Document Title: CALCULATION OF CONTAINMENT LEAKAGE DOSES FOR BROWNS FERRY

RECORD OF REVISIONS

Rev. No.	Explanation of Revision	Revised Page No.	Author	Reviewer	Approver	Date
0	Initial Document	n/a	R. E. Schneider	M. Michonski	R. E. Jaquith	9/29/95

CHECKLIST NO. 2
REVIEW OF DESIGN ANALYSIS

1. Is the material presented sufficiently detailed as to purpose, method, assumptions, design input, references, and units? ☑ Yes ☐ N/A

2. Were the inputs correctly selected and incorporated into the analysis? ☑ Yes ☐ N/A

3. Have the assumptions necessary to perform the analysis been adequately documented and justified? ☑ Yes ☐ N/A

4. Are applicable codes, standards, and regulatory requirements, including issue and addenda, employed in the analysis properly identified, and were their requirements met? ☑ Yes ☐ N/A

5. Have interface requirements been satisfied? ☑ Yes ☐ N/A

6. Have the adjustment factors, uncertainties, and empirical correlations used in the analysis been correctly applied? ☑ Yes ☐ N/A

7. Was an appropriate analysis or calculation method used? ☑ Yes ☐ N/A

8. Have the versions of the computer codes employed in the analysis been certified for application? If not, has sufficient information been provided to enable verification of the program and results? ☑ Yes ☐ N/A

9. Is the purpose sufficiently clear, and are the results and conclusions reasonable when compared to inputs? ☑ Yes ☐ N/A

10. Has an appropriate title page similar to Exhibit 3.4-1 been used? ☑ Yes ☐ N/A

11. Are all pages sequentially numbered and marked with the analysis number? ☑ Yes ☐ N/A

12. Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed? ☑ Yes ☐ N/A

13. Is the presentation legible and reproducible? ☑ Yes ☐ N/A

14. Have all cross-outs or overstrikes in the documentation been initialed and dated by the Author? ☑ Yes ☐ N/A

Michael D. Michonski 9/29/95
Reviewer Signature Date

REVIEWER'S COMMENTS

No significant errors were discovered. The results of the cases are reasonable considering the inputs which were used.

TABLE OF CONTENTS

I. INTRODUCTION/PURPOSE

The purpose of this calculation is to provide containment leakage dose assessments for the TVA Browns Ferry Nuclear Unit for a design basis maximum hypothetical accident using source term input based on the revised source term as defined in NUREG-1465 (Reference 7). This information will be combined with other calculations to be performed by Polestar Applied Technology, Inc.(PSAT) to establish the total radiological dose following the 10CFR100 maximum hypothetical accident.

Specifically, this report provides two calculations for the 2 hour and 30 day doses in the control room and at the exclusion area boundary (EAB) and low population zone (LPZ):

Case 1: Three SGTS Fans
no SGTS/CREVS charcoal filters
SGTS flow of 22,000 CFM

Case 2: Two SGTS Fans
no SGTS/CREVS charcoal filters
SGTS flow of 15,000 CFM

Main steam line leakage is assumed to be initially 120 CFH and increase after 7230 seconds to 177.5 CFH and remain at that value for the remainder of the 30 day calculation.

Calculations will be performed using TVADOSE (See Reference 2) (version TVD92395). The equations governing TVADOSE are presented herein. The validation of TVADOSE for application to Browns Ferry is contained in Reference 2.

Results of this calculation show that the leakage contribution to the 30 day control room dose is under 18 Rem. Case 1 was predicted to be the more limiting case due to the higher outflow from the Reactor Building. Additional details, including the EAB and LPZ thyroid, and whole body are presented in Section VII.

II. DISCUSSION and METHOD OF ANALYSIS

II.1 Introduction

The methodology used for this calculation is defined below. The equations defined in this section are implemented for the Browns Ferry Nuclear Unit. The resultant program (TVADOSE) has evolved from an Combustion Engineering's "in house" LDOSE computer code (Reference 1). The modified code version is fully described and specifically qualified for use in the Browns Ferry calculation in Reference 2 . Modifications made to LDOSE to create TVADOSE are based on the workscope outlined in Reference

The model and equations implemented in TVADOSE are based on standard engineering methodology for the calculation of activity transport; doses calculations are based on a Dose Conversion Factors (DCFs) Methodology with DCFs provided to ABB via Polestar Applied Technology , Inc. in Reference 3. All relevant equations are presented in this document.

II.2 Overview of Model

II.2.1 Calculation of Area Activity

The computer model to be used for TVADOSE consists of 7 nodes, with eight identified regions (See Figure II-1). These regions are:

1. Atmosphere
2. Drywell
3. Wetwell
4. Control Room
5. Reactor Building
6. Stack Base
7. Main Steam Line Piping
8. Condenser

The model follows the guidelines of Reference 2. The regions are defined as follows:

FIGURE II.1. NODAL ARRANGEMENT FOR TVADOSE

FIGURE II.2. SOURCES OF ACTIVITY TO THE CONTROL ROOM

FIGURE II.3 SOURCES OF ACTIVITY TO THE EXCLUSION AREA BOUNDARY

FIGURE II.4 SOURCES OF ACTIVITY TO THE LOW POPULATION ZONE

Region 1: Atmosphere

Region 1 is not used in the activity calculations. Their impact on dose is evaluated based on the methodology defined in section II.2.2.

Region 2: Drywell

Region 2 is the drywell. All source releases from the RCS fuel are directed into region 2. Radionuclide removal in this region is allowed via natural and active removal mechanisms. This is accomplished by providing a radionuclide removal time constant. This time constant is user specified.

Region 3: Wetwell

Region 3 is the BWR wetwell (torus). All flow from the drywell that passes into the wetwell must pass through vent pipes. When the vacuum breakers on the vent pipes are closed, the gas mixture driven from the drywell into the wetwell will exit at a submerged elevation within the suppression pool. Fission products traversing this pass will be scrubbed (decontaminated) prior to entering the wetwell air space.

When the vent pipe vacuum breakers are open, the gas space of the wetwell and drywell communicate directly, without further scrubbing.

Fission product removal in the wetwell air space is considered via a user input table of radionuclide removal time constant.

Region 4: Control Room

Region 4 is the Control Room. The control room receives air intake from the environment. Both filtered and unfiltered air enters the control room.

The activity entering the control room originates from:

reactor building and wetwell releases through the stack
stack room releases
drywell releases
main condenser releases due to leakages through the main steam isolation valve

Region 5: Reactor Building

Region 5 is the reactor building. In this model the reactor building accepts containment leakage from the drywell and wetwell air space. ESF leakage is not modeled in the TVADOSE model. This is consistent with the guidelines of Reference 4.

Region 6: Stack Room

Region 6 is room at the base of the stack. Leakage may enter this room via filtered leakage originating in the wetwell and the reactor building. Leakage from the stack room is not filtered.

Region 7 and 8:Main Steam Line and Main Condenser

Node 7 and 8 are only weakly coupled to the remainder of the model. Flow leaving the Drywell (Region 2) through the MSIVs enters the main steam line volume (Region 7). The activity of this flow is decremented by the decontamination factor associated with pipe settling, plateout and natural deposition processes.

Region 8 is the Main Condenser. Fission products enter the main condenser at a low rate and are diluted by the large air volume of the condenser. Settling of the fission products within the condenser is modeled with a user input radionuclide removal time constant.

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Introduction of the source into the containment drywell is based on the Revised Source Term and is provided as input into this calculation via Reference 3. This information is summarized below.

Source Term (S(I,i,k)) (fraction of initial inventory released to the drywell) (Ref. 3)					
	TIME(SEC)				
	0 TO 1830	1830-7230			
Noble Gases	.05	0.95			
Iodine	.05	0.25			
Cesium	.05	0.20			
Te-132	0.0	0.05			
other	0.0	0.01			

III. ASSUMPTIONS / INPUT

A list of assumptions follows:

1. ESF leakage is not considered

   Per scope of project as defined by Polestar Technology ., Inc. and TVA (Reference 4).

2. Source term composition and release characteristics are based on NUREG-1465 Revised Source Term. This information is provided in Reference 4.

   This is consistent with the intent of the calculations. Releases into containment are assumed linear over the appropriate time interval.
   All releases are delayed until 30 seconds.

3. Dose Conversion Factors based on specifications supplied in the Polestar data base. (Reference 4)

4.. Tellurium is considered to behave as elemental iodine for purposes of scrubbing. Tellurium doses are based on I-132 DCFs (See workscope, Reference 4))

5. Atmospheric dispersion from regions 3 and 6 are assumed equal. No impact on results since region 3 releases do not contribute to dose calculations

6. Kr-90 contribution is neglected due to its short half life (see References 3 and 6). This is accomplished by setting DCFs for Kr-90 equal to zero.

7. All filters neglect removal due to charcoal filters. Elemental and organic Iodine removal in HEPA filters assumed to have a zero efficiency (References 3 and 4)

8. No fission product removal due to settling or plateout is assumed in the Main condenser.

   This is a conservative assumption in that increased airborne activities implies increased leakage.

9. Fission product removal in the main steam line allowed for elemental iodine and particulates (See References 3 and 4)

10. radionuclides are assumed to instantaneously mix with volume atmosphere. This assumption is consistent with standard review plan methodology and the current workscope (Reference 4)

11. Fumigation time interval is selected at the worst one half hour period over the first two hours. This occurs in the 1.5 to 2 hour time frame. Including this effect later in the event acknowledges the impact of the later release of radionuclides. Doses will be maximized with this assumption since the later interval has the higher atmospheric releases.

IV. CALCULATION

IV.1 Radionuclide and Dose Data

Data for Radionuclide Activity, decay constants and Dose Conversion Factors are obtained from Reference 3, Section 1.

	ISOTOPE	LDOSE NAME	ACTIVITY	DECAY LAMBDA	WB DCF	BETA DCF (skin)	THYROID DCF
			CI	DIS /SEC	Rem-m^3/(Ci-S)	Rem-m^3/(Ci-S)	10^4/ci*
IS		ISNAM(IS)	AOREF(IS)	YD(IS)	DCFWB2(IS)	DCFSK2(IS)	DCFTH2(IS)
10	Kr-83m	KRYPTO	1.127E+07	1.04E-04	1.27E-05	0	0
11	Kr-85m	KRYPTO	2.351E+07	4.39E-05	2.30E-02	4.97E-02	0
12	Kr-85	KRYPTO	1.360E+06	2.04E-09	3.31E-04	4.84E-02	0
13	Kr-87	KRYPTO	4.481E+07	1.52E-04	1.33E-01	3.36E-01	0
14	Kr-88	KRYPTO	6.303E+07	6.89E-05	3.38E-01	7.76E-02	0
15	Kr-89	KRYPTO	7.653E+07	3.63E-03	3.03E-01	3.47E-01	0
16	Kr-90**	KRYPTO	7.554E+07	.215E-1	0.0	0.0	0
17	Xe-131m	XENON	1.050E+06	6.68E-07	1.25E-03	1.33E-02	0
18	Xe-133m	XENON	5.960E+06	3.49E-06	4.29E-03	2.96E-02	0
19	xe-133	XENON	1.847E+08	1.52E-06	4.96E-03	9.67E-03	0
20	Xe-135m	XENON	3.761E+07	7.40E-04	6.37E-02	2.14E-02	0
21	Xe-135	XENON	6.610E+07	2.09E-05	3.59E-02	6.32E-02	0
22	Xe-137	XENON	1.655E+08	2.96E-03	2.83E-02	4.59E-01	0
23	Xe-138	XENON	1.552E+08	6.80E-04	1.87E-01	1.47E-01	0
1	I-131	IODINE	9.378E+07	9.96E-07	5.59E-02	3.07E-02	110
2	I-132	IODINE	1.355E+08	8.27E-05	3.55E-01	1.10E-01	0.63
3	I-133	IODINE	1.898E+08	9.22E-06	9.11E-02	8.90E-02	18
4	I-134	IODINE	2.081E+08	2.23E-04	4.11E-01	1.42E-01	0.11
5	I-135	IODINE	1.778E+08	2.86E-05	2.49E-01	7.86E-02	3.1
6	Cs-134	CESIUM	2.508E+07	9.55E-09	2.58E-01	1.15E-01	0
7	Cs-137	CESIUM	1.503E+07	7.29E-10	9.30E-02	1.27E-01	0
8	Te-132	TELLUR	1.333E+08	2.51E-06	3.55E-01	1.10E-01	0.63
9	Other	OTHER	4.967E+9	7.05E-5	.168	0	0

* INHALED

** DCF set = 0.00 per ref. 6

The following factors are not used in the TVADOSE calculation: DCFTH1, DCFSK1, DCFWB1, EB and EG these parameters appear in the database but are not used in TVADOSE.

LOCATOR CONSTANTS:

I131- "IS" SUBCRIPT FOR ENTRY FOR I-131 : 1
NID - TOTAL NUMBER OF IODINE ISOTOPES: 5
KRN -TOTAL NUMBER OF KRYPTON ISOTOPES: 7
IST -TOTAL NUMBER OF ISOTOPES: 23
NOI-TOTAL NUMBER OF OTHER ISOTOPES: 1
NCS-TOTAL NUMBER OF CESIUM ISOTOPES:2
NXE-TOTLA NUMBER OF XENON ISOTOPES :7

IV.2 Revised Source Term Release Profile

Data obtained from Reference 3 items 2.1, 2.2, 2.3.

a) Fractional Releases (REVISED SOURCE TERM)

Fractional Releases into Containment in time Interval

Time Interval (sec)	Noble Gases	Iodine	Cesium	Tellurium	Other
0 to 30	0	0	0	0	0
30 to 1830	0.05	0.05	0.05	0	
1830-7230	0.95	0.25	0.20	0.05	0.01
7230 to End	0.0	0.0	0.0	0.0	0.0

Note that the Iodine contribution includes aerosols (CsI) + elemental + organic iodine

From Reference 5 item 2.2 and 2.3

total iodine between 30 to 1830 sec = .0024 + .000075 + .0475 =.049975 (round up to .05)

Ratio of CSI/Total = .0475/.05= 0.95
Ratio of I2(elemental) / Total = 0.048
Ratio of Organic I/Total = .000075/.05 =.0015
Note the ratios do not add to 1. Therefore,

Fraction CsI = 0.95
Fraction I2(gas)=.0485
Fraction Organic=.0015

This composition applies to both early and late releases

b) Iodine Composition

From (a) above

FRCTK:

Particulate Iodine: 0.95
Elemental Iodine : 0.0485
Organic Iodine : 0.0015

The code assumes the following:

All aerosols are particulate in nature and can be filtered via particulate filters

All noble gases cannot be filtered

Tellurium is an aerosol that is assumed to not be filtered via filter flowpaths (treated like elemental iodine)

IV.3 Browns Ferry Model System Description

a) Volumes (Data taken from Reference 3)

NODE NO	TVADOSE var.	ITEM NO Ref. 3	DESCRIPTION	VOLUME CUBIC FT	COMMENT
1			ENVIRONMENT	1.00E+08	ARBITRARY VV(1) NOT USED
2	VV(2)	3.1	DRYWELL	159000	ITEM 3.1
3	VV(3)	3.2	WETWELL	124000	ITEM 3.2
4	VV(4)	3.6	CONTROL ROOM	210000	ITEM 3.6
5	VV(5)	3.4	REACTOR BLDG	1.93E+06	ITEM 3.4
6	VV(6)	3.5	STACK ROOM	34560	ITEM 3.5
7	VV(7)	3.7	MS PIPE	692	ITEM 3.7
8	VV(8)	3.8	MAIN CONDENSER	122400	ITEM 3.8

IV.4 REGION FLOWRATES

TVADOSE ALLOWS FIXED AND VARIABLE FLOWRATES. THE FLOWRATES THAT ARE FIXED ARE:

L25, L35 L31U, L81,L51, L61, L56, L78, L14U, L14F

FLOWRATES THAT VARY WITH TIME INCLUDE:

L21, L23, L31F, L32, L27

ALL LEAKAGES ARE INPUT TO THE CODE AS A PARAMETER WITH GG REPLACING L.

SUMMARY OF FIXED FLOWRATES			
VARIABLE NAME	DESCRIPTION	VALUE (CFH)	REF 3, ITEM
GG14U	CR UNFILTERED INFLOW	2.23E+5	3.22
GG14F	CR FILTERED INFLOW	1.8E+5	3.21
GG25	DW TO RB LEAKAGE	132.5	3.12
GG31U	WW TO STACK THROUGH HARDENED VENT	10	3.16
GG61	STACK ROOM TO ENVIRONMENT	300	3.19
GG35	WETWELL LEAKAGE TO RB	103.3	3.13
GG51 CASE 1	RB FLOW TO SGTS FILTER TO STACK	1.32E+6	3.14
GG51 CASE 2	RB FLOW TO SGTS FILTER TO STACK	0.9E+6	3.14
GG56	FLOW FROM SGTS FILTER TO STACK ROOM	300	3.15
GG78	MAIN STEAM LINE TO CONDENSER	475	3.24
GG81	LEAKAGE FROM MAIN CONDENSER	250	3.25

IV.4 (CONT'D)

TIME VARYING FLOWS:

FLOW GG21: FLOW FROM DRYWELL TO ENVIRONMENT: UNFILTERED (ITEM 3.20 OF Ref 3)

GG21 = 0 for all time for case 1.

TIME(sec)	FLOWRATE (CFH)
0.0	3.1E-3 (case 2)
105	0.0
3.E+6	0.0

FLOW GG23 AND GG32 MIXING AND TRANSPORT FLOWS WITHIN THE CONTAINMENT (ITEMS 3.10 AND 3.11)

TIME (SEC)	GG23:DW-WW	GG32:WW-DW
	CFH	CFH
0.0	0.0	0.0
1830.	1.6E+5	0.0
7230	1.2E+6	0.0
7890	1.2E+6	1.2E+6
3.E+6	1.2E+6	1.2E+6

NOTE 30 DAYS = 2.592E+6 SECONDS (CASE RUNS TO 30 DAYS)

GG27: LEAKAGE FROM DRYWELL TO MAIN STEAM LINE (3.23)

TIME(SEC)	GG27 (CFH)
0.0	120
7230	177.5
3.E+6	177.5

GG31F: FILTERED FLOE FROM WW THROUGH CAD (ITEM3.17)

TIME (DAYS)	GG31F (CFH)
0	0.0
10	8340
11	0.0
20	8340.
21	0.0
29	8340
30	0.0

SET FLOW BEYOND 30 DAYS =0. (NOT USED IN CALC)

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B) REMOVAL COEFFICIENTS:

Removal coefficients are provided for the drywell, wetwell gas space and the main condenser gas space. See items 4.3 and 4.4 of Reference 3.

