Waste Management System Test			X	X		X	X																		
Gaseous Radwaste Management System Test			X	X		X	X																		
Process and Effluent Radiation Monitoring System Test						X	X																		

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-1E POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER								
Airborne and Area Radiation Monitoring System Test						X	X																		
4160 Volt Class 1E Auxiliary Power System Test				X	X	X		X																	
480 Volt Class 1E Auxiliary Power System Test				X	X	X		X																	
Unit Main Power System Test						X	X																		
13800 Volt Normal Auxiliary Power System Test				X	X	X	X																		
4160 Volt Normal Auxiliary Power System Test				X	X	X	X																		
480 Volt Normal Auxiliary Power System Test				X	X	X	X																		
Non-Class 1E DC Power Systems Test				X	X	X		X																	
Class-1E DC Power Systems Test				X	X	X		X																	
Offsite Power System Test				X		X	X																		
Balance of Plant (BOP) Piping Thermal Expansion Measurement Test **																									
BOP Piping Vibration Measurement Test **																									

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

MATRIX OF SUPPORT SYSTEMS REQUIRED FOR PRE-OPERATIONAL TESTS

SUBJECT	IRWST	CCS-ESF	CCS-PROCESS	DPS	DIAS	MCR	NON-IE POWER	I-E POWER	ALTERNATE AC	DIESEL GENERATOR	COOLING WATER	COMPRESSED AIR	AUXILIARY STEAM	SERVICE WATER	EFW TANK	REMOTE SHUTDOWN PANEL	CHILLED WATER
Containment Integrated Leak Rate Test and Structural Integrity Test **																	
Fuel Transfer Tube Functional Test and Leak Test *																	
Equipment Hatch Functional Test and Leak Test *																	
Containment Personnel Airlock Functional Test and Leak Test *																	
Electrical Penetration Test **																	
Containment Isolation Valves (CIVs) Test							X	X									
Loss of Instrument Air Test **																	
Mid-Loop Operations Verification Test **																	
Seismic Monitoring Instrumentation System Test							X										
Auxiliary Steam System Test *						X	X										
Containment Isolation Valve Test		X				X	X	X									
Post Accident Monitoring Test						X	X										

* Support Systems not required for testing.

** This test requires essentially a mechanically operable plant.

14.2.12.1.6 CVCS Purification Subsystem Test

1.0 OBJECTIVE

1.1 To verify flowpaths between the reactor makeup water system, the purification and deborating ion exchangers and the Solid Waste Management System.

1.2 To verify flowpaths between the purification and deborating ion exchanger and Gaseous Waste Management System.

2.0 PREREQUISITES

2.1 Construction activities on the Chemical and Volume Control System (CVCS) Purification Subsystem have been completed.

2.2 CVCS Purification Subsystem instrumentation has been calibrated.

2.3 Test instrumentation has been calibrated.

2.4 Support systems required for operation of the CVCS purification subsystem are complete and operational.

3.0 TEST METHOD

3.1 Lineup the purification system ion exchangers to complete a flowpath from the Reactor Makeup Water (RMW) System through each CVCS Purification Subsystem ion exchanger to the Solid Waste Management system. Start an RMW pump and sequentially, so that only one ion exchanger is in use at a time, valve-in each ion exchanger. Verify flow by observing RMW flow indicators and changes in RMW and spent resin tank levels. Select all possible flow paths to the Solid Waste Management System.

3.2 Individually connect each purification ion exchanger and the deborating ion exchanger to the plant air system and connect a pressure gage to the ion exchanger vent. Adjust the plant air supply to 15-20 psig. Start air flow to the ion exchangers and individually open each ion exchanger vent valve and valve the ion exchanger to the gaseous waste management system. Observe the ion exchanger vent pressure, gas supply pressure, and flow rate.

4.0 DATA REQUIRED

4.1 RMW flow rate

4.2 RMW and Spent Resin Tank levels

4.3 Air supply pressure and flow rate

4.4 Ion exchanger test pressure

5.0 ACCEPTANCE CRITERIA

5.1 Verification of flowpaths between the RMW system, the purification and deborating ion exchangers, and the Solid Waste Management System will have been demonstrated upon successful completion of Test Method 3.1.