TIME (SEC)	LAMBDA DW		TIME (SEC)	LAMBDA WW	
	(PER HOUR)	REF. ITEM 4.3		(PER HOUR)	REF ITEM 4.4
0	0		0	0	
30	.35		7890	.95	
2400	.45		8570	.85	
3200	.55		9840	.75	
4000	.65		11760	.65	
4885	.75		14530	.55	
6300	.85		18650	.45	
7360	.95		24980	.35	
8570	.85		35570	.25	
9840	.75		57220	.162	
11760	.65		100000	0	
14530	.55		3.E+6	0	
18650	.45				
24980	.35				
35570	.25				
57220	.162				
10000	0.0				
3. E + 6	0.0				

IV.6 Atmospheric Dispersion

CHI/Q Values for Various Regions (SEC/M3)					
node 5	STACK RELEASE				
XQ5..	ITEM 5.1	TIME INT(XQ5T) (hrs)	EAB XQ5EB	LPZ XQ5LP	CR XQ5CR
	1	0 TO 1.5	9.70E-07	8.00E-07	5.91E-15
	2	1.5 TO 2	2.40E-05	1.300E-05	3.31E- 5
	3	2 TO 8		8.00E-07	3.80E-15
	4	8 TO 24		4.00E-07	3.00E-15
	5	24 TO 96		2.00E-07	1.90E-15
	6	96 TO 720		6.50E-08	9.60E-16
node 6	STACK ROOM RELEASE				
XQ6..	ITEM 5.2	XQ6T(hrs)	EAB- XQ6EB	LPZ XQ6LP	CR XQ6CR
	1	0 TO 2	1.22E-04	5.65E-05	8.89E-04
	2	2 TO 8		5.65E-05	7.30E-04
	3	8 TO 24		2.24E-05	6.60E-04
	4	24 TO 96		7.94E-06	5.40E-04
	5	96 TO 720		1.71E-06	4.00E-04
node 8	MAIN CONDENSER RELEASE				
XQ8..	ITEM	end time interval			
	ITEM 5.3	XQ8T(hrs)	EAB XQ8EB	LPZ XQ8LP	CR XQ8CR
	1	0 TO 2	2.70E-04	1.32E-04	1.74E-04
	2	2 TO 8		6.02E-05	1.47E-04
	3	8 TO 24		4.07E-05	1.27E-04
	4	24 TO 96		1.73E-05	1.01E-04
	5	96 TO 720		5.10E-06	7.20E-05
node 2	DRYWELL RELEASE				
XQ2..	5.4				
	1	2	0	0	1.12E-03

XQ2 ONLY IMPACTS RELEASE FOR DURATION OF GG21 RELEASE

IV.7 Breathing Rates and Occupancy Factors

(Data From Ref. 3)

a. breathing rates		ITEM 5.5
	end time	rate
	(hr)	m3/sec
	8	3.47E-04
	24	1.75E-04
	720	2.32E-04

b. occupancy factors			ITEM 5.6
	end time		factor
	(hrs)		
	24		1
	96		0.6
	720		0.4

V. CODE USED / UPDATES

The calculation employs the TVADOSE computer code (See Reference 2) version TVD92395.

VI. COMPUTER RUN SUMMARY

Case 1 refers to the base analysis as defined in Section IV. Case 2 is similar to case 1 except that per Reference 3:

GG21 = 3.1E-3 CFH for 105 seconds

and

GG51 = 0.9E+6 CFH

COMPUTER OUTPUT LIST

Case # or Name	JOB ID	Run Date/Time Cd Rom Date	Description
1 or BFCASE1	see below	Run date 9/23/95 Run time 11:22:57 Cd Rom date 9/25/95	HIGH RB FLOW; OUTPUT: BFCASE1.OUT
2 or BFCASE2	see below	Run date 9/23/95 Run time 11:25:40 Cd Rom date 9/25/95	LOW RB FLOW OUTPUT; BFCASE2.OUT

	Filename:	Job Id:	Description:
	bfcase1.inp	0u5ha26c.cdf	Input file
	bfcase1.out	0u5hab88.cdf	Standard output file
*	bfcase1.act	0u5h980o.cdf	Regional activity file
	bfcase1.dpr	0u5h9s4g.cdf	Detailed debug edits
	bfcase2.inp	0u5hcdho.cdf	Input file
	bfcase2.out	0u5hcsjk.cdf	Standard output file
*	bfcase2.act	0u5hb8c0.cdf	Regional activity file
	bfcase2.dpr	0u5hclfs.cdf	Detailed debug edits

Computer output stored in directory /misc/6602r00/out on the CD-ROM volume identified on the cover sheet. Output files are also included in Appendices.

* Note, in the file which contains the activity information, the following heading is used:

```
 ACTIVITIES (CI) -- REGION NUMBERS IN ( )

DW (2)      WW (3)      RR (4)      RB (5)      SR (6)      MSL (7)      MC (4)
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In this heading "RR (4)" should be interpreted to mean the Control Room (Region 4) and "MC (4)" should actually be indicating Main Condenser (Region 8). This code anomaly should not cause any significant problems

VII. RESULTS AND CONCLUSIONS

The criteria for control room habitability given in NUREG-0800, Standard Review Plan 6.4, Rev. 2, 1981 is as follows:

limit for 30 day dose accumulation:

5 Rem whole body
30 Rem Thyroid (iodine inhalation)
30 Rem skin

The NRC allowable offsite doses are given in 10CFR100.11 to be:

25 Rem total whole body
300 Rem total to the thyroid due to iodine exposure.

These limits apply to EAB 2 hour and LPZ thirty day doses.

The results of the two cases are summarized in the table below: The maximum thyroid dose in the CR is under 18 rem. Dose summaries can be found on pages A-56 thru A-58 and B-56 thru B-58 of Appendices.

LOCATION	DOSE TYPE	CASE 1		CASE 2	
		2 HR DOSE (REM)	30 DAY DOSE (REM)	2 HR DOSE (REM)	30 DAY DOSE (REM)
CR	THYROID		17.9		17.41
CR	SKIN		1.794		1.782
CR	WHOLE BODY		0.046		0.045
EAB	THYROID	3.159		2.738	
EAB	SKIN	0.05658		0.0441	
EAB	WHOLE BODY	0.0750		0.059	
LPZ	THYROID		5.786		5.552
LPZ	SKIN		0.4928		0.4826
LPZ	WHOLE BODY		0.2823		0.269

The contribution of "other" isotopes to the whole body LPZ and EAB doses are as follows:

Case 1: EAB 2 hour : other contribution = .000527 Rem
Case 1: LPZ 30 day: other contribution = .00034 Rem
Case 2: EAB 2 hour : other contribution = .00050 Rem
Case 2: LPZ 30 day: other contribution = .00033 Rem

For case 1 Iodine -131 contributes 16.64 rem to the control room thyroid dose (30 day), distributed among the three iodine forms as follows:

elemental : 3.21 Rem
organic: 12.51 Rem
Particulate:0.9232 Rem

Additional details associated with the radionuclide contributions to the dose can be found in the computer output files (See Appendices A and B). Also included in the appendices are the predicted plant activities for various times into the event.

VIII. REFERENCES

1. TML-90-120, "Analytical Models for LOCA Radiological Dose Consequences (Basis for LDOSE Program)", S. Rosen, November, 1, 1990.(Combustion Engineering Proprietary)

2. 1066-S&T95-C-001 TVADOSE:COMPUTER PROGRAM FOR THE CALCULATION OF BROWNS FERRY ADVANCED SOURCE TERM LEAKAGE DOSES, M. Michonski, September.29, 1995, (Combustion Engineering Proprietary)

3. PSAT-04000U.03,Rev. 1, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", J. Metcalf, September 22, 1995

4. Attachment A "Workscope" to Letter L. Brown-Herzl (Polstar Applied Technology, Inc) to Raymond Schneider (Combustion Engineering, Inc.) August 31, 1995.

5. PVNGS UPDATED FSAR (APPENDIX 15B)

6. Letter from J. Metcalf (PSAT) to Ray Schneider (ABB), dated September 20, 1995

7. NUREG-1465, Accident Source Terms for Light Water Power Plants, January 1995, USNRC

IX. APPENDICES

APPENDIX A

PROPRIETARY

Appendix C

Browns Ferry Data Base

(12 pages in data base)

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PSAT 04000U.03

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Revision: 1

DESIGN DATA BASE FOR
APPLICATION OF THE REVISED DBA SOURCE TERM
TO THE TVA BROWNS FERRY NUCLEAR POWER PLANT

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	PROJECT MGR		REVIEWER		TVA TECH CONT	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REV: 0	James Metcalf /s	9/1/95	Dave Leaver /s By fax direction (on file)	9/1/95	Don McCamy /s By fax direction (on file)	9/1/95
Reason for Revision: Initial Issue						
REV: 1	James Metcalf /s	9/22/95	Dave Leaver /s	9/22/95	Don McCamy /s	9 /22/95

Reasons for Revision:

1. General - Added revision numbers to PSAT calc references.

2. Item 1 - Added NUREG reference.

3. Item 2 - Finalized reference and removed exception (based on deletion of Item 2.4, see 5 below).

4. Item 2.3 - Corrected typo in note and added clarification that "Other" could be included in dose calculation of record if contribution negligible.

5. Item 2.4 - Deleted because power purge is not being used to model Case 2 SGTS bypass.

6. Item 3.7 - Provided steamline/drainline volume and reference.

7. Item 3.8 - Changed MC volume to be slightly more conservative.

8. Item 3.20 - Finalized reference, changed value, and removed option for flow from torus (not needed because Case 2 SGTS bypass is being modeled using a specific flowpath from drywell - see 5 above).

PROJECT MGR		REVIEWER		TVA TECH CONT	
Print/Sign	Date	Print/Sign	Date	Print/Sign	Date

REV: 1

Reasons for Revision (continued)

9. Item 3.23 - Provided "two-step" drywell leakrates to steamline and clarified reference.

10. Items 3.24 and 3.25 - Provided values for steamline and MC volumetric flows and references.

11. Items 3.27 and 3.28 - Added "free" for clarification.

12. Item 3.29 - Added "per line" MSIV test limit.

13. Item 3.30 - Added RB non-SGTS volumetric exchange when RB pressure positive.

14. Item 4.1 - Clarified "bid" spec.

15. Items 4.3 and 4.4 - Added clarification that values are medians over cited intervals.

16. Items 4.7 and 4.8 - Changed "DF" to "filter efficiency", provided values and references for Item 4.7, and deleted Item 4.8 because Item 4.7 particulate removal is so high that Item 4.8 not needed.

17. Item 5.1 - Made fumigation X/Qs 1.5 to 2 hours instead of 0 to 0.5 hours.

18. Item 5.4 - Changed reference to TVA calc.

19. Items 7.3, 7.4, and 7.5 - Added steamline ID, effective steamline length, and approximate containment dimensions.

20. Item 8.9 - Added use: steamline initial temperature determination.

21. Item 8.15 - Added site-specific "standard conditions" reference pressure for converting drywell to steamline volumetric flow.

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1. Radionuclide Data (References - Ci Inventories: TVA Bid Spec, except Cs and Te from GE Letter BFSE 95-023 and "Other" is PSAT Calc 04011H.05, Rev 0; Decay Constants and DCFs: TACT5 data file MLWRICRP.30 from NUREG/CR-5106 except Kr-90, Cs-134, Cs-137 and "Other" are PSAT Calc 04011H.05, Rev 0)

Nuclide	Core Inventory @ t=0 10^7Ci	(per sec) DKlambda	(10^{-2} Rem-m^3/Ci-sec) WB DCF	Skin DCF	(10^4 Rem/Ci) Thyroid DCF
Kr-83m	1.127	1.04E-04	0.00127	0.0	0.0
Kr-85m	2.351	4.39E-05	2.3	4.97	0.0
Kr-85	0.136	2.04E-09	0.0331	4.84	0.0
Kr-87	4.481	1.52E-04	13.3	33.6	0.0
Kr-88	6.303	6.89E-05	33.8	7.76	0.0
Kr-89	7.653	3.63E-03	30.3	34.7	0.0
Kr-90	7.554	-	-	-	-
Xe-131m	0.105	6.68E-07	0.125	1.33	0.0
Xe-133m	0.596	3.49E-06	0.429	2.96	0.0
Xe-133	18.47	1.52E-06	0.496	0.967	0.0
Xe-135m	3.761	7.40E-04	6.37	2.14	0.0
Xe-135	6.610	2.09E-05	3.59	6.32	0.0
Xe-137	16.55	2.96E-03	2.83	45.9	0.0
Xe-138	15.52	6.80E-04	18.7	14.7	0.0
I-131	9.378	9.96E-07	5.59	3.07	110.0
I-132	13.55	8.27E-05	35.5	11.0	0.63
I-133	18.98	9.22E-06	9.11	8.90	18.0
I-134	20.81	2.23E-04	41.1	14.2	0.11
I-135	17.78	2.86E-05	24.9	7.86	3.1
Cs-134	2.508	9.55E-09	25.8	11.5	0.0
Cs-137	1.503	7.29E-10	9.3	12.7	0.0
Te-132	13.33	2.51E-06	35.5	11.0	0.63
Other	496.7	7.05E-05	16.8	0.0	0.0

2. Source Terms (Reference Calc PSAT 04011H.06, Rev 0)

2.1 Fraction of core inventory, 0 - 30 seconds: no releases

2.2 Fraction of core inventory, 30 - 1830 seconds: Gases -
Xe, Kr - 2.8E-5 /sec (0.05 total)
Elemental I - 1.3E-6 /sec (2.4E-3 total)
Organic I - 4.2E-8 /sec (7.5E-5 total)

Aerosols -
Iodine - 2.6E-5 /sec (0.0475 total)
Cesium - 2.8E-5 /sec (0.05 total)

2.3 Fraction of core inventory, 1830 - 7230 seconds: Gases - Xe, Kr - 1.8E-4 /sec (0.95 total)

Elemental I - 2.2E-6 /sec (1.2E-2 total)
Organic I - 6.9E-8 /sec (3.8E-4 total)

Aerosols - Iodine - 4.4E-5 /sec (0.2375 total)
Cesium - 3.7E-5 /sec (0.2 total)
Tellurium - 9.3E-6 /sec (0.05 total)
Other - 1.9E-6 /sec (0.01 total)

Note that the Other is specified only to verify the assumption that its inclusion in the dose calculation will have little or no impact - should not be included in calculations of record unless contribution is negligible.

2.4 - DELETED

3. Volumes and Volumetric Flowrates

3.1 Volume of Drywell - 159000 ft^3	(Reference NEDO-24580, Rev 2, Table BF 4.1.1-1)
3.2 Volume of Torus Airspace - 124000 ft^3	(Reference TVA Bid Spec total containment volume of 283000 ft^3 less drywell volume)
3.3 Volume of Suppression Pool - 127800 ft^3	(Reference BFNP UFSAR Table 5.2-1, Amendment 9 and given as 123000-128000 ft^3 in TVA Calc ND-Q0999-880163, Rev 1)
3.4 Volume of Reactor Building (RB) - 1.932E6 ft^3	(Reference TVA Bid Spec)
3.5 Volume of Stack Room (SR) - 34560 ft^3	(Reference TVA Bid Spec - 1/2 volume of 69120 ft^3 used to account for 50% mixing)
3.6 Volume of Control Room (CR) - 210000 ft^3	(Reference TVA Bid Spec)
3.7 Volume of Main Steamlines and Associated Drainlines - 692 ft^3	(Steamlines only used - 692 = 4 steamlines x 68.25 ft length x $\{21.562" \text{ diameter}\}^2$ x $\pi/4/144$, see Items 7.3 and 7.4)
3.8 Volume of Main Condenser (MC) - 122400 ft^3	(Reference TVA Bid Spec - for conservatism only 90% of condenser volume credited)
3.9 Volume of Water in the Hotwell - 25400 ft^3	(Reference TVA Bid Spec as 190000 gallons)

3.10 Volumetric Flowrate, Drywell to Torus (Filtered): (Reference Calc PSAT 04011H.01, Rev 0)

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3.11 Volumetric Flowrate, Torus to Drywell (Unfiltered): (Reference Calc PSAT 04011H.01, Rev 0)

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3.12 Volumetric Flowrate, Drywell to RB - 132.5 cfh (Reference TVA Bid Spec total primary containment leakrate of 235.8 cfh apportioned between drywell and torus volumes)

3.13 Volumetric Flowrate, Torus to RB - 103.3 cfh (Reference TVA Bid Spec total primary containment leakrate of 235.8 cfh apportioned between drywell and torus volumes)

3.14 Volumetric Flowrate, RB to Stack (Filtered) - 1.32E6 cfh for Case 1 (Reference TVA Bid Spec)
- 9.0E5 cfh for Case 2

3.15 Volumetric Flowrate, RB to SR (Filtered) - 300 cfh (Reference TVA Bid Spec)

3.16 Volumetric Flowrate, Torus to Stack (Unfiltered) - 10 cfh (Reference TVA Bid Spec)

3.17 Volumetric Flowrate, Torus to Stack (Filtered): (Reference TVA Bid Spec for CAD flow*)

From t=0 to t=10 days - 0 cfh
From t=10 to t=11 days - 8340 cfh
From t=11 to t=20 days - 0 cfh
From t=20 to t=21 days - 8340 cfh
From t=21 to t=29 days - 0 cfh
From t=29 to t=30 days - 8340 cfh

*CAD flow not specified to be from torus, but primary containment is well mixed at this time

3.18 Volumetric Flowrate, Torus to SR (Filtered): 0.033% of the CAD flowrates given above (Reference TVA Bid Spec for CAD flow**)

**CAD flow bypassing the stack not given, but assumed to bypass with same fraction as SGTS flow = 300 cfh/9.0E5 cfh = 0.00033 max

3.19 Volumetric Flowrate, SR to Environment - 300 cfh (Reference TVA Bid Spec - assumed to be same as leakage into SR from stack)

3.20 Volumetric Flowrate, Drywell to Environment Via Reactor Building (Reference Calc PSAT

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04011H.07, Rev 0)

(Unfiltered - Case 2 CR Only)
From t=0 to t=105 seconds - 3.1E-3 cfh
From t=105 seconds to end - 0 cfh

3.21 Volumetric Flowrate, Environment to CR (Filtered) - 1.8E5 cfh (Reference TVA Bid Spec as 3000 cfm)

3.22 Volumetric Flowrate, Environment to CR (Unfiltered) - 2.23E5 cfh (Reference TVA Bid Spec as 3717 cfm)

3.23 Volumetric Flowrate, Drywell to Main Steamlines/Drainlines
- 120 cfh from t=0 to t=7230 seconds
- 177.5 cfh from t=7230 seconds to end

(Reference Calc PSAT 04001H.02, Rev 0***)
***Rates developed in this calculation are for one-steam line at 100 scfh. These rates are multiplied by 2.5 to obtain the values given in this data base for total flow. See Item 3.28.)

3.24 Volumetric Flowrate, Main Steamlines/Drainlines to MC - 475 cfh
(Reference 250 scfh from Item 3.28 corrected for reference temperature in steamlines versus standard conditions from Calc PSAT 04002H.08, Rev 0; i.e., 250 x {558.5 K x 9/5)/529.67 R}= 475)

3.25 Volumetric Flowrate, MC to Environment - 250 cfh (Based on Item 3.28 with observation that flow to main condenser through drain lines will be at near-standard temperature after approximately 6 hours - see Exhibit 1 of Calc PSAT 04002H.09, Rev 0)

3.26 Free Volume In-Vessel, below BAF - 4100 ft^3 (Reference TVA Calc MD-Q0063-920470 Rev 0, Attachment A)

3.27 Free Volume In-Vessel, within Core Shroud, BAF-TAF - 1759 ft^3 (Reference TVA Calc MD-Q0063-920470 Rev 0, Attachment A as 5077.7 ft^3 - 4099.7 ft^3 + 486.6 ft^3 + 294.5 ft^3 = 1759.1 ft^3)

3.28 Combined MSIV Tested Leakrates - 250 scfh (Reference TVA Fax McCamy to Metcalf dated 9/1/95)

3.29 Per Line MSIV Tested Leakrate - 100 scfh (Reference TVA Bid Spec)

3.30 Non-SGTS Volumetric Exchange when RB Pressure Positive - 1540 cfh (Reference TVA Calc ND-Q0065-900052, Rev 2)

4. Filter Efficiencies, Removal Lambdas, and Decontamination Factors

4.1 Filter Efficiency - All Filtered Flowpaths Except Drywell to Torus: (Reference Tech Specs 3.7.B and 3.7.E, Amendment 231 and TVA Bid Spec - Charcoal Filters Assumed Removed from SGTS and CREVS)

- For Particulate Iodine, Cesium, and Other - 99%
- For Elemental and Organic Iodine, Tellurium, and Noble Gasses - 0%

4.2 Filter Efficiency - Drywell to Torus: (Reference Calc PSAT 04011H.04, Rev 0)

From t=0 to t=7230 seconds:

- For Particulate and Elemental Iodine, Cesium, Tellurium, and Other - 72%
- For Organic Iodine and Noble Gasses - 0%

From t=7230 to t=7890 seconds:

- For Particulate and Elemental Iodine, Cesium, Tellurium, and Other - 95%
- For Organic Iodine and Noble Gasses - 0%

From t=7890 seconds to end:

- For All Species - 0%

4.3 Removal (Sedimentation) Lambdas in Drywell: (Reference Calc PSAT 04001H.02, Rev 0 - values are medians over cited intervals)

For Particulate and Elemental Iodine, Cesium, Tellurium, and Other:

- From t=0 to t=30 seconds - 0/hour
- From t=30 to t=2400 seconds - 0.35/hour
- From t=2400 to t=3200 seconds - 0.45/hour
- From t=3200 to t=4000 seconds - 0.55/hour
- From t=4000 to t=4885 seconds - 0.65/hour
- From t=4885 to t=6300 seconds - 0.75/hour
- From t=6300 to t=7360 seconds - 0.85/hour
- From t=7360 to t=8570 seconds - 0.95/hour
- From t=8570 to t=9840 seconds - 0.85/hour
- From t=9840 to t=11760 seconds - 0.75/hour
- From t=11760 to t=14530 seconds - 0.65/hour
- From t=14530 to t=18650 seconds - 0.55/hour
- From t=18650 to t=24980 seconds - 0.45/hour
- From t=24980 to t=35570 seconds - 0.35/hour
- From t=35570 to t=57220 seconds - 0.25/hour

- From t=57220 to t=100000 seconds - 0.162/hour
- From t=100000 seconds to end - 0/hour

For Organic Iodine and Noble Gasses

- From t=0 to end - 0/hour

4.4 Removal (Sedimentation) Lambdas in Torus: (Reference Calc PSAT 04001H.02, Rev 0 - values are medians over cited intervals)

For Particulate and Elemental Iodine, Cesium, Tellurium, and Other:

- From t=0 to t=7890 seconds - 0/hour
- From t=7890 to t=8570 seconds - 0.95/hour
- From t=8570 to t=9840 seconds - 0.85/hour
- From t=9840 to t=11760 seconds - 0.75/hour
- From t=11760 to t=14530 seconds - 0.65/hour
- From t=14530 to t=18650 seconds - 0.55/hour
- From t=18650 to t=24980 seconds - 0.45/hour
- From t=24980 to t=35570 seconds - 0.35/hour
- From t=35570 to t=57220 seconds - 0.25/hour
- From t=57220 to t=100000 seconds - 0.162/hour
- From t=100000 seconds to end - 0/hour

For Organic Iodine and Noble Gasses

- From t=0 to end - 0/hour

4.5 Maximum DF for Elemental Iodine in Drywell - 8000 (Reference Calc PSAT 04011H.03, Rev 0)

4.6 Maximum DF for Elemental Iodine in Torus - 8000 (Reference Calc PSAT 04011H.03, Rev 0)

4.7 Filter Efficiency for Flowpath From Drywell to Main Steamlines/Drainlines:
(Reference Calcs PSAT 04002H.08, Rev 0 and .09, Rev 0)

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4.8 - DELETED

5. X/Q Values, Breathing Rates, and Occupancy Factors

	EAB	LPZ	CR
5.1 Stack Release X/Q (sec/m^3):	(Reference TVA Bid Spec - data presented as "Top of Stack" - modified by telecon with TVA's Tech Contact Don McCamy 9/13/95)		

From t=0 to t=1.5 hours	9.7E-7	8.0E-7	5.9E-15
From t=1.5 to 2 hours	2.4E-5	1.3E-5	3.31E-5
From t=2 to t=8 hours		8.0E-7	3.8E-15
From t=8 to t=24 hours		4.0E-7	3.0E-15
From t=24 to t=96 hours		2.0E-7	1.9E-15
From t=96 to t=720 hours		6.5E-8	9.6E-16

5.2 SR Release X/Q (sec/m³): (Reference TVA Bid Spec - data presented as "Base of Stack")

From t=0 to t=2 hours	1.22E-4	5.65E-5	8.89E-4
From t=2 to t=8 hours		5.65E-5	7.30E-4
From t=8 to t=24 hours		2.24E-5	6.60E-4
From t=24 to t=96 hours		7.94E-6	5.40E-4
From t=96 to t=720 hours		1.71E-6	4.00E-4

5.3 MC Release X/Q (sec/m³) (Reference TVA Bid Spec - data presented as "Turbine Building" - CR X/Qs decreased by a factor of two to account for intakes being on either side of Turbine Building as described in TVA Bid Spec)

From t=0 to t=2 hours	2.70E-4	1.32E-4	1.74E-4
From t=2 to t=8 hours		6.02E-5	1.47E-4
From t=8 to t=24 hours		4.07E-5	1.27E-4
From t=24 to t=96 hours		1.73E-5	1.01E-4
From t=96 to t=720 hours		5.10E-6	7.20E-5

5.4 RB Ground-Level Release X/Q (sec/m³)**** - 1.12E-3 (Reference TVA Calc ND-Q0065-900052, Rev 2)

****From t=0 to t=105 seconds, Case 2 CR Only

5.5 Breathing rates: (Reference TVA Bid Spec)

0 - 8 hour - 3.47E-4 m³/sec
8 - 24 hour - 1.75E-4 m³/sec
24 - 720 hour - 2.32E-4 m³/sec

5.6 Occupancy Factors: (Reference TVA Bid Spec)

From t=0 to t=1 day - 1.0
From t=1 to t=4 days - 0.6
From t=4 to t=30 days - 0.4

6. Chemistry Data

6.1 Initial Pool pH - 6.0 (Reference TVA Memo Ricketts to McCamy dated 8/30/95 stating range 6.0 - 6.3)

6.2 Initial Hotwell pH - 7.0 (Reference TVA Memo Ricketts to McCamy dated 8/30/95 stating range 7.0 - 7.7)

6.3 Mass of Chloride-Bearing Cable Insulation in Containment (Reference TVA QIR BFEBFN95038)

- Hypalon - 333 lbm
- PVC - 3472 lbm

6.4 Thickness of Cable Insulation - less than 80 mils (Reference preliminary - will be TVA Letter ...)

6.5 Mass of B-10 Needed for Reactor Shutdown - 186 lbm (Reference TVA Calc MD-Q0063-920470, Rev 0)

6.6 Mass B-10 per Lbm of Enriched Sodium Pentaborate - 0.108 lbm (Reference TVA Calc MD-Q0063-920470, Rev 0)

6.7 Formula of ESPB - $Na_2O\text{-}5B_2O_3\text{-}10H_2O$ (Reference TVA Calc MD-Q0063-920470, Rev 0)

6.8 Formula Weight of ESPB - 585.9 lbm/lbm-mole (Reference TVA Calc MD-Q0063-920470, Rev 0)

6.9 Reactor Coolant Mass Excluding Suppression Pool - 1.22E6 lbm (Reference TVA Calc MD-Q0063-920470, Rev 0)