5.2 Verification of flowpaths between the purification and deborating ion exchangers and the Gaseous Waste Management system will have been demonstrated upon successful completion of Test Method 3.2.

5.3 The CVCS Purification Subsystem performs as described in Section 9.3.4. [0]

14.2.12.1.9 Chemical Addition Subsystem Test

1.0 OBJECTIVE

1.1 To demonstrate that the Chemical Addition Subsystem can inject water into the charging pump discharge line down stream of the Seal Injection take off connection. H

1.2 To verify a flowpath from the Chemical Addition Tank to the Miscellaneous Liquid Waste Management System.

2.0 PREREQUISITES

2.1 Support systems required for operation of the Chemical Addition Subsystem are complete and operational.

2.2 The Chemical Addition Tank has been filled from the makeup system with a pre-determined amount of RMW.

2.3 Charging Subsystem is in operation.

2.4 Associated instrumentation has been calibrated.

3.0 TEST METHOD

3.1 With a charging pump in operation, start the chemical Addition Metering Pump and observe the chemical addition tank level.

3.2 Drain the Chemical Addition Tank to the Miscellaneous Liquid Waste Management System and observe the Chemical Addition Tank level. E

4.0 DATA REQUIRED

4.1 Chemical Addition Tank levels.

5.0 ACCEPTANCE CRITERIA

5.1 Chemical addition to charging line down stream of the seal injection take off connection is demonstrated when Test Method 3.1 is completed with a decreasing chemical addition tank level. H

5.2 A flowpath to the Miscellaneous Liquid Waste Management System is demonstrated when Test Method 3.2 is completed with a decreasing Chemical Addition Tank level.

5.3 The Chemical Addition Subsystem performs as described in Section 9.3.4. [O]

14.2.12.1.50 Remote Shutdown Panel

1.0 OBJECTIVE

1.1 To verify proper operation of the Remote Shutdown Instrumentation.

1.2 To determine that the plant can be cooled down from the Remote Shutdown Panel.

2.0 PREREQUISITES

2.1 All construction activities on the Remote Shutdown Panel have been completed.

2.2 All Remote Shutdown Panel instrumentation has been calibrated.

2.3 The communication systems between the control room and Remote Shutdown Panel location has been demonstrated to be operational.

3.0 TEST METHOD

3.1 Using simulated signals, verify proper operation of remote shutdown panel instrumentation.

3.2 During preoperational Post-Core Hot Functional tests, perform a controlled cooldown from the Remote Shutdown Panel.

E

4.0 DATA REQUIRED

4.1 RCS temperatures, pressures.

5.0 ACCEPTANCE CRITERIA

5.1 The ability to cooldown using Remote Shutdown Instrumentation has been demonstrated.

5.2 The Remote Shutdown Panel performs as described in Section 7.4.1.1.10. [O]

14.2.12.1.58 Pre-core Reactor Coolant System (RCS) Heat Loss

1.0 OBJECTIVE

1.1 Measure RCS heat loss under hot, zero power conditions.

1.2 Measure pressurizer heat loss under hot, zero power conditions.

2.0 PREREQUISITES

2.1 Test instrumentation is available and calibrated.

2.2 Construction activities on the RCS and associated systems are completed.

2.3 All permanently installed instrumentation on the system to be tested is available and calibrated.

3.0 TEST METHOD

3.1 Determine the RCS heat loss using the steam-down method:

3.1.1 Stabilize the Steam Generator levels with the RCS at hot, zero power conditions.

3.1.2 Secure Steam Generator feedwater and blowdown.

3.1.3 Measure both the Pressurizer heater power required to maintain RCS temperature and pressure and RCP power.

3.1.4 Perform a heat balance calculation to determine heat loss.

3.2 Determine the Pressurizer heat loss, with* and without continuous spray flow, by measuring the pressurizer heater power required to maintain the RCS at hot, zero power conditions, and then performing a heat balance calculation. E