7. Fission Product Transport Data

7.1 Sedimentation Area in Drywell - 8183 ft^2 (Reference TVA Calc ND-Q0999-950021, Rev 0)

7.2 Sedimentation Area in Torus - 13635 ft^2 (Reference TVA Calc ND-Q0999-950021, Rev 0)

- 10859 ft^2 wet
- 2776 ft^2 dry

7.3 Steamline ID - 21.562" (Reference TVA Calc ND-Q0031-920075, Rev 6)

7.4 Length of Steamline from Outboard MSIV to Drain Line Tap - 68.25' (Reference TVA Calc ND-Q0999-950021, Rev 0)

7.5 Approx Containment Dimensions: (Reference TVA Browns Ferry Dwg, General

Plans and Sections, Sheet 2)

- Diameter of Drywell Cylinder - 38.5 feet
- Height of Drywell Cylinder - 55 feet
- Height of Drywell Sphere Sidewall - 50 feet
- Diameter of Drywell Sphere - 67 feet
- Major Torus Diameter - 111.5 feet
- Minor Torus Diameter - 31 feet

8. Thermal-Hydraulic Data

8.1 Drywell State after 30 Seconds, but Prior to Start of Containment Heat Removal (End of Debris Quench)
- 27 psig
- 275 F

(Reference TVA Calc ND-Q0031-920075, Rev 6)

8.2 Drywell State after Start of Containment Heat Removal
- 10 psig
- 175 F

8.3 Core Power - 3458 Mw(t) (Reference TVA Bid Spec)

8.4 Core Spray Flowrate - 6250 gpm (two pumps) (Reference BFNP UFSAR Table 14.6-3, Amendment 7)

8.5 ID of Core Shroud - 203" (Reference TVA Calc MD-Q0063-920470, Rev 0, Attachment A as 207" OD minus 2x thickness of 2")

8.6 Elevation of TAF - Approx 360" (Reference TVA Calc MD-Q0063-920470, Rev 0, Attachment A)

8.7 Elevation of BAF - Approx 216" (Reference TVA Calc MD-Q0063-920470, Rev 0, Attachment A)

8.8 Inside Radius of Vessel and Lower Head - 125.5" (Reference TVA Calc MD-Q0063-920470, Rev 0, Attachment A)

8.9 Reference Pressure for Determination of Coolant Mass***** - 1015 psia (Reference TVA Calc MD-Q0063-920470, Rev 0, Attachment A)

*****Also used as reference pressure for determining initial steamline temperature (saturation temperature at the pressure specified)

8.10 Liquid Specific Volume at Reference Pressure - 0.02166 ft^3/lbm (Reference TVA Calc

MD-Q0063-920470,
Rev 0, Attachment A)

8.11 Test Basis for Drywell-to-Torus Vacuum Breakers -
Leak rate less than 1" orifice equivalent (Reference BFNP UFSAR Q5.1 Response, Amendment 24)

A/√K approx 0.0033 ft^2

8.12 Maximum Suppression Pool Temperature - 173 F (Reference BFNP UFSAR Table 14.6-3, Amendment 7, and Figure 14.6-12, Amendment 7)

8.13 Minimum ECCS Injection Temperature Post-Blowdown - 150 F (Reference BFNP UFSAR Figure 14.6-12, Amendment 7)

8.14 Vent Submergence - 3.5 ft (Reference NEDO-24580, Rev 2, Table BF 4.1.1-1)

8.15 Reference Pressure for Drywell to Steamline Volumetric Flow Conversion - 14.4 psia (Reference TVA Calc ND-Q0031-920075, Rev 6)

Appendix D

	Page
Workscope (ATTACHMENT A TO Purchase order)	D-2 to D-8
Letter September 20, 1995	D-9 to D15

Attachment A

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Volumetric flowrate (drywell to torus) corresponding to minimum steaming rates

The two volumetric flows of interest can be determined assuming the drywell is steam-filled at 41.7 psia and near-saturation (based on Reference 6, Item 8.1). From Exhibit 1:

$$v_g = v_g \text{ @ } 40 \text{ psia} - (1.7\text{psi}/10\text{psi})(v_g \text{ @ } 40 \text{ psia} - v_g \text{ @ } 50 \text{ psia})$$

$$v_g = 10.5 - (1.7/10)(10.5 - 8.5)$$

$$v_g = 10.2 \text{ ft}^3/\text{lbm}$$

Volumetric flow corresponding to 4.4 lbm/sec = 4.4(10.2) = 45 cfs (to be used from 1830 sec to 7230 sec)

Volumetric flow corresponding to 31.9 lbm/sec = 31.9(10.2) = 325 cfs (to be used from 7230 sec to 7890 sec)

For a drywell volume of 159000 ft^3 (Reference 6, Item 3.1) the quench flowrate of 325 cfs corresponds to a drywell sweep-out rate of 7.4 per hour, comparing favorably with the 10 per hour rate given in Reference 5. This rate is sufficiently high to permit it to be used to characterize the "well-mixed" behavior of the containment beyond the core debris quench.

A question that could be raised regarding the volumetric sweep-out rate is the effect of condensation in the drywell on the correspondence between the minimum sweep-out rate and the minimum steaming rates; i.e., could condensation decrease the sweep-out rate for a given steaming rate. The answer is two-fold. First, Appendix B discusses the fact that condensation would not be expected during core degradation because of heat-sink saturation during and immediately after blowdown. This explanation in Appendix B, however, is not Safety-Related because it is not necessary to the defense of the position that neglecting condensation is conservative. It is true that drywell condensation could decrease the sweep-out rate; but condensation also brings about diffusiophoretic removal of aerosol. Since the Reference 1 source term is dominated by aerosol, this is an important effect. If one considers the expression for diffusiophoretic aerosol removal in Reference 3 (recognizing the drywell is steam-filled), it reduces to:

Removal rate = Steam Condensation Rate/Steam Density

And this expression is the same as one would obtain for the volumetric sweep-out rate of the drywell if the steam generated in the drywell were flowing into the torus instead of condensing in the drywell. Therefore, the two phenomena are essentially equivalent; and as a matter of fact, the radionuclide removal efficiency would be expected to be greater for diffusiophoretic deposition than for flow to the torus because of pool bypass and the difficulty of scrubbing small aerosols. Therefore, steam condensation in the drywell, to the small extent it may occur, can be neglected.

Results

The volumetric flows to be used for the exchange between the drywell and the torus are as follows:

From t=0 to t=1830 seconds:	Flow from drywell to torus = 0	(no source term for first 30 seconds, no steaming during gap release)
	Flow from torus to drywell = 0	(no return flow during release phase)
From t=1830 to t=7230 secs:	Flow from drywell to torus = 45 cfs = 1.6E5 cfh	
	Flow from torus to drywell = 0	(no return flow during release phase)
From t=7230 to t=7890 secs:	Flow from drywell to torus = 325 cfs = 1.2E6 cfh	
	Flow from torus to drywell = 0	(no return during core debris quench)
From t=7890 seconds to end:	Flow from drywell to torus = 1.2E6 cfh	(mixing flow - no scrubbing)

Flow from torus to drywell = 1.2E6 cfh (mixing flow)

A comparison of these results to similar results for severe accident analyses of various sources is provided in Appendix C. It is useful to review these comparisons because these comparisons confirm the behavior discussed in this calculation. However, Appendix C is not Safety-Related because the results presented above do not depend on any of the Appendix C observations.

Conclusions

The flow from the drywell to the torus during the core degradation is about one drywell volume per hour. This is comparable to other natural removal rates. This value, by itself, will decrease the average radioiodine concentration in the drywell during the core degradation by about a factor of 1.6 if referenced to the Reference 1 source term (without removal) or by about a factor of 3.0 if referenced to the Reference 2 source term. See Appendix A.

The flow from the drywell to the torus during the final core debris quench is about seven and a half drywell volumes per hour, but it only lasts for 11 minutes. The final core debris quench will decrease the radioiodine in the drywell by about a factor of four (i.e., $1/e^{-7.5(11/60)}$).

These effects combine with suppression pool scrubbing (of the flow from the drywell to the torus) and with aerosol sedimentation to yield significant decontamination of the containment atmosphere.

APPENDIX A

APPENDIX TITLE:

"Use of a Uniform Sweep-Out Rate During the Release Phase"

SAFETY-RELATED APPENDIX: Yes

CALCULATION NUMBER: PSAT 04011H.01

CALCULATION TITLE:

"Volumetric Flowrate as a Function of Time from Drywell to Torus (and Return)"

Purpose

The purpose of this appendix is to justify a uniform sweep-out rate from the drywell to the torus during the release phase from essentially t=0 to t=120 minutes.

Approach

The approach is to set up a spread-sheet wherein:

- A release of 5% radioiodine is introduced over 30 minutes with no removal, and
- An additional 25% is added over 90 minutes using (1) no removal, (2) removal at a constant rate ("lambda") of one per hour, and (3) a linearly increasing removal rate beginning at zero and increasing to two per hour at the end of the 90 minutes.

The percent airborne is plotted and the integral under each of the curves is also calculated. The area under the curve (in %-minutes) is indicative of the release that would occur from the drywell for a constant leak rate and no decay. An assumption of no decay is acceptable since I-131 is the dominant radioiodine nuclide and it has a half-life of 8.1 days compared to the two-hour duration of this calculation.

Results

The results are shown on Figure A-1. The accuracy of the spread-sheet can be checked by observing the slope of the calculation for any percent airborne. For example, for the increasing

lambda case the maximum airborne percent (about 13.1%) is reached at about 84 minutes. At 84 minutes the variable removal rate would be:

0 + 2 x (84 min -30 min)/ 90 min = 1.2 /hour

The removal in terms of %/hour would be:

1.2 x 13.1 = 15.7 %/hour = 0.261 %-min

This is almost exactly the addition rate (0.278 %-min) which explains the zero slope.

As another example, the constant removal rate case ends with an increasing slope of about 0.3 %/ 6 min or 0.05 %/min with an airborne percent of about 13.7%. The removal rate at this percent would be:

1 /hour x 13.7% x 1/60 hours/minute = 0.228 %/min

The net increase would be:

0.278 %/min (added) - 0.228 %/min (removed) = 0.05 %/min

The results in terms of areas under the curves is shown on the figure. Note that the area under the constant removal curve is only 5% less than the area under the increasing removal curve. This shows that using a constant removal rate to approximate the increasing removal rate is acceptable, at least for the case of limited removal (i.e., one per hour). A larger removal rate would increase this difference and make the constant removal rate approximation increasingly non-conservative.

It is also of interest to note that either of the removal cases are about a factor of 1.6 better than the no-removal case.

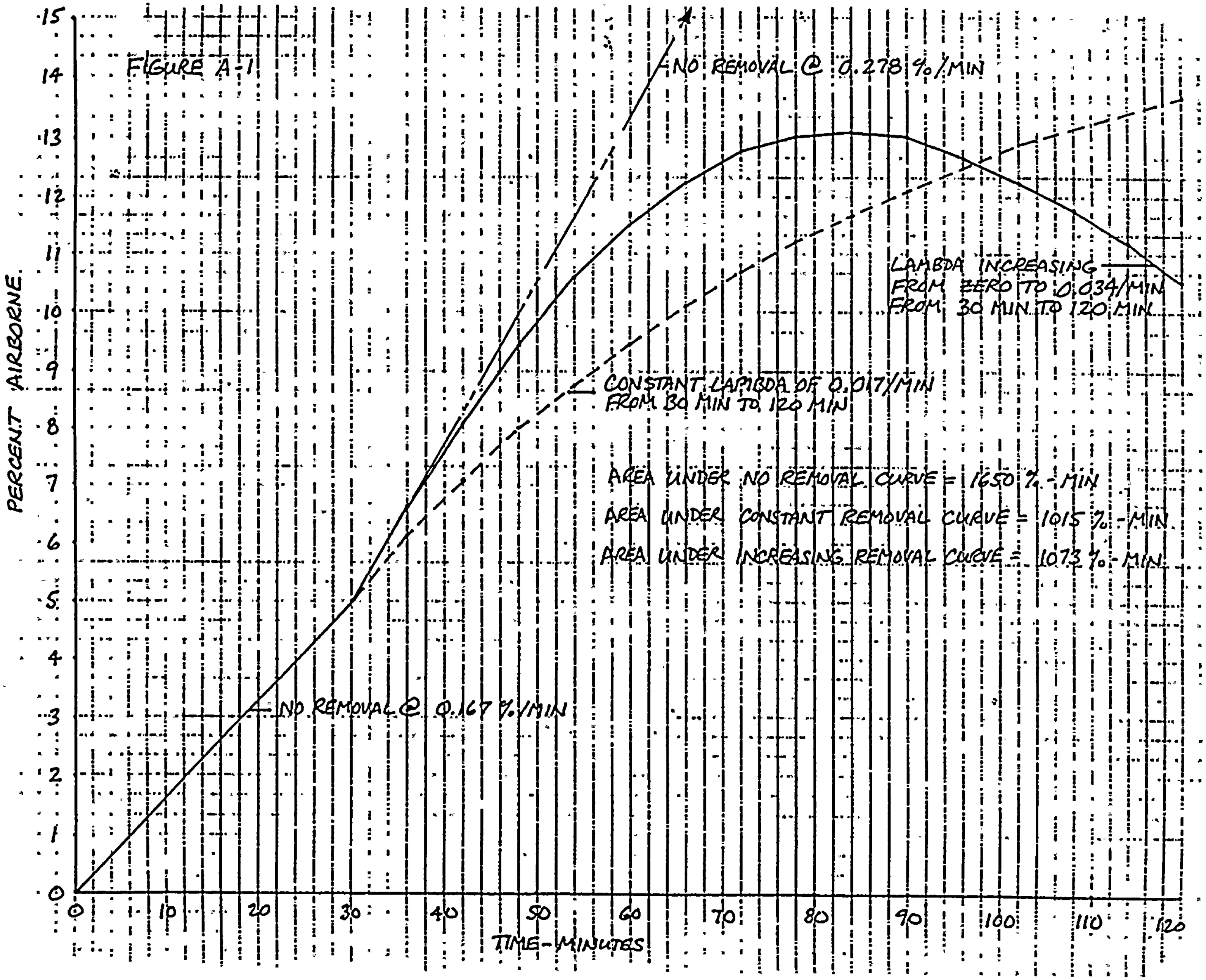

FIGURE A-1

AREA UNDER NO REMOVAL CURVE = 1650 %-MIN
AREA UNDER CONSTANT REMOVAL CURVE = 1015 %-MIN
AREA UNDER INCREASING REMOVAL CURVE = 1073 %-MIN

APPENDIX B

APPENDIX TITLE:

"Impacts of Transient Heat Conduction"

SAFETY-RELATED APPENDIX: No

CALCULATION NUMBER: PSAT 04011H.01

CALCULATION TITLE:

"Volumetric Flowrate as a Function of Time from Drywell to Torus (and Return)"

Purpose

The purpose of this appendix is to show (1) that the drywell shell is likely to saturate thermally well before significant fission product release begins and (2) that the reactor vessel will still retain a significant amount of sensible heat at the time the fission product release begins. The first finding supports the view that little condensation will be occurring in the drywell during core degradation and the second supports the view that neglecting sensible heat transfer from the vessel shell is a significant conservatism when considering steam generation during core degradation and the associated purge flow from the drywell to the torus.

Approach

The approach involves estimating the equilibration time for transient heat transfer to the drywell shell and from the vessel shell and comparing those times to the start of the bulk of the fission product release to the containment (i.e., the start of the in-vessel release phase at t=30 minutes). Exhibit 1, taken from "Principles of Heat Transfer" by Kreith, constitutes the basis for these estimates.

The drywell shell assumed data is as follows:

$L = 0.125$ ft (assumed thickness of shell = 1.5 inches)
$\theta = 0.5$ hours (30 minutes - time before start of bulk of fission product release)
$\alpha = 0.5$ ft^2/hour (thermal diffusivity for carbon steel)
$h = 100$ BTU/ft^2-F-hr (typical steam condensation heat transfer coefficient when noncondensibles are present)
$k_s = 26$ BTU/ft-F-hr (thermal conductivity for carbon steel)

The vessel shell data is assumed to be the same except L = 0.75 ft (9" thickness). The surface heat transfer coefficient of h = 100 BTU/hr-ft^2-F is also representative of heat transfer from a surface to liquid water.

Results

For the drywell shell, Bi = 0.5 and Fo = 16 at 30 minutes. From Exhibit 1, Q/Q_0 is essentially unity indicating that all heat transfer that can occur (for a given temperature difference) will have occurred by this time; i.e., the shell is thermally saturated. The shell would be 95% saturated by the time Fo = 8; i.e., by about 15 minutes.

For the vessel shell, Bi = 2.9 and Fo = 0.4 at 30 minutes. From Exhibit 1, Q/Q_0 is about 0.5 indicating that about 50% of the sensible heat initially in the vessel shell remains at 30 minutes with the other 50% having been transferred to the residual water. (Note that during the 30 seconds or so of blowdown, the Fo would be less than 0.01 and virtually no sensible heat would have been transferred). The 50% of the initial sensible heat transferred during the first 30 minutes after blowdown, in terms of actual BTUs, can be estimated by assuming the weight of the portion of the vessel shell in contact with the residual water to be about 60 tons (half of the lower head). Given this assumption, 50% of the original stored energy (remembering that the outside is insulated) would be about 2 MBTU. If transferred over 30 minutes, the average heat transfer rate would be about 4 MBTU/hr or 1.2 Mw. This is comparable to the initial heat transfer rate calculated from the core debris at 30 minutes.

By 120 minutes (end of the fission product release to the containment) Fo would be 1.6 and the 50% remaining sensible heat would have been largely transferred to the residual water. If transferred uniformly over the 90 minute interval corresponding to the bulk of the fission product release, the transfer rate would be about 1.3 MBTU/hour or 0.4 Mw. This is about 10 percent of the average heat transfer rate from the core debris assumed in the main calculation.

Based on the above, ignoring the contribution of the sensible heat stored in the lower head after blowdown is a significant conservatism. This heat would produce more than one megawatt of steaming during the first half hour (i.e., during the gap release when no steaming was assumed) and would add about 10 percent to the average steaming rate during the bulk of the fission product release.

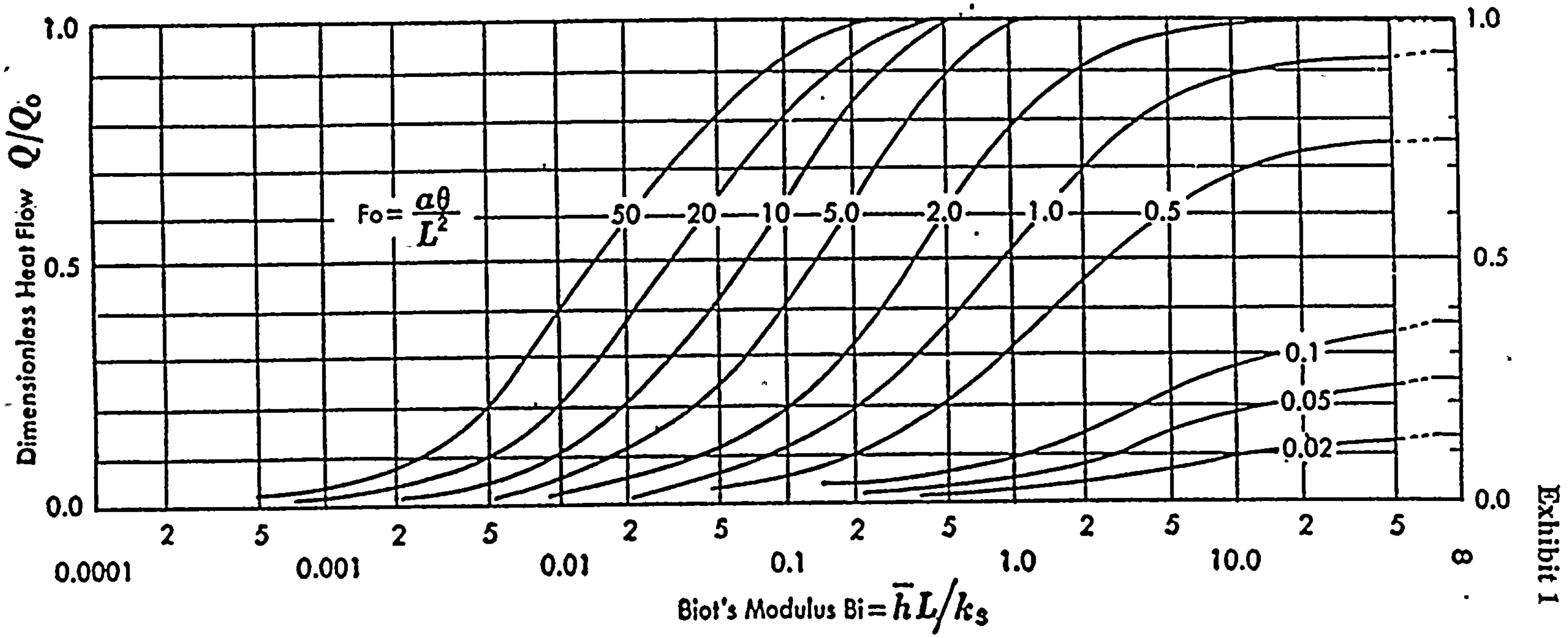

FIG. 4-9. Dimensionless heat flow to or from a wall subjected to a sudden change in environmental temperature.

Exhibit 1

Figure 4-9 is a plot of Q/Q_o vs. the Biot modulus for various values of Fo. Here Q represents the total change in internal energy per unit area, i.e., the amount of heat transferred per unit area in the time interval between $\theta = 0$ and $\theta = \theta$ in Btu per square foot; Q_o represents the initial internal energy per unit area relative to the fluid temperature T_∞, i.e., $c\rho L(T_o - T_\infty)$. A positive value of Q indicates, therefore, that heat is transferred from the wall to the fluid, while a negative value of Q shows that the direction of heat flow is into the slab.

APPENDIX C

APPENDIX TITLE:

"Comparison to Severe Accident Analyses"

SAFETY-RELATED APPENDIX: No

CALCULATION NUMBER: PSAT 04011H.01

CALCULATION TITLE:

"Volumetric Flowrate as a Function of Time from Drywell to Torus (and Return)"

Purpose

The purpose of this appendix is to present severe accident analyses done by Battelle Columbus (an NRC contractor) and by TVA, itself, that add support to the estimates of accident progression and thermal-hydraulic behavior for the DBA LOCA that constitute the main part of this calculation.

Approach

Two Battelle analyses have been done in which the initiating event is a large LOCA. The plant actually analyzed is Peach Bottom, but as can be seen on Exhibit 1 (3 pages) taken from Table 4.1-1 of the Browns Ferry Individual Plant Examination (IPE), Peach Bottom and Browns Ferry are nearly identical. The Source Term Code Package (STCP) was used for these analyses.

The two Battelle analyses include a recirc suction LOCA with no injection (AE-γ, where the γ indicates a large, early containment failure) and an interfacing-system LOCA outside containment (so-called V-sequence which involves loss of injection, as well, because the line break outside containment knocks out the ECCS). These analyses are documented in BMI-2104 Volume II (July 1984) and BMI-2139 Volume 1 (NUREG/CR-4624, July 1986), respectively. Since in both cases the containment function is assumed to be lost either prior to or very early in the accident progression, it is not useful to look at the containment response. However, a comparison of overall event timing (to the assumptions used in the main part of this calculation for the DBA LOCA) and of primary system parameters is useful.

Several large LOCA analyses have also been made by TVA using MAAP3B. These include a recirc suction LOCA with no injection, the same event with recovery of ECCS injection prior to vessel failure, and a main steamline LOCA (inside containment) with recovery of ECCS prior to vessel failure. For these analyses the overall timing is compared to the assumptions used in the main part of this calculation; and also, a detailed comparison of noble gas transport in containment is made to investigate the overall thermal-hydraulic behavior of the containment and to further support the transport analyses and assumptions made in the main body of this calculation

Results

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Exhibit 1, Sheet 1

Table 4.1-1 (Page 1 of 3). Basic RCS and Containment Comparison

Plant Name Type of Reactor Type of Containment	Peach Bottom BWR/4 Mark I	Browns Ferry BWR/4 Mark I
<u>Reactor Core</u>		
Thermal Power (Mwt)	3,293	3,293
Number of Fuel Assemblies	764	764
Number of Control Rods	185	185
<u>Reactor Vessel</u>		
Inside Diameter (inches)	251	251
Inside Height (feet)	72.92	72.92
Design Pressure (psig)	1,250	1,250
Number of Safety Valves	2	0
Lowest Safety Valve Setpoint (psig)	1,230	N/A
Safety Valve Capacity (klb/hr)	925	N/A
Safety Valves Vent To	Drywell	N/A
Number of Relief Valves	11	13
Lowest Relief Valve Setpoint (psig)	1,105	1,105
Relief Valves Capacity (klb/hr)	889	851
Relief Valves Vent To	Suppression Pool	Suppression Pool
<u>RHR System</u>		
Number of Loops	2	2
Number of Pumps	4	4
Flow Rate per Pump (gpm at psid reactor vessel to drywell)	10,000 at 20	10,000 at 0
Number of Heat Exchangers	4	4
Maximum Capacity of Heat Exchanger (Btu/hr)	70,000,000	70,000,000
<u>RHR Service Water System</u>		
Number of Pumps	3	8
Flow Rate per Pump (gpm)	4,666	4,500
<u>Core Cooling Systems</u>		
RCIC		
Number of Pumps	1	1
Capacity (gpm at psid)	616 at 1,120	616 at 1,120
HPCI		
Number of Pumps	1	1
Flow Rate per Pump (gpm at psid)	5,000 at 1,120	5,000 at 1,120

Exhibit 1, Sheet 2

Table 4.1-1 (Page 2 of 3). Basic RCS and Containment Comparison Table

Plant Name Type of Reactor Type of Containment	Peach Bottom BWR/4 Mark I	Browns Ferry BWR/4 Mark I
LPCI (RHR)		
Number of Divisions	2	2
Number of Pumps per Division	2	2
Flow Rate per Pump (gpm at psid reactor to dry vessel)	10,000 at 20	10,000 at 0
Core Spray		
Number of Divisions	2	2
Number of Pumps per Division	2	2
Flow Rate per Pump (gpm at psid)	3,125 at 122	3,125 at 105
Shutoff Head (psid)	N/A	~ 400
<u>Containment</u>		
Constructor	CBI	PDM
Drywell Material and Construction	Steel	Steel
Drywell Free Volume (ft^3)	175,800	159,800
Drywell Design Temperature (°F)	281	281
Torus Material and Construction	Steel	Steel
Torus Minimum Free Volume (ft^3)	123,000	126,200
Torus Maximum Water Volume (ft^3)	N/A	127,800
Torus Design Temperature (°F)	281	281
Containment Design Pressure (psig)	56	56
Drywell to Torus Vent Configuration	Diagonal large-diameter vertical piping venting below the water level of the pool.	Diagonal large-diameter vertical piping venting below the water level of the pool.
Drywell Spray (RHR)		
Number of Trains	2	2
Flow Rate per Pump (gpm at psid reactor to dry vessel) (Amendment 8, FSAR)	10,000 at 20	10,000 at 0

Exhibit 1, Sheet 3

Table 4.1-1 (Page 3 of 3). Basic RCS and Containment Comparison Table

Plant Name	Peach Bottom	Browns Ferry
Type of Reactor	BWR/4	BWR/4
Type of Containment	Mark I	Mark I
<u>Secondary Containment</u>		
Reactor Zone Free Volume below Refueling Floor (ft^3)	1,122,000	1,360,000
Blowout Panel Design Pressure		
Hatch Cover (psid)	N/A	0.25
Refueling Floor (psid)	0.25	0.25
Steam Tunnel (psid)	0.30	0.625
Standby Gas Treatment System		
Design Flow (Unit 2, CFM)	N/A	4,660
Refueling Floor Area (three units)		
Free Volume (ft^3)	1,314,000	2,601,000
Blowout Panel Design Pressure (psid)	N/A	0.35
<u>Turbine Building</u>		
Volume (ft^3)	2,100,000	5,700,000

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CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04001H.02

CALCULATION TITLE:

"Aerosol Decay Rates (Lambda) in Drywell"

		ORIGINATOR		CHECKER		IND REVIEWER	
		Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION:	0	Jun Li Li Jun	8/28/95	R. SHER R. Sher	9/1/95	R. SHER R. Sher	9/1/95
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	3						
	4						

REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this analysis is to calculate the aerosol decay (removal) rates in the drywell due to natural removal mechanisms that remove fission and non-fission product aerosols from the drywell atmosphere.

Methodology

The problem to be solved can be described as follows:

During a design base accident (DBA), fission product aerosols are released from the damaged core into the drywell, together with significant amounts of steam and non-condensable gases. The steam and gases, as well as the heat transfer to the gases in the drywell, will cause an increase in drywell pressure and result in a significant sweeping flow into the wetwell through the vent/downcomers that connect the drywell and wetwell. Leakage flows into the main steam lines through the MSIVs and directly to the reactor building are also expected. All these flows will dilute or remove the aerosols in the drywell and, at the same time, the aerosols will experience other removal processes, such as sedimentation, diffusiophoresis, thermophoresis, etc., the rates of which are to be determined in this analysis.

Based on the mass conservation law, the suspended aerosol mass in the drywell is governed by the following equation:

Suspended mass = Injected mass - Leaked mass - Removed mass

The injected mass of aerosols include both fission and non-fission product aerosols from the primary system. The leaked mass accounts for the aerosols entrained in the leak flows through several leakage pathways, such as the vent and bypass that connect the drywell and wetwell, the MSIV leakage, and the drywell leakage, and the removed mass represents the aerosols deposited on the surfaces in the drywell due to sedimentation, diffusiophoresis, thermophoresis, and other aerosol removal processes. All of the quantities in the equation can be functions of time.

The above equation is solved by the STARNAUA code [reference 1] in which the aerosol removal processes mentioned above are modeled, and the suspended aerosol concentration is calculated for the specified timing and rates of injected aerosols and the specified aerosol leakage rates through different pathways.

Assumptions

Assumption 1: The drywell is well-mixed during the entire time period of the accident.

Justification: Given the fact that steam, non-condensable gases (e.g., hydrogen) and fission product gases and aerosols are blowing into the drywell atmosphere, while significant heat and mass transfers are going on in the drywell, this assumption is reasonable.

Assumption 2: Condensation and sensible heat transfer onto the drywell walls are neglected.

Justification: Since the drywell walls are insulated, the initial blowdown before any release of fission product aerosols is expected to heat up the walls very quickly so that further heat transfers (both condensational and sensible) to the wall during and after the release of fission product aerosols will not be significant. Nevertheless, this assumption is conservative in the sense that it will result in a smaller aerosol decay rate.

Assumption 3: Hygroscopicity of aerosols is ignored and relative humidity in the drywell is assumed to be 98% through-out the accident.

Justification: The cesium and iodine species (mostly CsI and CsOH) released into the drywell are likely to be soluble and the hygroscopic effect on the growth of the soluble aerosols is significant, which enhances the removal of such aerosols by increasing sedimentation. The assumption to ignore the hygroscopicity will then be conservative. A relative humidity of 98%, on the other hand, has no impact on this analysis since both the hygroscopic effect on aerosol growth and diffusiophoresis (that is indirectly affected by the relative humidity) are not considered. Neglecting diffusiophoresis is also conservative.

Assumption 4: The amount and timings of the fission product releases are obtained from NRC documents. The release fractions are obtained from NUREG-1465 [reference 2] (see Tables 3.8 and 3.12) and the core inventories are from Table 4.6 NUREG/CR-4624 [reference 3], all of which are summarized in Table 1 below. The timings are also obtained from NUREG-1465. Two phases of the fission product release are assumed. First, the gap release starts at 30 seconds after the initiation of the accident and lasts 1800 seconds. It is then followed by the early in-vessel release that lasts 1.5 hours.

According to NUREG-1465, the iodine specie released to the containment is in the forms of particulate and gases (organic and elemental). 95% of the iodine released to the containment is aerosol, while 5% is gases. Of the iodine gases, 97% are elemental and 3% are organic. Organic iodine behaves like a noble gas, so it is assumed to be non-removable. Elemental iodine, on the other hand, tends to deposit on aerosols or other surfaces, and is assumed to be removed similarly to the aerosols.

Discussion: The core inventories in NUREG/CR-4624 were actually from NUREG/CR-2181 done in 1982. Since then, the total burn-up of the fuel assemblies in a fuel cycle in Browns Ferry has increased, which will result in an increase in the fission product core inventories. However, as far as the calculation of the aerosol decay rates is concerned, it is conservative to use smaller core inventories, since the only possible impact from an increased core inventory is to get a higher aerosol concentration and, consequently, a larger aerosol decay rate. On the other hand, the assumption of smaller core inventories is not overly conservative in this analysis since the aerosol removal processes are less significant than the removal due to the sweeping flow from the drywell to the wetwell.

Assumption 5: The amount of non-fission product aerosols released to the containment is the same as that of fission product aerosols (i.e.,

about 77 kg). They are released uniformly during the in-vessel release period, similar to the fission product aerosol release. The average density of the non-fission product aerosols is assumed to be 5.6 g/cm^3.

Justification: The assumption that the ratio of fission to non-fission in-vessel releases is 1:1 is obtained from reference 4. It should be pointed out that it was mentioned in NUREG-1465 that about 780 kg of in-vessel non-fission masses was calculated in NUREG-0956 for one Peach Bottom sequence. Since the Peach Bottom reactor is almost identical to the Browns Ferry reactor that is analyzed here, the same order of magnitude of non-fission product release is expected. But, the non-fission product release that we assume is only 10% of what was calculated in NUREG-0956. Our assumption should then be conservative, since a larger amount of non-fission product release will enhance overall aerosol agglomeration and, therefore, increase aerosol sedimentation. As for the density, most of the non-fission product aerosols are Zr, Fe_2O_3 and UO_2 species whose densities are 6.4, 5.24 and 10.09 g/cm^3, respectively. So, a density of 5.6 g/cm^3 for the non-fission product aerosols represents a conservative value, considering that the Zr inventory in the core is almost three times higher than that of the iron (table 4.5, reference 3).

Table 1. Fission Product Releases Into Containment

Group	Title	Elements in group	Gap release[1]	Early in-vessel release[1]	Core inventory (kg)
1	Noble Gases	Xe, Kr	0.05	0.95	413
2	Halogens	I, Br	0.05	0.25	16.6
3	Alkali Metals	Cs, Rb	0.05	0.20	230
4	Tellurium Group	Te, Sb, Se	0.	0.05	34.9
5	Barium, Strontium	Ba, Sr	0	0.02	167.7
6	Noble Metals	Ru, Rh, Rd, Mo, Tc, Co	0	0.0025	584
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0	0.0002	837
8	Cerium Group	Ce, Pu, Np	0	0.0005	992

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[1] Fractions of core inventories.

Assumption 6: The flow exchange between the drywell and the wetwell is ignored after containment heat removal (or reflood) is over.

Justification: According to PSAT 04000U.03 [reference 5] (Items 3.10 and 3.11), before 7890 seconds the flow exchange between the drywell and the wetwell is only in one direction, i.e., from the drywell to the wetwell. So, the flow can be considered as a leakage flow out of the drywell. After 7890 seconds the flow from the drywell to the wetwell is balanced by the flow from the wetwell to the drywell. To fully model the two-way flow exchange, the calculation of aerosol behavior in both the drywell and the wetwell needs to be conducted in parallel, which will be very difficult. This assumption, evidently, simplifies the problem. The implication of the effect on the drywell aerosol decay rate calculation needs to be discussed when the result is used. Nevertheless, it should be pointed out that the aerosol decay rate in the wetwell is almost always higher than that in the drywell, since

- the aerosols entering the wetwell from the drywell are more or less scrubbed, especially if the suppression pool is sub-cooled.
- the wetwell has a smaller airspace volume than the drywell (1:1.28), and a larger sedimentation area than the drywell (1.67:1). Thus the wetwell is more favorable for aerosol sedimentation.

Assumption 7: Aerosol size distribution is log normal, with a geometric mean radius of 0.22 micron and a geometric standard deviation of 1.81.

Justification: As discussed in Reference 6 (page 12-13), the overwhelming majority of aerosols are observed to have a lognormal size distribution. It is also a common practice to assume such a distribution for the fission product aerosols in nuclear safety studies. A lognormal distribution is defined by the geometric mean radius and the geometric standard deviation. The values for them to be used in this calculation are based on an analysis of data from several degraded fuel experiments [reference 7]. It should be pointed out that the aerosols size distribution specified here yields a mass mean diameter of about 1.3 microns. For comparison, the mass mean diameters used in NUREG/CR-5966 [reference 8] range from 1.5 to 5.5 microns and the geometric standard deviations range from 1.6 to 3.7 (see page 84). Thus, our assumption is evidently at the lower end of what were used in reference 8, and is thus conservative compared with reference 8.

Reference

Reference 1: PSAT C101.02, "STARNAUA - A Code for Evaluating Severe Accident Aerosol Behavior in Nuclear Power Plant Containment: A Validation and Verification Report, Revision 0, May 1995

Reference 2: Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, February 1995

Reference 3: Denning, R. S., et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios, BWR, Mark I Design", NUREG/CR-4624, BMI-2139, Vol. 1, July 1986

Reference 4: Letter from J. C. DeVine, Jr. to Leonard Soffer, "Additional ALWR Program comments on the NRC draft source term report, NUREG 1465", July 30, 1993

Reference 5: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 0

Reference 6: Fuchs, N. A., "The Mechanics of Aerosols", Dovers Publications, Inc., New York, 1964

Reference 7: Polestar Memo from R. Sher to D. E. Leaver, "Aerosol Source Size Parameters", July 28, 1995

Reference 8: Powers, D. A. and Burson, S. B., "A Simplified Model of Aerosol Removal by Containment Sprays", NUREG/CR-5966, SAND92-2689, June 1993

Calculation

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Table 4. Sedimentation Lambda as A Function of Time

Time[1] (second)	Sed. Lambda (1/hour)	Tot. Lambda (1/hour)	Time[1] (second)	Sed. Lambda (1/hour)	Tot. Lambda (1/hour)
48	0.30	0.30	8627	0.90	0.90
550	0.29	0.29	9868	0.80	0.80
1115	0.30	0.30	11786	0.70	0.70
2426	0.40	1.41	14615	0.60	0.60
3229	0.50	1.51	18774	0.50	0.50
4016	0.60	1.61	25011	0.40	0.40
4916	0.70	1.71	35598	0.30	0.30
6321	0.80	1.81	57247	0.20	0.20
7393	0.90	8.45	99807	0.12	0.13
7902	0.99	0.99			

The STARNAUA output files are given in Appendices C and D. The headings are added to the plot file in Appendix C to make it understandable. The STARNAUA output file, on the other hand, has been shortened to avoid an unnecessarily long printout. The time in those output files is the STARNAUA time that starts at core uncovery, 30 seconds after the initiation of the accident.

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[1] Accident time, which is STARNAUA time + 30 seconds.

APPENDIX A

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APPENDIX C: "STARNAUA Plot File"

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CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.03

CALCULATION TITLE:

"Maximum Elemental Iodine Decontamination Factors"

	ORIGINATOR		CHECKER		IND REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	James Metcalf Jim Metcalf	8/31/95	David Leaver	9/1/95	David Leaver	9/1/95
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REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this calculation is to determine the maximum DF for elemental iodine (and for Te-132 which is being treated as elemental I-132 except for half-life) that can be credited in the Browns Ferry drywell and torus for the purpose of applying the revised DBA source term of Reference 1.

Methodology

The methodology will be that of References 2 and 3.

Assumptions

Assumption 1: The suppression pool pH will be maintained at a value of 6.0 or above.

Justification: A separate TVA project is being undertaken by Polestar (in progress at the time of this calculation preparation) to ensure that the suppression pool pH in the long term is not less than 7. According to Reference 4, Item 6.1, the initial suppression pool pH is not less than 6.0.

Assumption 2: The water on the drywell floor and that in the suppression pool will be

well-mixed.

Justification: The approach to maintaining an adequately high suppression pool pH will be to inject the SLCS sodium pentaborate as a buffer for any accident involving substantial core damage (such as the accidents identified in Reference 1 as the basis for the DBA source term). Mixing of this sodium pentaborate solution with all available water inside containment (which is a recognized necessity in the work of the separate TVA pH control project mentioned above) will also provide mixing and a uniform distribution of the radioiodine between water on the floor of the drywell (or in the reactor vessel) and that in the suppression pool.

References

Reference 1: Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, February 1995

Reference 2: Beahm, E. C., Lorenz, R. A., and Weber, C. F., "Iodine Evolution and pH Control", NUREG/CR-5950, November 1992

Reference 3: Leaver, D. E. and Metcalf, J. E., "Generic Framework for Application of Revised Source Term to Operating Plants", Draft EPRI Report, EPRI Research Project 4080-2, July 1995

Reference 4: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 0

Calculation

From Reference 4, Items 3.3 and 6.9, the water volume which could ultimately dissolve the iodine released from the core is:

V = 127800 ft^3 (suppression pool volume) + 1.22E6 lbm/ 62.4 lbm/ft^3 (mass of RCS, recirc loops, and LPCI loops/ nominal water density at low temperature)

= 127800 ft^3 + 19550 ft^3 = 147350 ft^3

= 147350 ft^3 x 28.3 liters/ft^3 = 4.17E6 liters

From Reference 3, for a high burn-up core, the core iodine mass is approximately 7.5 grams per Mw. From Reference 4, Item 8.3 the core power is 3458 Mw(t). This means the iodine mass is