* Pressurizer heat loss with continuous spray flow to be determine during post core hot functional test after spray valve adjustments have been performed per Section 14.2.12.2.6. E

4.0 DATA REQUIRED

4.1 RCS temperatures.

4.2 Pressurizer pressure and level.

4.3 Steam generator pressures and levels.

4.4 Pressurizer heater power.

4.5 RCP power.

5.0 ACCEPTANCE CRITERIA

5.1 The measured heat loss is less than~~, or equal to, the anticipated heat loss or an engineering evaluation finds the results acceptable.~~

The capacity of the containment cooling subsystem to remove the heat loads as described in Section 9.4.6.

14.2.12.1.75 Feedwater Heater and Drains System Test

1.0 OBJECTIVE

1.1 To demonstrate the Feedwater Heater and Drain System alarms and controls operate as designed.

1.2 To demonstrate that the Feedwater Heaters and Drains System is capable of heating the Main Feedwater System to the design temperature for normal plant operation. (PAT)

2.0 PREREQUISITES

2.1 Construction activities on the Feedwater Heater and Drains System have been completed.

2.2 Feedwater Heater and Drains System instrumentation has been calibrated.

2.3 Individual component testing is complete

2.4 The power conversions systems are operating as required to support the test.

3.0 TEST METHOD

3.1 Verify the setpoints of alarms and interlock. H

3.2 Operate control valves from all appropriate control positions. Observe valve operation an position indication and measure opening and closing times.

3.3 Simulate failed conditions and observe valve response.

3.4 Verify Main Feedwater temperature to the Steam Generators at 100% flow is as designed. (PAT)

3.5 Demonstrate that high pressure feedwaters level controls maintain proper level and drain to the deareator. (PAT)

3.6 Demonstrate that the low pressure feedwater heaters level control maintains proper level and drain to the main condenser. (PAT)

4.0 DATA REQUIRED

4.1 Valve opening and closing times, where required.

4.2 Valve position indication.

4.3 Response of valves to simulated failed conditions.

4.4 Setpoints at which alarms and interlocks occur.

4.5 Feedwater temperature at 100% flow for each heater group. H

5.0 <u>ACCEPTANCE CRITERIA</u>

5.1 The Feedwater Heater and Heater Drains System ~~maintains Main Feedwater temperature as designed.~~ performs as described in Section 10.4.7. [O]

14.2.12.1.91 Heat Tracing Systems Test

1.0 OBJECTIVE

1.1 Verify that plant heat traced components are maintained at design temperature.

2.0 PREREQUISITES

2.1 Construction activities on the Heat Tracing System have been completed.

2.2 Heat Tracing System instrumentation has been calibrated.

2.3 Support systems required for operation of the Heat Tracing system are complete and operational. H

2.4 Test Instrumentation is available and calibrated.

2.5 Electrical power supply available.

3.0 TEST METHOD

3.1 With system process at design flow, verify the Heat Tracing System maintains each component within its minimum and maximum design temperature limits by checking temperatures at various points.

3.2 Demonstrate the operation of controls and alarms.

4.0 DATA REQUIRED

4.1 Temperature data for the heat traced components.

4.2 Setpoints of alarms and control points.

5.0 ACCEPTANCE CRITERIA

5.1 The Heat Tracing System maintains designated components within design temperature limits, as described in Section 9.3.4.

14.2.12.1.137 Mid-Loop Operations Verification Test

1.0 OBJECTIVES

1.1 To verify that installed instrumentation for operations at reduced Reactor Coolant System (RCS) inventory is accurate and reliable.