approximately 2.6E4 grams. The iodine core inventory (most of which is stable or near-stable iodine) would be approximately 6.2E-3 grams per liter if 100 percent were released. The Reference 1 source term, however, involves only a 30% release of iodine for a BWR; and therefore, the iodine concentration (taken to be I^-) is 1.87E-3 grams per liter or about 1.4E-5 gm-atoms per liter.

From Reference 2 if $H^+ = 10^{-6.0}$ (i.e., pH = 6.0 - see Assumption 1), then:

$I_2 = (H^+)^2(I^-)^2/[d + e(H^+)]$ where: d = 4.22E-14, and
e = 1.47E-9

I_2 in the liquid phase = 4.5E-9 gm-moles/liter

I in the liquid phase = 9.0E-9 gm-atoms/liter

Since I^- in the liquid phase = 1.45E-5 gram-atoms/liter, then I/I^- = 6.2E-4 in the liquid phase.

From Reference 2, the partition coefficient is:

$\log_{10} PC(I) = 6.29 - 0.0149T$, where T is in K

From Reference 4, Item 8.12, the maximum pool temperature is 173 F = 352 K

Then:

PC(minimum) = 11.1 (i.e., the minimum concentration of iodine, as I_2 in the liquid phase is 11.1 times that in the gas phase. A lower temperature would yield a higher PC)

Since the gas phase volume = volume of drywell + volume of torus airspace

= (159000 + 124000) ft^3 (based on Reference 2, Items 3.1 and 3.2)

= 283000 ft^3

And since the volume of the liquid phase is 127800 ft^3 (suppression pool volume, Reference 4, Item 3.3), the ratio of the gas phase volume to the liquid phase volume is 2.2:1. This means that once removed from the gas phase, the mass of iodine, as I_2, in the liquid phase would never be less than (11.1/2.2 = 5) that in the gas phase. Since the maximum mass ratio of I/I^- in the liquid phase is 6.2E-4, the maximum mass ratio of I in the gas phase to I^- in the liquid phase is 6.2E-4/5 = 1.2E-4. This means that the minimum <u>ultimate</u> DF of elemental iodine (i.e., of molecular I_2 in the gas phase) for this system is approximately 1/1.2E-4 = 8000 <u>if</u> the iodine can be removed from the

gas phase initially.

Reference 1 indicates that 0.0015 of the iodine released to containment must be considered to be organic. This fraction is 13 times larger than the fraction of the iodine released which could re-evolve as I_2 (as calculated above). Therefore, as a practical matter, there is no need to limit the removal of inorganic iodine in the analysis of the revised DBA source term for Browns Ferry; the organic iodine (which is not removed by deposition or pool scrubbing) will always dominate. By Assumption 2 the water in the drywell and that in the suppression pool will have the same pH and radioiodine concentration; therefore, the concentration ratio (I_2 in the gas phase to I^- in the liquid phase) will be the same. This means that the I_2 concentration in the gas phase of the torus and the drywell will be the same, and a single control volume model of the containment is acceptable in the long-term from the standpoint of the potential for iodine re-evolution.

Results

The minimum justifiable long-term DF for elemental iodine in both the drywell and the torus is 8000. If this degree of decontamination can be achieved by removal mechanisms, then the associated re-evolved I_2 will not exceed eight percent of the organic iodine in the Reference 1 source term specification.

Conclusions

There is no need to limit elemental iodine removal in the analyses supporting application of the revised DBA source term to Browns Ferry.

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.04

CALCULATION TITLE:

"Suppression Pool Scrubbing Efficiency (Including Pool Bypass)"

	ORIGINATOR		CHECKER		IND REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	James Metcalf	8/31/95	David Leaver	9/1/95	David Leaver	9/1/95
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REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this calculation is to determine the maximum removal efficiency that can be credited for the passage of particulates and of elemental iodine (including Te-132 which is being treated as elemental I-132 except for half-life) through (and around) the Browns Ferry suppression pool for the purpose of applying the revised DBA source term of Reference 1.

Methodology

The methodology is that of Reference 2. Per Reference 2, a pool DF of 100 can be used for any particulate or elemental iodine passing through the pool. To account for pool bypass, a steam mass flow corresponding to 10 times the drywell-to-torus vacuum breaker surveillance test acceptance value is used. It is compared to the mass flow out of the drywell during and immediately after the source term release (referred to as the slow drywell sweep and the fast drywell sweep, respectively, in Reference 2) to determine the bypass fraction. No removal credit is taken for the fraction of the drywell sweep-out flow which bypasses the suppression pool. The overall pool DF (expressed as a "filter efficiency") is calculated accordingly.

Assumptions

Assumption 1: There is no dynamic pressure drop through the vent system during either

the slow sweep or fast sweep of the drywell.

Justification: The fast sweep-out of the drywell occurs with a vent flow of about 32 lbm/sec (Reference 3). As was discussed in Reference 3, during the blowdown of the reactor vessel for a DBA recirc suction LOCA all coolant is released except for a portion in the lower reactor vessel. If it is estimated that the total coolant released is of the order of 400000 lbm, that about 40% of that flashes to steam (Reference 3 calculates 38%), and that the blowdown requires not more than 30 seconds (the start of the gap release in Reference 1), the average steam flow through the vent system is approximately 5300 lbm/sec. Considering that the vent system differential pressure would not be expected to exceed 30 psid for such an event, and further that pressure drop would be, at worst, linear with flow (at best, proportional to flow velocity squared), the differential pressure for a flow of only 32 lbm/sec would be expected to be in the range of 0.001 to 0.2 psid. For a vent submergence of 3.5 ft (Reference 4, Item 8.14) the static differential pressure (to clear the vents) would be approximately 1.5 psid. Therefore, at worst, the dynamic pressure drop, even for the fast sweep-out flow of 32 lbm/sec, would be of the order of 10 percent of the static pressure difference and can therefore be neglected.

References

Reference 1: Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, February 1995

Reference 2: Leaver, D. E. and Metcalf, J. E., "Generic Framework for Application of Revised Source Term to Operating Plants", Draft EPRI Report, EPRI Research Project 4080-2, July 1995

Reference 3: PSAT 04011H.01, "Volumetric Flowrate as a Function of Time from Drywell to Torus (and Return)", Revision 0

Reference 4: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 0

Calculation

Polestar
PROPRIETARY

Results

The overall efficiency of the suppression pool as a filter for removing particulate and elemental iodine is 0.72 during the slow sweep-out (prior to 7230 seconds) and 0.95 during the fast sweep-

out (from 7230 seconds to 7890 seconds).

Conclusions

Because of the relatively low flowrate corresponding to the slow sweep of the drywell, the pool bypass (which is not sensitive to flowrate since it depends only on the hydrostatic pressure difference needed to clear the vents) is relatively large during that period leading to a relatively low overall efficiency. During the high flow of the fast sweep the pool bypass is correspondingly smaller and the efficiency correspondingly greater.

PSAT 04011H.05

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CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.05

CALCULATION TITLE:

"Additional Radionuclide Data"

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	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	James Metcalf	8/31/95	David Leaver	9/1/95	David Leaver	9/19/95
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REASON FOR REVISION:	Nonconformance Rpt
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Purpose

The purpose of this calculation is to present additional data regarding the dose conversion factors (DCFs) for Kr-90, Cs-134, and Cs-137, to develop a treatment for Te-132 based on its ability to decay to I-132 in elemental form, and to develop a treatment for radionuclides other than noble gas, radioiodine, radiocesium, and Te-132; i.e., the "Other".

Methodology

In Reference 1 it is pointed out that the thyroid dose due to I-132 formed by the decay of Te-132 is expected to be negligible compared to that due to I-131. Reference 1, however, does indicate that there may be an impact of the Te-132 (and of radiocesium) on the whole body dose. Therefore, the Te-132 and the radiocesium must be included and given appropriate DCFs. (Alternatively, Reference 1 states that a factor of 1.5 may be applied to the whole body dose due to radioiodine as a conservative means of addressing the impact of Te-132 and radiocesium on the whole body dose).

The basis for the majority of the DCFs in Reference 2 is Reference 3. The whole body and skin DCFs given in Reference 3 for Kr-90, Cs-134, Cs-137 (and its important, short-lived decay daughter Ba-137m), and Te-132 are zero (in fact, Kr-90 is not included at all). The thyroid DCF for Te-132 is a small, non-zero value which does not reflect its decay daughter (I-132). The "Other" is a concept introduced by Reference 4 (and not explicitly treated in the revised DBA source term of Reference 1 and Reference 5) which must be treated specially in any case.

Each one of these cases is discussed separately below, with whole body DCFs being based on

Reference 6 and skin DCFs being based on beta energies from Reference 7 and the expression for skin DCF, given beta energy, from Reference 8. Reference 8 is also used as the basis for the whole body DCF for the "Other".

Assumptions

Assumption 1: Te-132 is treated as I-132 with the half-life of Te-132.

Justification: The half-life of Te-132 is more than thirty times greater than that of I-132, and I-132 has a half-life of only 2.4 hours. Therefore, it is a conservative, but not unreasonable, to assume that I-132 appears immediately upon decay of Te-132. This can be accommodated by simply adding the DCFs for I-132 to those for Te-132. Based on Reference 3 this effectively means using I-132 DCFs.

Assumption 2: The "Other" can be conservatively treated as described in Reference 4.

Justification: Reference 4 suggests that "other" radionuclides (not including noble gas and radioiodine) be assumed to be released to the containment at the level of one percent of the core inventory. This release is comparable, or quite conservative, with respect to the release magnitude of "other" radionuclides in Reference 1 (based on Reference 5) with the exception of tellurium (Te-132 in particular) and cesium (Cs-134 and Cs-137 in particular). But since the Te-132, Cs-134, and Cs-137 are being explicitly treated separately, the "Other" can be treated as described in Reference 4 to verify that its contribution to the whole body dose is small.

References

Reference 1: Leaver, D. E. and Metcalf, J. E., "Generic Framework for Application of Revised Source Term to Operating Plants", Draft EPRI Report, EPRI Research Project 4080-2, July 1995

Reference 2: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 0

Reference 3: TACT5 Data File MLWRICRP.30 from "User's Guide for the TACT5 Computer Code", NUREG/CR-5106, June 1988

Reference 4: DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 1962

Reference 5: Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, February 1995

Reference 6: Chanin, D. I., et al., "MELCOR Accident Consequence Code System (MACCS) User's Guide", NUREG/CR-4691, Volume 1, February 1990

Reference 7: The Chemical Rubber Co., Handbook of Chemistry and Physics, 51st Edition, Cleveland, Ohio, 1970

Reference 8: NRC Regulatory Guide 1.3

Calculation

Kr-90

Kr-90 has a half-life of 33 seconds (Reference 7). Given the fact that the gap release does not even begin in Reference 1 until 30 seconds and that the release is at a rate of only 0.0028 %/second during the first 1800 seconds, the percent released over four half-lives would be less than 0.3 percent of the core inventory. Since Kr-90 is not further distinguished by either its abundance (Reference 2) or by its energy of disintegration (Reference 7), it is appropriate that it be dropped from further consideration.

Cs-134.

External exposure DCF in Reference 6 of:

6.97E-14 Sv-m^3/Bq-sec x 3.7E12 Rem-Bq/Sv-Ci = 0.258 Rem-m^3/Ci-sec

Beta energy from Reference 7 = 0.28(0.089)+0.01(0.410)+0.71(0.662) = 0.5 Mev

DCF = 0.23 x beta energy (by Reference 8) = 0.23 x 0.5 = 0.115 Rem-m^3/Ci-sec

Cs-137

External exposure DCF in Reference 6 of:

2.53E-14 Sv-m^3/Bq-sec (which includes impact of Ba-137m) x 3.7E12 Rem-Bq/Sv-Ci = 0.093 Rem-m^3/Ci-sec

Beta energy from Reference 7 = 0.94(0.511)+0.04(1.176) = 0.55 Mev

DCF = 0.23 x beta energy (by Reference 8) = 0.23 x 0.55 = 0.127 Rem-m^3/Ci-sec

Note that according to Reference 7, Ba-137m is not a beta emitter

Te-132

By Assumption 1, use I-132 DCFs already in Reference 3 to represent Te-132.

<u>Other</u>

The core inventory of "Other" is taken from Reference 4 based on an initial gamma source strength of 3.72E16 Mev/sec-Mw, an average gamma energy of 0.7 Mev/dis, 3.7E10 dis/Ci-sec, and the Reference 2 power level of 3458 Mw; i.e.,

Ci inventory = 3458 x 3.72E16 / (0.7 x 3.7E10) = 4.967E9 Ci

The effective half-life over the first two hours is given in Reference 4 (Table IV) as 2.72 hours. This represents a decay constant of 7.05E-5/second. Beyond two hours the 2.72 hour half-life overstates decay, but the particulate is largely removed by this point in time.

The expression for the whole body DCF from Reference 8 is:

WB DCF = 0.24 x average energy = 0.24 x 0.7 Mev = 0.168 Rem-m^3/Ci-sec

Results and Conclusions

1. Kr-90: Can be neglected altogether.
2. Cs-134: WB DCF = 0.258 Rem-m^3/Ci-sec
   Skin DCF = 0.115 Rem-m^3/Ci-sec
3. Cs-137: WB DCF = 0.093 Rem-m^3/Ci-sec
   Skin DCF = 0.127 Rem-m^3/Ci-sec
4. Te-132: Same as I-132 except for half-life
5. Other: 4.967E9 Ci in core at shutdown
   Decay Constant = 7.05E-5/second
   WB DCF = 0.168 Rem-m^3/Ci-sec

The "Other" is intended to show only that it's contribution is small.

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.06

CALCULATION TITLE:

"Source Term for Use on Browns Ferry Application of NUREG-1465"

	ORIGINATOR		CHECKER		IND REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	James Metcalf	8/31/95	David Leaver	9/27/95	David Leaver	9/27/95
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REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this calculation is to relate the source terms of Reference 1 to their application to the TVA Browns Ferry Nuclear Power Plant. In addition, a source term for radionulcides other than the important isotopes of the Noble Gasses, iodines, cesiums, and telluriums must be defined in order to be able to evaluate the impact of the "other" radionuclides on the TVA Browns Ferry offsite and control room doses for the DBA LOCA.

Methodology

The application of the revised source terms of Reference 1 to any plant requires the identification of the plant type (PWR or BWR) and a decision as to the time of the start of the gap release. For application to Browns Ferry the plant type is BWR, and for BWRs (according to Reference 1) the time to the onset of activity release (i.e., the start of the gap release phase) would be conservatively established if PWR timing were used. This is what has been done.

To obtain an estimate for the "other" release, the "other" source term from Reference 2 is compared to that from Reference 1 to make a determination of which is more conservative. The more conservative is chosen as the basis for evaluation of the "Other" to dose calculations for Browns Ferry.

Assumptions

Assumption 1: For application to Browns Ferry the PWR timing for the start of release is applied. This timing is approximated by the use of a 30 second delay from the time of reactor shutdown to that of the start of the gap activity release. Once begun, the gap activity release is assumed to be at a uniform rate over the 30 minute duration of the gap release phase.

Justification: By the commentary of Reference 1, this is conservative for Browns Ferry. Reference 1 states that for accidents where long-term cooling of fuel is maintained (e.g.,for a fuel handling accident), the release of the gap activity in failed pins (during the transient overheating of the fuel or immediately after mechanical damage) must be assumed to be instantaneous. This is a reasonable position. It also states that for accidents where long-term cooling is not maintained (e.g., for the 10CFR100 DBA which is the subject of this calculation), the release of the gap activity in the failed pins would be instantaneous, followed by an additional release (equal to 2/3 of the instantaneous release) over the full duration of the "gap release" (that release which occurs prior to the onset of fuel melting). This may be a reasonable position for an individual pin that has been operating at a high power level, but the timing of pin failures and the subsequent temperature rise in individual pins varies across core. This variation needs to be considered, as well as the fact that the magnitude of the gap inventory will not be uniform; i.e., higher burnup pins will, to a degree, exhibit higher gap activity.

According to Reference 1, the failures of the first pin is predicted to occur for PWRs at about 30 seconds after the loss of coolant; other pin failures will follow. A review of some of the analyses supporting Reference 1 (e.g., those listed on Tables 3.1 and 3.2 of Reference 1) indicate that the average core temperature can lag the peak core temperature by many minutes; and while this effect accounts for both radial and axial temperature distributions (and only the radial distribution is significant for the issue of relative timing of pin failures), it still suggests that the assumption of all pins failing in unison at approximately 30 seconds after the loss of coolant accident is excessively conservative.

A more reasonable assumption is one of a uniform release (over the duration of the gap release) totaling 1.67 times the assumed maximum gap inventory available for release at the start of the accident. This takes into account both the progressive nature of the pin failures and the additional release which will occur as pins increase in temperature after failure (but prior to fuel melting). In other words, if one assumes that 3% of the core inventory of a radionuclide of interest is in the gap at the time of the coolant loss, then 5% would be assumed to be released uniformly over the 30 minute duration of the gap release. This would correspond

to a rate of 0.17 % of the core inventory/minute for that radionuclide.

Assumption 2: HI may be neglected in terms of containment behavior and all iodine other than particulate CsI and organic iodine may be considered I_2.

Justification: Reference 3 states that I and HI will coexist and that I will be favored if hydrogen pressures are low and/or if temperatures are relatively high in the location where equilibrium is attained. Specifically, ~~is~~ of seven accident sequences studied in Reference 3, the only sequence in which the overall I + HI release exceeded 0.1% of the total iodine was a large break PWR LOCA. For this case, the relatively high temperature gradients within the RCS and the relatively low production of hydrogen (both due to the low steam generation rates characteristic of large break LOCAs) contributed to a relatively high percentage of non-CsI iodine (about 3.2%) but also to a relatively low ratio of HI to I (only 0.4% out of the 3.2%). It should be noted that a large break BWR LOCA was also studied (as one of the other six sequences for which almost no HI or I was found). Given these findings, it is evident that for relatively large release fractions of non-CsI iodine (characteristic of a PWR large break LOCA), little HI will be found, and that for BWRs, even for large break LOCAs, little HI will be found. I_2, on the other hand, has non-RCS sources as well as RCS sources and must be considered even for BWRs. Reference 1 also requires its consideration.

Once in containment, both I_2 and HI are reactive. The solubility of HI, however, is considerably greater than I_2 (nearly 3000 times greater on a molar basis); therefore, one would expect the persistance of HI as an airborne component to be less than I_2 in a steam and water environment. For this reason, as well as for its small release relative to I under the conditions where non-CsI iodine releases occur, it is considered reasonable to treat all non-particulate, non-organic iodine in containment as I_2.

References

Reference 1: Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, February 1995

Reference 2: DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 1962

Reference 3: Beahm, E. C., et al., "Iodine Chemical Forms in LWR Severe Accidents", NUREG/CR-5732, April 1992

Reference 4: Taylor, J., "Proposed Issuance of Final NUREG-1465, 'Accident Source Terms for Light-Water Nuclear Power Plants'", SECY-94-300, December 15, 1994

Calculation

Specification of Release Phases

Reference 1 describes four release phases: gap, early in-vessel, ex-vessel, and late in-vessel. Reference 4 establishes a precedent for advanced reactors (judged to be applicable to operating plants, as well) that only the first two phases need to be considered for DBA applications. Therefore, two release phases will be referred to: the gap release phase and the fuel release phase, with the fuel release phase making use of only the early in-vessel contribution from Reference 1.

Beginning, Duration, and Release Magnitudes of the Gap Release Phase

By Assumption 1 the gap release starts at 30 seconds and is uniform over time. By Reference 1 the duration of the gap release is 30 minutes. Release magnitudes are as follows (from Reference 1) given as fractions of core inventory and fractions of core inventory per second:

Noble Gas - 0.05 or 2.8E-5 /sec
Iodine* ---- particulate (CsI) - 0.0475 or 2.6E-5 /sec
---- elemental - 2.4E-3 or 1.3E-6 /sec
---- organic - 7.5E-5 or 4.2E-8 /sec
Cesium - 0.05 or 2.8E-5 /sec

*Based on 95% particulate, 4.85% elemental (see Assumption 2), and 0.15% organic

Beginning, Duration, and Release Magnitudes of the Fuel Release Phase

This phase begins at 1830 seconds (i.e., at the end of the gap release phase). The duration (from Reference 1) is 1.5 hours for BWRs; therefore, this release phase ends 7230 seconds after the beginning of the accident. Release magnitudes are as follows (from Reference 1) given as fractions of core inventory and fractions of core inventory per second:

Noble Gas - 0.95 or 1.8E-4 /sec
Iodine* ---- particulate (CsI) - 0.2375 or 4.4E-5 /sec
---- elemental - 1.2E-2 or 2.2E-6 /sec
---- organic - 3.8E-4 or 6.9E-8 /sec

Cesium - 0.2 or 3.7E-5 /sec
Tellurium - 0.05 or 9.3E-6 /sec

*Based on 95% particulate, 4.85%

elemental (see Assumption 2), and
0.15% organic

To calculate the contribution of "other" radionuclides, consider that Reference 1 provides the following release fractions:

Ba-Sr Group - 0.02
Noble Metal Group - 0.0025
Cerium Group - 0.0005
Lanthanide Group - 0.0002

By comparison, Reference 2 calls for the release of one percent of other radionuclides (other than noble gas and iodine). Since the tellurium and cesium are already being treated separately, it is only the Ba-Sr contribution from Reference 1 that exceeds the specification of Reference 2. On a per unit mass basis the Ba-Sr Group totals approximately 120% of the energy of the other three groups combined (at 5 hours after shutdown, see Reference 3), but the mass available for release is only about seven percent as great (also from Reference 3). Therefore, in combination the other three groups represent more than ten times the energy of the Ba-Sr Group. The fact that the Ba-Sr release fraction in Reference 1 is twice that of Reference 2 is more than compensated for by the fact that the Reference 2 Noble Metal release is four times greater than that of Reference 1, the cerium release is 20 times greater, and the lanthanide release is 50 times greater. Therefore, the one percent release of the "Other" (as presented in Reference 2) will bound the dose effects associated with the release fractions of the "other" from Reference 1 for those dose effects where energy is a good measure of importance (e.g., external whole body dose). For this reason, the one percent release of the "Other" will be used to show that the "Other" has no impact on whole body dose. The specified release of the "Other" is as follows (during the fuel release phase):

Other - 0.01 or 1.9E-6 /sec

Results

Fraction of core inventory, 0 - 30 seconds: no releases

Fraction of core inventory, 30 - 1830 seconds: Gases - Noble Gas - 2.8E-5 /sec (0.05 total)
Elemental I - 1.3E-6 /sec (2.4E-3 tot)
Organic I - 4.2E-8 /sec (7.5E-5 total)

Aerosols - Iodine - 2.6E-5 /sec (0.0475 total)
Cesium - 2.8E-5 /sec (0.05 total)

Fraction of core inventory, 1830 - 7230 seconds: Gases - Noble Gas - 1.8E-4 /sec (0.95 total)

Elemental I - 2.2E-6 /sec (1.2E-2 tot)
Organic I - 6.9E-8 /sec (3.8E-4 total)

Aerosols - Iodine - 4.4E-5 /sec (0.2375 total)
Cesium - 3.7E-5 /sec (0.2 total)
Tellurium - 9.3E-6 /sec (0.05 total)
Other - 1.9E-6 /sec (0.01 total)

Conclusions

The source term specification based on Reference 1 has the following characteristics:

1. Two release phases: a Gap Release Phase beginning at t=30 seconds, lasting 1800 seconds, and
a Fuel Release Phase beginning at t=1830 seconds, lasting 5400 seconds.

2. Iodine is in either particulate (dominant, as CsI aerosol) or in gaseous form (as I_2 or organic).

3. One percent of the non-noble gas, iodine, cesium, and tellurium radionuclides are released (i.e., the "Other") in order to bound the actual release fractions of Reference 1 in terms of impact on whole body dose. Note that the "Other" is specified only to verify the assumption that its inclusion in the dose calculation will have little or no impact. It should not be included in calculations of record unless its contribution is negligible.

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.07

CALCULATION TITLE:

"Drywell Leakage Rate Direct to Environment Mimicking Case 2 Early Bypass of SGTS"

	ORIGINATOR		CHECKER		IND REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	James Metcalf	9/22/95	David Leaver	9/27/95	David Leaver	9/27/95
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REASON FOR REVISION:	Nonconformance Rpt
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Purpose

The purpose of this calculation is to address an issue raised in Reference 1 regarding a 90 second interval (from t=15 seconds to t=105 seconds) at the beginning of the DBA LOCA during which the RB may exhibit a positive pressure and during which some flow out of the RB, therefore, may bypass the SGTS. In Reference 1 this issue was handled with a supplementary model shown on Exhibit 1. In this model the RB is explicitly modeled as a "hold-up" control volume between the drywell as a source and the environment. For the revised source term analysis (covered by this calculation) a direct release model is being used in which there is no "hold-up" control volume between the drywell and the environment for this release. The purpose of this calculation, therefore, is to develop a surrogate leak rate directly from the drywell to the environment that would conservatively represent the model of Exhibit 1.

Methodology

The approach is to:

(1) Calculate the release from the drywell to the RB assuming the containment is leaking at the design leakrate with "A" as the time-averaged activity airborne in the drywell over the first 105 seconds,

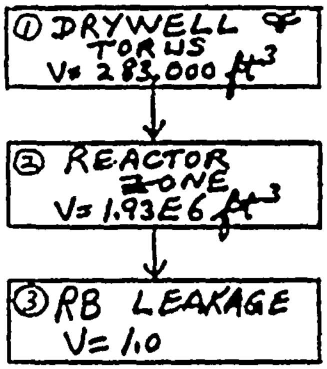
(2) Calculate the effective reduction in what would then be leaked to the environment

Exhibit 1

The STP model used to determine the RB leakage contribution to the CR doses is shown in Figure 3. The flows associated with the model are shown in Figure 4.

FIGURE 3.
STP MODEL
"S2LEAK"

SOURCE: 100% NG
25% I (INORG.)
25% I (ORG.)
0% PARTICULATES



(COMPONENT 3 IS AN ACCUMULATOR)

FIGURE 4
FLOWS IN STP MODEL S2LEAK

COMPONENT		FLOW
1	2	2%/d = 235.8 cfh
2	3	t=0-0.00417hr (15 sec) 0.0 cfh t=0.00417hr-0.02917hr (105 sec) 1542.4 cfh t>0.02917hr 0.0

To guarantee conservatism to the RB leakage dose, there is no SGTS flow assumed during the 15 to 105 second time period when the leakage occurs. This assures the maximum RB concentration during the period of RB leakage and hence a maximum dose from the leakage.

considering the presence of the RB, and

(3) Calculate a revised <u>direct</u> leakrate from the drywell that would then match that release to the environment from (2).

Assumptions

Assumption 1: It is conservative to place all drywell leakage that would occur over the first 105 seconds into the RB at the start of the accident.

Justification: The release from the containment does not begin until the start of the gap release at t=30 seconds (see Item 2.1 of Reference 2). Therefore, the RB has lost its residual negative pressure 15 seconds before the start of the gap release to the drywell and the corresponding release from the drywell to the RB. During the next 75 seconds (t=30 seconds to t=105 seconds), there will be a progressive release from the drywell to the RB as the drywell radionuclide concentration builds. During the first 105 seconds of the event, a time-averaged airborne radionuclide concentration, A, in the drywell can be defined. The leakage into the RB during the first 105 seconds can then be calculated as "A" times the fraction of the drywell volume leaked into the RB over the first 105 seconds. For simplification, then, it can be conservatively assumed that this product "AxB" (where "B" is the fraction of the drywell volume leaked to the RB over the first 105 seconds) appears in the RB at t=0 since this will maximize the radionuclide leakage from the RB to the environment over the subsequent 105 seconds.

References

Reference 1: TVA Calc ND-Q0065-900052, "CR Doses for 2 SGTS Fans Including RB Leakage", Revision 2, 5/4/93

Reference 2: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 1, September 22, 1995

Calculation

By Assumption 1, the radioactivity in the RB during the first 105 seconds of the DBA LOCA may be conservatively calculated to be:

$$\text{RB activity} = \text{A}\times\frac{\text{Volumetric Leakrate, Drywell to RB (Item 3.12 of Reference 2)}}{\text{Volume of Drywell (Item 3.1 of Reference 2)}}\times 105\text{ sec}$$

where "A" is the time-averaged airborne activity in the drywell over the first 105 seconds.

$$= \text{A} \times (132.5\text{ cfh}/159000\text{ ft}^3) \times 105\text{ sec}/3600\text{ sec/hr} = 2.43\text{E-}5 \times \text{A}$$

This activity, if placed in the RB at t=0 and if leaked from the RB at the RB leakrate of 1540 cfh (the flow out of the RB that does not pass through the SGTS when the RB pressure is positive, Item 3.30 of Reference 2), would yield a corresponding release of activity to the environment over the first 105 seconds (even neglecting the first 15 seconds when the RB pressure is negative) of:

$$\text{Activity released} = \frac{2.43\text{E-}5 \times \text{A} \times 1540\text{ cfh} \times 105\text{ seconds}}{\text{Volume of the RB} \times 3600\text{ sec/hr}}$$

$$= 0.0011\text{A}/\text{Volume of the RB in ft}^3$$

$$= 0.0011\text{A}/1.932\text{E}6\text{ ft}^3 \text{ (Item 3.4 of Reference 2)} = 5.7\text{E-}10 \times \text{A}$$

To release the same amount of activity directly from the drywell over 105 seconds, the leakrate (in cfh) would have to be:

$$\text{Leakrate} = (5.7\text{E-}10 \times \text{A} \times 3600\text{ sec/hr} \times \text{drywell volume in ft}^3)/(\text{A} \times 105\text{ seconds})$$

$$= 1.95\text{E-}8 \times \text{drywell volume in ft}^3$$

$$= 1.95\text{E-}8 \times 159000\text{ ft}^3 = 3.1\text{E-}3\text{ cfh}$$

Results

A drywell leakrate directly to the environment which would conservatively mimic the "hold-up" model presented in Exhibit 1 is 3.1E-3 cfh.

Conclusions

Using this approach, about 1E-4 ft^3 of drywell atmosphere (3.1E-3 cfh x 105/3600 hours) is assumed to be released directly to the environment over the first 105 seconds as opposed to the four cubic feet that would actually be released (to the RB) if the leakrate were the design value of

132.5 cfh. Dilution of this four cubic feet in the secondary containment atmosphere (with a volume of about 2 million cubic feet) would amount to about a factor of 500000. Since the leakrate out of the RB, however, is a factor of 12 greater than that from the drywell (1540 cfh vs 132.5 cfh) the "effective" dilution in the RB is reduced to about a factor of 40000. Therefore, one would expect that four cubic feet of drywell atmosphere released through the RB would contain about the same amount of activity as 4/40000 cubic feet of drywell atmosphere released without benefit of mixing and dilution in the RB. This value "4/40000 cubic feet" is 1E-4 ft^3, the same "drywell volume released" value calculated above using the leakrate of 3.1E-3 cfh.

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04002H.08

CALCULATION TITLE:

"Aerosol Decontamination Factor in Main Steam Lines"

	ORIGINATOR		CHECKER		IND REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	Jun Li	9/19/95	Rudolph Sher	9/19/95	Rudolph Sher	9/19/95
1						
2						
3						
4						

REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this analysis is to model the behavior of the aerosols as they travel through the main steam lines and, therefore, to calculate the decontamination factor of the aerosols.

Methodology

The problem to be solved can be described as follows:

During a postulated DBA accident, the aerosols suspended in the drywell may be entrained in the flow that enters the main steam lines through the MSIV leakage. These aerosols will then experience removal processes, such as sedimentation, diffusion, diffusiophoresis and thermophoresis, and so on. Since the leakage flow is small but the size of a main steam line is large, the bulk flow velocity (driven by the leakage flow) in the main steam line is very small. Due to the fact that the average velocity of the aerosols entrained in the leakage flow is the same as the bulk velocity of the flow, the average residence time for particles (i.e., the time the aerosols spend within the volume the main steam lines) can be very long for any typical length of the main steam lines.

To calculate retention of aerosols in the main steam lines, the average residence time for the aerosols is determined first. Then, the removal rates of the aerosols are calculated. Finally, integration of the removal rate over the average residence time yields the amount of the aerosols removed from the total aerosols entering the main steam lines.

Assumptions

Assumption 1: Thermal hydraulic conditions in the main steam lines are assumed to be 558.5 °K in temperature and 1 atm in pressure over the time period in the accident of interest, i.e., from 0 to about 28 hours to be consistent with other aerosol removal calculations (e.g., drywell aerosol calculation in reference 1).

Justification: The initial temperature in the main steam lines is the same as that during normal operation, so it equals 558.5 °K, which is the steam saturation temperature at the RCS pressure of 1015 psia (the pressure is given in reference 2 as Item 8.9). After the MSIV closes following a postulated severe accident, the pressure in the main steam lines drops to about atmospheric pressure, while the wall temperature remains unchanged at least for a while. The temperature is expected to drop as time goes on, but the drop is ignored here since the process will be very slow due to the insulation. Ignoring the temperature drop leads to a smaller decontamination factor, as will be shown later, and is thus conservative.

Assumption 2: The gas flow in the main steam lines, which carries the aerosols, is a "plug flow" (i.e., a uniform flow along the length of the main steam line with velocity based on the volumetric flow from MSIV leakage) with the possibility of localized laminar natural circulation flow.

Justification: The possible driving forces for the gas movement in the main steam lines are the following:

- The MSIV leakage, which is 100 scfh according to Reference 2,
- Wall temperature variation along the pipe.

The limited volumetric flow from MSIV leakage leads to plug flow, since the leakage rate is small while the pipe size is large. Some of the leakage may enter the main steam line as a jet-like flow if there is a leak pathway with large enough area and a large enough aspect ratio. Otherwise, the leakage will tend to diffuse into the main steam line through multiple pathways with the total leakage flow rate less than 100 scfh.