1.2 To verify the Shutdown Cooling System (SCS) pumps can be operated at reduced RCS level without cavitation.

2.0 PREREQUISITES

2.1 Construction activities on the RCS Mid-Loop Instrumentation system have been completed.

2.2 RCS Mid-Loop System instrumentation has been calibrated.

2.3 Support systems required for Mid-Loop operations are completed and operational.

2.4 Test instrumentation of high accuracy to measure RCS level changes is available and calibrated. H

2.5 The RCS is at normal shutdown level in the Pressurizer and depressurized.

2.6 The Shutdown Cooling System is operable.

3.0 TEST METHOD

3.1 Verify the operation of the RCS mid-loop level instrumentation indication and alarms.

3.2 Verify the operation of the SCS pumps while operating at mid-loop level.

3.3 Establish the minimum level at which the SCS pumps can operate without cavitation.

3.4 Establish the maximum flow the SCS pumps can operate at mid-loop without cavitation.

4.0 DATA REQUIRED

4.1 Setpoints of alarms.

4.2 Mid-Loop Instrumentation Data

4.3 Minimum level and maximum flow limits for the SCS pumps.

5.0 ACCEPTANCE CRITERIA

5.1 The Mid-Loop Instrumentation provides accurate indication of RCS parameters, as described in Section 16.13.2 H

5.2 The SCS pump operating limits at mid-loop are established and within the expected design range, as described in Section 19.6.3.9

14.2.12.3 Low Power Physics Tests

14.2.12.3.1 Low Power Biological Shield Survey Test

1.0 OBJECTIVE

1.1 To measure radiation in accessible locations of the plant outside of the biological shield.

1.2 To obtain baseline levels for comparison with future measurements of level buildup with operation.

2.0 PREREQUISITES

2.1 Radiation survey instruments calibrated.

2.2 Background radiation levels measured in designated locations prior to initial criticality.

3.0 TEST METHOD

3.1 Measure gamma and neutron dose rates during low power (<5% Reactor Thermal Power (RTP)) operation. E

4.0 DATA REQUIRED E

4.1 Power level.

4.2 Gamma and neutron dose rates at each specified location.

5.0 ACCEPTANCE CRITERIA E

5.1 Baseline neutron and Gamma surveys have been completed.

5.2 The biological shield performs as described in Section 12.3.2.2. [0]

14.2.12.4.5 Turbine Trip Test

1.0 OBJECTIVE

1.1 To demonstrate that the plant responds and is controlled as designed following a 100% turbine trip without RPCS in service.

2.0 PREREQUISITES

2.1 The reactor is operating above 95% power.

2.2 The SBCS, FWCS, RRS, and pressurizer pressure and level control systems are in automatic operation.

2.3 The RPCS is in Auto Actuate Out of Service.

3.0 TEST METHOD

3.1 The turbine is tripped.

3.2 The plant behavior is monitored to assure that the RRS, SBCS, FWCS, and pressurizer pressure and level control systems maintain the NSSS within operating limits.

4.0 DATA REQUIRED

4.1 Plant condition prior to trip. E

4.2 The following acceptance criteria parameters are monitored prior to and throughout the transient:

4.2.1 Pressurizer pressure and level

4.2.2 RCS hot leg temperatures

4.2.3 SG pressures

4.2.4 CEA drop times

4.3 Additional key plant parameters will be monitored for base line data.

5.0 ACCEPTANCE CRITERIA

5.1 ~~The test will be evaluated against single valued acceptance limits.~~ E
The measured values of the acceptance criteria parameters in section 4.2 (above) are within the single valued acceptance limits based on test predictions.

5.2 The reactor trips on high pressurizer pressure.

14.2.12.4.6 Unit Load Rejection Test

1.0 OBJECTIVE

1.1 To demonstrate that the plant responds and is controlled as designed following a 100% load rejection with RPCS in service.

2.0 PREREQUISITES

2.1 The reactor is operating above 95% power.

2.2 The SBCS, FWCS, RRS, CEDMCS, RPCS, and pressurizer pressure and level control are in automatic operation.

3.0 TEST METHOD

3.1 A breaker(s) is tripped so as to subject the turbine to the maximum credible overspeed condition.

3.2 The plant behavior is monitored to assure that the RRS, CEDMCS, SBCS, RPCS, FWCS, and pressurizer pressure and level control systems maintain the monitored parameters.

4.0 DATA REQUIRED

4.1 Plant condition prior to load rejection.

4.2 The following acceptance criteria parameters are monitored prior to and throughout the transient:

4.2.1 Pressurizer pressure and level

4.2.2 RCS hot leg temperatures

4.2.3 SG pressures

4.3 Additional key plant parameters will be monitored for baseline data.

5.0 ACCEPTANCE CRITERIA

5.1 ~~The test will be evaluated against single valued acceptance limits.~~ E
The measured values of the acceptance criteria parameters in section 4.2 (above) are within the single valued acceptance limits based on test predictions.

5.2 A reactor trip does not occur during the test.

5.3 The Reactor Power Cutback System operates as described in Section 7.7.1.1.6.

RESPONSE TO OPEN ITEM 14.2.13-1

14.2.12.1.85 **Normal and Security Lighting Systems Test**

1.0 OBJECTIVE

1.1 To demonstrate that the Normal and Security Lighting Systems provide adequate illumination for plant operations.

2.0 PREREQUISITES

2.1 Construction activities on the Normal Lighting System have been completed.

2.2 Construction activities on the Security Lighting System have been completed.

2.3 Test Instruments are properly calibrated and available. H

3.0 TEST METHOD

3.1 Place the plant lighting in service and check that illumination levels are adequate.

3.2 Demonstrate that a single circuit failure will not cause the loss of all lighting in a room which requires normal access.

3.3 Demonstrate that loss of normal power results in proper activation of the Security Lighting System for each affected room, where required to monitor isolation zones, and outdoor areas within the plant protected perimeter. J

3.4 Demonstrate the Security Lighting System provides adequate illumination levels, including, but not limited to, those required to support plant Closed Circuit TV security functions.

4.0 DATA REQUIRED

4.1 Illumination levels in designated areas. H

5.0 ACCEPTANCE CRITERIA

5.1 The Normal and Security Lighting Systems operate as described in Section 9.5.3.

ATTACHMENT 3

ABB
ASEA BROWN BOVERI

I.5 Cont'd

Calculation Number | Rev.

27
Page Number

I.5.2 List of Runs

Run No.	ID	Description	Input File(s)	Output File(s)
1	IKCF32K5 2/17/93	surge line analysis - all loading conditions and ASME Code Class 1 design check	S80PSL.INP	S80PSL.OUT
2	IKCBWMAJ 12/9/92	generate plots of nozzle spectra (unbroadened) for individual soil cases, N-411 damping	see microfiche	spec.noz.out
3	IKCBW0B3 12/9/92	generate plots of building spectra (unbroadened) for individual soil cases, N-411 damping	"	spec.169.out
4	IKCBXRCZ 12/9/92	generate and plot broadened nozzle spectra for individual soil cases as well as envelope of soil cases (except for soil case C1.5, y direction)	"	surge.noz.out
5	IKCBYVFF 12/9/92	generate and plot broadened nozzle spectra for soil case C1.5, y direction	"	surge.C15y.noz.out

I. 5 CONT'D

I. 5.2 LIST OF RUNS

RUN NO.	ID	DESCRIPTION	INPUT FILE	OUTPUT FILE
1	1KC9JSNH Microfiche ID: 1KCZR5CT	Surge Line Analysis for all Loadings and ASME Code Class I Design Check	S8OPSL. REV1	S8OPSL. REV1.OUT *

* APPENDIX C

[redacted]

COMPLETE COMPUTER RUN LISTING

MEMBER NAME	MICROFICHE ATT. #	RUN ID #	DATE	TIME	ANALYSIS TYPE
PREFW	M1	Job #10848	01-05-93	12:31:09	SUPERPIPE VERSION 22E

INPUT DECKS FOR THE RUNS LISTED ABOVE ARE STORED ON PARTITIONED DATA SET DK831.P8363.MCGPIP AS THE MEMBERS LISTED ABOVE.