If the leakage flow is jet-like, jet-induced vortices will occur in the immediate vicinity of the leak pathway. It is expected that these vortices will efficiently mix the incoming leakage flow with the bulk gas. If there are multiple leak paths, the leakage flow mix with the bulk gas even more efficiently. Thus, the MSIV leakage is considered to result in plug flow in the main steam line starting from the immediate vicinity of the MSIV.

Variation of the wall temperature along the pipe, on the other hand, tends to cause local circulation. Since the main steam lines are insulated, heat loss during the post accident phase will be very slow. At the same time, the pipe wall is an excellent thermal conductor, which should result in smoothing out the temperature variation along the wall, especially when the heat loss is limited. Thus, the temperature variation is unlikely to be significant, and temperature variation induced local circulation, if it exists, is unlikely to be turbulent.

Assumption 3: Aerosols in the main steam lines travel, on average, at the plug flow velocity along the axis of the pipe.

Justification: The average axial velocity of the aerosols in the main steam lines is the combination of the convective flow velocity and the axial diffusion velocity of the aerosols. In general, the axial diffusion can be ignored, because it is much smaller than the convective flow velocity. But, when the convective flow velocity is very small (e.g., the plug flow velocity), whether or not the axial diffusion velocity can be ignored needs to be examined.

Consider a cross-section in the main steam line with the aerosols well-mixed per Assumption 1. The convective aerosol mass flux across the cross-section is uc, where u is the plug flow velocity and c is the aerosol concentration. The aerosol mass flux due to the aerosol axial diffusion is $-D(\partial c/\partial x)$, where D is the diffusion coefficient for aerosols and x is along the axis of the pipe. So, the average aerosol mass flux across the cross-section is $uc - D(\partial c/\partial x)$, and the average axial velocity of aerosols at that location is $u - (D/c)(\partial c/\partial x)$.

It will be shown later that the plug flow velocity in this analysis is about a half centimeter per second ($u \approx 0.5$ cm/s). Typically, the diffusion coefficient for a 0.1 micron particle is of the order of 10^{-6} cm^2/s (see Table 2.1 on page 33 of reference 4) and the bigger the particle, the smaller the diffusion coefficient. The aerosol concentration is, at most, of the order of a few grams per cubic centimeter. So, even for an aerosol concentration gradient of

several (g/cm^3)/cm, the diffusion velocity will be 5 to 6 orders of magnitude lower than the convective velocity. It should be pointed out that the gas diffusion coefficient is about 6 orders of magnitude higher than the particle diffusion coefficient (i.e., about 1 cm^2/s, see Table A-8 on page 545 of reference 5). So, the axial diffusion velocity may not be negligible for gas transport (e.g., organic iodine) in the main steam lines.

Should any local circulation occur in the main steam lines, it would be laminar flow per Assumption 2 and will not affect the average velocity of the aerosols (which is the plug flow velocity) The circulation does not increase the average velocity of aerosols, but rather moves some particles faster than the average and moves other particles slower than average or even backwards.

Assumption 4: Gas flows in the main steam lines are not affected by the conditions in the turbine building where the out leakage of the main steam lines (or Stop Valve) is located.

Justification: If the pressure in the turbine building is higher than that in the main steam lines, gas will enter the main steam lines via Stop Valve leakage and there will be no more leakage to the turbine building. All the MSIV leakage flow will then take the drain line pathway to the condenser, which is a large holdup volume for the aerosols that are still suspended after going through removal processes in the main steam lines and the drain lines.

If the pressure in the turbine building is lower than that in the main steam lines, the leakage flow rate out of the main steam lines into the turbine building will not exceed the MSIV leakage flow rate, at least for long, as a result of mass conservation.

Even if the leak path opening in the Stop Valve were large enough so that countercurrent gas flows could occur, there is no mechanism in the turbine building to sustain such flows. Further, it would be impossible for countercurrent flows to affect gas flows over any significant portion of the main steam line.

Assumption 5: Aerosol sedimentation is considered to be the only removal mechanism for aerosols in the main steam lines.

Justification: The main steam lines are insulated; heat transfer and condensation in the main steam lines are small and thus are not considered. As a result, diffusiophoresis and thermophoresis of aerosols are ignored. As discussed above, the particle diffusion coefficient is very small, and the flow in the main steam is a plug flow. Therefore, diffusion of aerosols on to the pipe walls is also ignored. Neglecting these aerosol removal mechanisms is conservative in the calculation of the main steam line decontamination factor.

Assumption 6: Aerosol size distribution is log normal, with a geometric mean radius of 0.22 micron and a geometric standard deviation of 1.81.

Justification: As discussed in Reference 6 (page 12-13), the overwhelming majority of aerosols are observed to have a lognormal size distribution. It is also a common practice to assume such a distribution for the fission product aerosols in nuclear safety studies. A lognormal distribution is defined by the geometric mean radius and the geometric standard deviation. The values of these parameters in this calculation are based on an analysis of data from several degraded fuel experiments [reference 7]. It should be pointed out that the aerosol size distribution specified here yields a mass mean diameter of about 1.3 microns. For comparison, the mass mean diameters used in NUREG/CR-5966 [reference 8] range from 1.5 to 5.5 microns and the geometric standard deviations range from 1.6 to 3.7 (see page 84). Thus, the size distribution used in this calculation is conservative compared with reference 8.

Reference

Reference 1: PSAT 04000U.03, "Design Data Base for Application of the Revised DBA Source Term to the TVA Browns Ferry Nuclear Power Plant", Revision 0

Reference 2: PSAT 04001H.02, "Aerosol Decay Rates (Lambda) in Drywell", Revision 0

Reference 3: White, F. M., "Viscous Fluid Flow", McGraw-Hill, New York, 1974

Reference 4: S.K. Friedlander, "Smoke, Dust and Haze - Fundamentals of Aerosol Behavior", John Wiley & Sons, New York, 1977

Reference 5: J.P. Holman, "Heat Transfer", 5th Edition, McGraw-Hill, New York, 1981

Reference 6: Fuchs, N. A., "The Mechanics of Aerosols", Dovers Publications, Inc., New York, 1964

Reference 7: Polestar Memo from R. Sher to D. E. Leaver, "Aerosol Source Size Parameters", July 28, 1995

Reference 8: Powers, D. A. and Burson, S. B., "A Simplified Model of Aerosol Removal by Containment Sprays", NUREG/CR-5966, SAND92-2689, June 1993

Calculation

Polestar
PROPRIETARY

Page: 1 of 9
Rev: [✓ 0] 1 2 3 4

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04002H.09

CALCULATION TITLE:

"Elemental Iodine Filter Efficiency in Main Steam Lines"

	ORIGINATOR		CHECKER		IND. REVIEWER	
	Print/Sign	Date	Print/Sign	Date	Print/Sign	Date
REVISION: 0	David Leaver D Leaver	9/28/95	James Metcalf James Metcalf	9/28/95	James Metcalf	9/28/95
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REASON FOR REVISION:	Nonconformance Rpt
0 - Initial Issue	N/A
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Purpose

The purpose of this calculation is to determine the effective filter efficiency for elemental iodine released into the main steam lines. Elemental iodine (i.e., I_2), released from the damaged core as specified in NUREG 1465 [1], plates out on the aerosol suspended in the drywell atmosphere and is transported with the aerosol. Thus the I_2 leaks with the aerosol through the MSIVs and deposits on the steam line pipewall (with the aerosol). A fraction of this I_2 resuspends as organic iodide and is then released to the environment. This calculation will estimate the fraction of I_2 which resuspends as organic and convert this resuspension fraction to an effective filter efficiency for I_2 entering the steam lines.

Methodology

In order to determine the effective filter efficiency, a manual calculation will be performed which does the following:

- Evaluates the plateout of I_2 on aerosol.
- Compares the resuspension rate of I_2 with the fixation rate in order to determine the fraction of deposited I_2 which resuspends over time.
- Converts the resuspended fraction to a filter efficiency.

Assumptions

Assumption 1:	The I_2 is reactive and will tend to plate out on surfaces in the drywell.
Justification:	Elemental iodine is a gas at containment temperatures and is reactive with many materials [2]. It is well documented that it will tend to deposit on surfaces by chemical adsorption [3].
Assumption 2:	The resuspended I_2 is converted to organic iodide.
Justification:	According to Reference [3], resuspended I_2 can change its chemical form (conversion) to organic. For simplicity and conservatism, this conversion is assumed to be 100%.

References

1. L. Soffer et al, "Accident Source Terms for light-Water Reactor Nuclear Power Plants," NUREG 1465, February, 1995.

2. "Handbook of Chemistry and Physics," 73rd Edition, 1992-1993.

3. J. Cline, "MSIV Leakage Iodine Transport Analysis," Prepared for the U.S. Nuclear Regulatory Commission under contract NRC-03-87-029, Task Order 75, March 26, 1991.

4. N. A. Fuchs, "The Mechanics of Aerosols," Dover Publishing, 1964.

5. "Aerosol Decay Rate (Lambda) in Drywell," Polestar QA Record PSAT 04001H.02, Revision 0, September 1, 1995.

6. "Design Data Base for Application of the Revised DBA Source Term for the TVA Browns Ferry Nuclear Power Plant, " Polestar QA Record PSAT 04000U.03, Revision 1, September 22, 1995.

7. D. McNeese and A. Hoag, "Engineering and Technical Handbook," Prentice Hall, 1963.

8. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. NRC, NUREG 0800, Section 6.5.2, Revision 0.

9. "Aerosol Decontamination Factor in Main Steam Line," Polestar QA Record PSAT 04002H.08, Revision 0, September 19, 1995.

Calculation

Calculation of Plateout Area of Aerosol vs. Plateout Area of Drywell Shell, Equipment, and Structural Surfaces

Per Assumption 1., the I_2 will tend to plate out on surfaces. This calculation is to determine the relative magnitude areas of potential plate out surfaces in the drywell.

The aerosol particle surface area is estimated as follows. From Reference [4], the mass fraction for aerosols of radius r is expressed by

$$f(r)dr = \frac{1}{\ln\sigma\cdot\sqrt{2\pi}}\exp\left\{-\frac{\left[\ln r-(\ln r_g+3\ln^2\sigma)\right]^2}{2\ln^2\sigma}\right\}d\ln r$$

$$= \theta(r)d\ln r$$

The subtotal of the mass for aerosols of radius r to r + dr is

$$\Delta m = Mf(r)dr = \frac{M}{\ln\sigma\sqrt{2\pi}}\exp\left\{-\frac{\left[\ln r-(\ln r_g+3\ln^2\sigma)\right]^2}{2\ln^2\sigma}\right\}d\ln r$$

$$= M\theta(r)d\ln r$$

where the total mass of aerosols is M.

The subtotal of the volume is

$$\Delta v = \frac{\Delta m}{\rho}$$

where the volume per particle is

$$v = \frac{4}{3}\pi r^3$$

Thus the number of particles in r to r + dr is

$$N(r) = \frac{\Delta v}{v}$$

where the surface area per particle is

$$A = 4\pi r^2$$

The subtotal of surface area for aerosols in r to r + dr is

$$S = N(r).A \; = \; \frac{\Delta v}{v} 4\pi r^2 \; = \; \frac{\Delta v}{\frac{4}{3}\pi r^3} 4\pi r^2 \; = \; \frac{3\Delta v}{r}$$

$$= \frac{3\Delta m}{\rho r} = \frac{3M}{\rho r}\theta(r) d\ln r = \frac{3M\theta(r)}{\rho r^2} dr$$

Using a total aerosol mass of 12.6 kg and a particle density ρ of 3760 kg/m^3, the total surface area of the aerosol is

$$\int_0^\infty \frac{3M\theta(r)}{\rho r^2} dr = 1.87\text{E}4 \text{ m}^2 \text{ for } r_g = 0.22\ \mu\text{m and } \sigma = 1.81.$$

These values of aerosol mass, density, and size distribution are taken from Reference [5] for the conditions existing at the start of the fuel release. This is very conservative with regard to aerosol mass and surface area since the peak aerosol suspended mass will be much larger after fuel release begins.

The drywell shell, equipment, and structural surface area is estimated by summing the following: (1) calculating the horizontal surface area of the drywell shell (A_h), (2) using a multiplicative factor based on a calculation by TVA to account for additional horizontal surface area (m), (3) calculating the vertical surface area of the drywell shell (A_v), (4) applying the same multiplicative factor to the vertical surface area, and (5) calculating the downward facing surface area of the drywell shell (A_d).

Using dimensional information from Reference [6], Item 7.5, A_h can be calculated as follows:

$$A_h = (\pi)(67/2)^2 = 3526 \text{ ft}^2$$

The total horizontal surface area for sedimentation from Reference [6], Item 7.1, is 8138 ft^2. Thus the multiplicative factor is

$$m = 8138/3526 = 2.31$$

A_V can be calculated as follows:

$$A_V = A_1 + A_2$$

where A_1 is the sidewall area of the cylinder (based on a height of 55 feet per Reference [6]), and A_2 is the sidewall area of the drywell sphere (based on a height of 50 feet per Reference [6]). From Reference [6],

$$A_1 = (38.5\pi)(55) = 6652 ft^2$$

From Reference [7], the surface area of the sphere sidewall may be calculated as

$$0.5A_2 = \pi l^2 / 4 + \pi h^2$$

where l is the height of the sidewall and h is the distance which the sidewall projects out from the cylinder. From Reference [6], this is

$$0.5A_2 = (\pi / 4)(50)^2 + \pi(67/2 - 38.5/2)^2 = 2601 ft^2$$

Thus,

$$A_2 = 5203 \text{ ft}^2$$

and

$$A_V = A_1 + A_2 = 11855 \text{ ft}^2$$

The downward facing area A_d can be calculated from Reference [6] as

$$A_d = \pi(38.5/2)^2 = 1164 ft^2$$

Thus, the total plateout area of drywell surfaces including equipment and structures is

$$A_{tot} = (A_h + A_v)(m) + A_d$$

Thus,

$$A_{tot} = 36694 \text{ ft}^2 \times 0.0929 \text{ m}^2/\text{ft}^2 = 3409 \text{ m}^2$$

The minimum aerosol surface area during fuel release is 18700/3409 = ~6 times that of the drywell surfaces. Thus, the I_2 will tend to plate out almost entirely on the aerosol.

A second consideration with regard to I_2 plateout on aerosol is the fact that the aerosol gradually is removed from the drywell and thus its effective plate out area decreases with time. However, the I_2 plateout rate constant ($\sim 1.7\ hr^{-1}$ from Reference [8]) is significantly larger than the sedimentation rate constant of the aerosol (0.3 to 0.9 hr^{-1} from Reference [5]). While the aerosol sweepout rate constant is somewhat larger, sweepout will remove both aerosol and I_2. Thus the I_2 will plateout on the aerosol much faster than the aerosol itself is removed from the drywell.

On the basis of the large aerosol surface area and the fact that the I_2 will plate out on the aerosol much faster than the aerosol itself will be removed, it is reasonable to assume that essentially all of the I_2 deposits on the aerosol and thus that the I_2 behaves as an aerosol up to the point that it deposits in the steam lines.

Fraction of I_2 Resuspended from Steam Lines

Based on Reference [9], essentially all of the aerosol which leaks through the MSIVs and into the steam lines will deposit on the pipewalls. Thus the I_2 attached to this aerosol will also be deposited, and some fraction of this I_2 will resuspend. This fraction is estimated by comparing the rate constant for fixation with the rate constant for resuspension.

From Reference [3], the resuspension rate of I_2 (assumed to be resuspended as 100% organic per Assumption 2) as well as the fixation rate of I_2 varies with temperature of the steam line wall. Also from Reference [3], main steam line temperature varies from about 565 K to 400 K over the first few days after shutdown (see Exhibit 1). From Exhibit 1, it may also be seen that the average fixation rate over the first 3 days (260,000 seconds) is about 1E-5 sec^{-1}, and the average resuspension rate is about 8E-6 sec^{-1}. Thus the fraction which resuspends is something less than half of the total deposited. For conservatism, it is assumed that half of the I_2 resuspends. This resuspension will occur over a period of several days (i.e., about 90% of the resuspension occurs in the first 72 hours).

Results

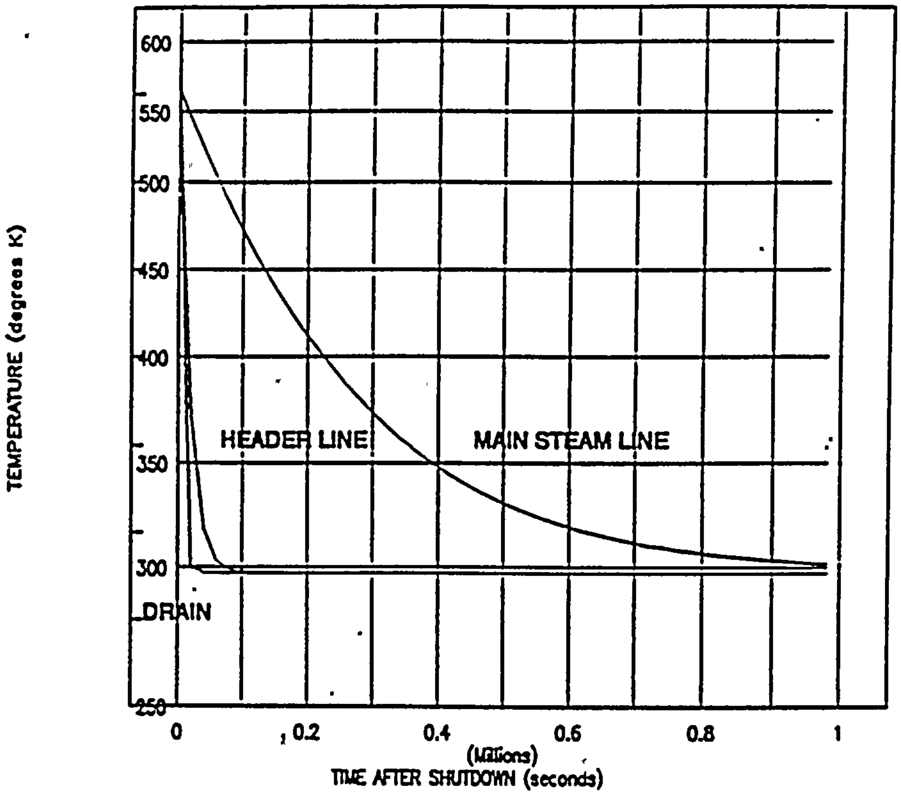
Treating the resuspension as a filtering process is conservative since the actual resuspension occurs over a several day period, whereas the filtering process assumes that the release is instantaneous at the time of deposition on the steam lines. The effective filter efficiency on the I_2 entering the steam lines is conservatively taken as 0.5. The unfiltered I_2 is then assumed to be released as organic iodide per Assumption 2.

Conclusions

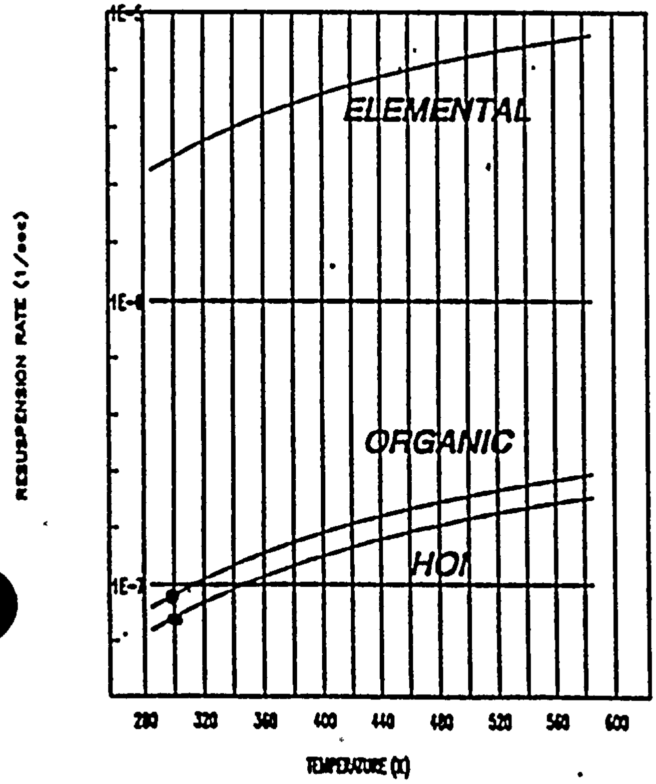
It is concluded that treating the elemental iodine as aerosol up to the point that it is deposited in the steam lines is reasonable, and that the elemental iodine entering the steam lines may be conservatively modeled with an effective filter efficiency of 50%.

Exhibit 1
(Taken from Reference [3])

TEMPERATURES OF THE MSIV LEAKAGE LINES



RESUSPENSION RATE



SURFACE FIXATION RATE

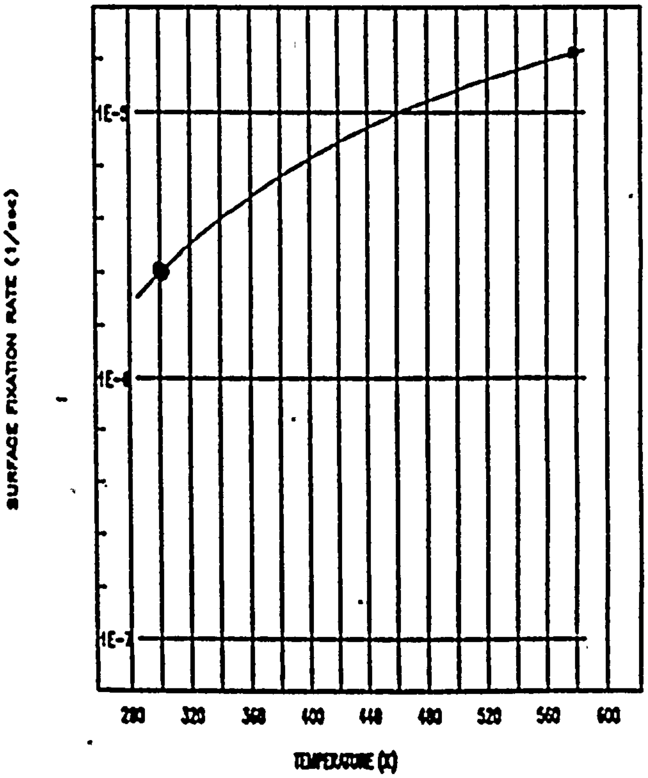


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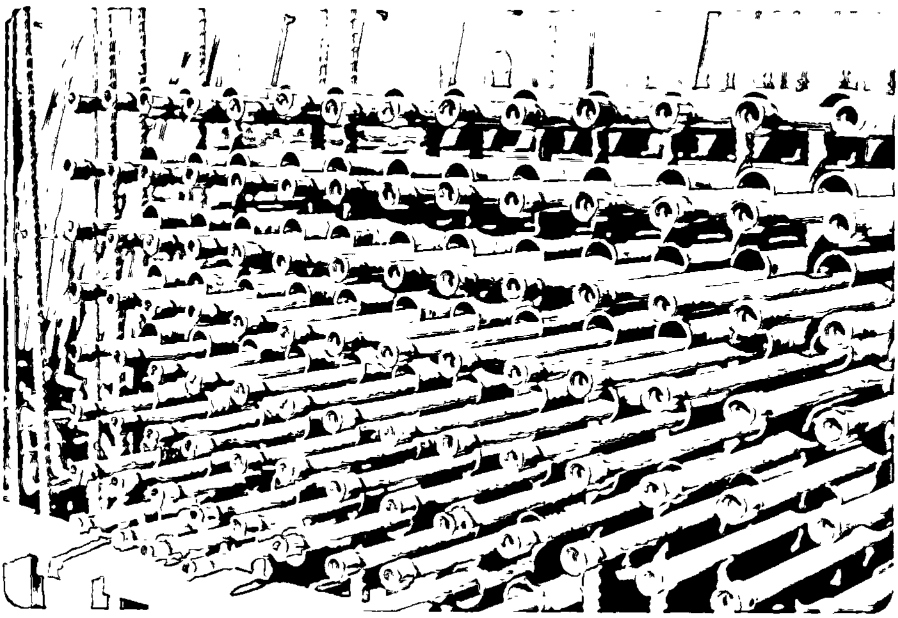
EXHIBIT A

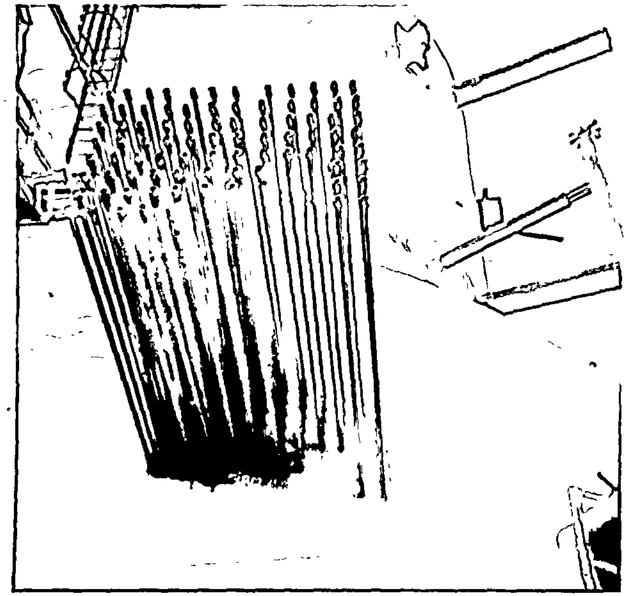


Unit No. 2 - Control Rod Hydraulic Piping

Closeup showing S.S. Piping on Carbon Steel Sleeves. 2-Vibration.

BROWNS FERRY UNIT NO. 1

SPRING HANGER INSTALLATION FOR RHR PIPING ABOVE SUPPRESSION TORUS - shows hanger tab askew, torch cut to accept pin bolt. Bolt is in flexure, over stressing base of threads.



BROWNS FERRY UNIT NO. 2 - shows stainless steel penetrations left open to contamination.

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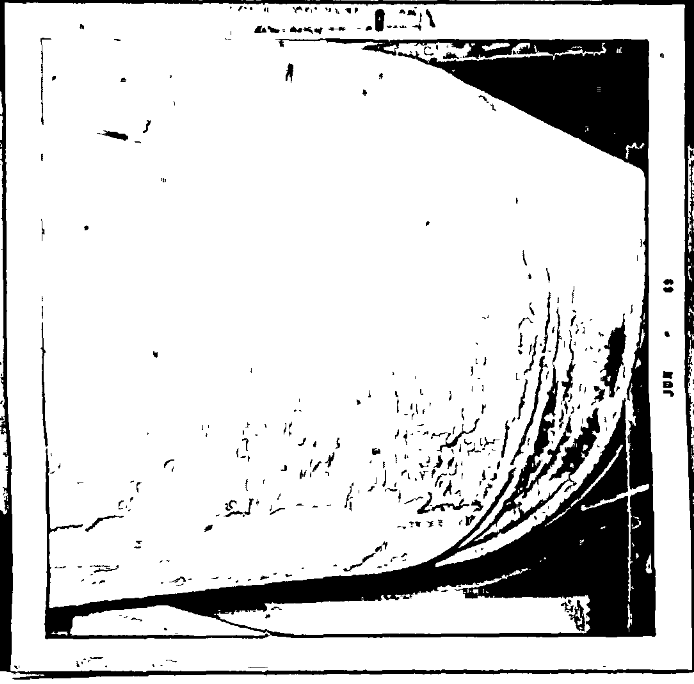