REV#	ORIG	DATE	CHKD	DATE
0	JMH	1-25-93	GAP	2-15-93

DUKE ENGINEERING & SERVICES, INC.

PROJECT: Main Feedwater Line for ABB-CE System 80+ ALWR Plant

CALC. NO. 4248-04-1627.00-0003

[redacted]

COMPLETE COMPUTER RUN LISTING

MEMBER NAME	MICROFICHE ATT. #	RUN ID #	DATE	TIME	ANALYSIS TYPE
PREMSA	M1	TSU #07974	01-04-93	19:00:13	SUPERPIPE VERSION 22E
PREMSB	M2	Job #10784	01-06-93	15:34:11	SUPERPIPE VERSION 22E

INPUT DECKS FOR THE RUNS LISTED ABOVE ARE STORED ON PARTITIONED DATA SET DK831.P8363.MCGPIP AS THE MEMBERS LISTED ABOVE.

REV#	ORIG	DATE	CHKD	DATE
0	JMH	1-10-93	GAP	2-15-93

DUKE ENGINEERING & SERVICES, INC.

PROJECT: Main Steam Line for ABB-CE System 80+ ALWR Plant

CALC. NO. 4248-04-1627.00-0004

[redacted]

COMPLETE COMPUTER RUN LISTING

MEMBER NAME	MICROFICHE ATT. #	RUN ID #	DATE	TIME	ANALYSIS TYPE
DVIPRE93	M1	Job # 10126	1-11-93	22:06:35	SUPERPIPE VERSION 22E

INPUT FOR THE RUN LISTED ABOVE IS STORED ON PARTITIONED DATA SET <u>DK831.P8363.MCGPIP(DVIPRE93)</u>.

REV#	ORIG	DATE	CHKD	DATE
0	CER	1-30-93	JH	1/30/93

DUKE ENGINEERING & SERVICES, INC.

PROJECT: Direct Vessel Injection Line for ABB-CE System 80+ ALWR

CALC. NO. 4248-04-1627.00-0001 PAGE: 52

[redacted]

COMPUTER ANALYSES RECORD
Shutdown Cooling System

Member Name	IDENTIFIER		Fiche Job No.	Comments/Description	Fiche Att. #
	Date	Time			
GEOMETRY	1/14/93	21.00.24	14598	System geometry	M3
GRAVITY	1/14/93	21.16.49	14985	Gravity analysis	M5
THERMAL	1/14/93	21.17.43	14986	All thermal expansion analyses	M4
SEISMIC	1/14/93	21.12.09	14982	Dynamic properties, response spectrum analyses	M6
THTRDAT	1/14/93	21.19.42	14988	Thermal transient data stored	M9
CLASS1	1/14/93	21.37.17	03023	ASME class 1 stresses	M11
CLASS2	1/14/93	21.20.35	14990	ASME class 2 stresses and stored load combinations	M12
NOZLOADS	2/4/93	20.09.26	5643	Endload summary for hot leg branch nozzle	M10
COMBPRT	2/4/93	12.11.47	12216	Various results sets combinations	M8
SAMS	1/14/93	21.16.06	14984	Seismic anchor motion	M7
TR2A1	9/22/92	10.37.45	06271	Thermal transient thru-wall stress (Cases ONE to EIGH)	M1
TR2A2	10/6/92	11.02.40	06487	Thermal transient thru-wall stress (Case NINE)	M2
CLASS1R1	3/4/93	17:21:29	12788	ASME Class 1 stresses w/corrections (see pg 58)	M14
GEO2	3/4/93	17:20:20	12783	Corrected geometry for use with CLASS1R1 stress check	M13

Member are located in DPCo PDS file: DK831.P8363.DESCOPIP(ALWRFILE)

REV#	ORIG	DATE	CHKD	DATE	DUKE ENGINEERING & SERVICES, INC.
0	GJS	3/11/93	HVD	3-12-93	PROJECT: Piping Analysis for Shutdown Cooling System for ALWR Plant
					FILE: 4248-04-1627.00-0002
					PAGE: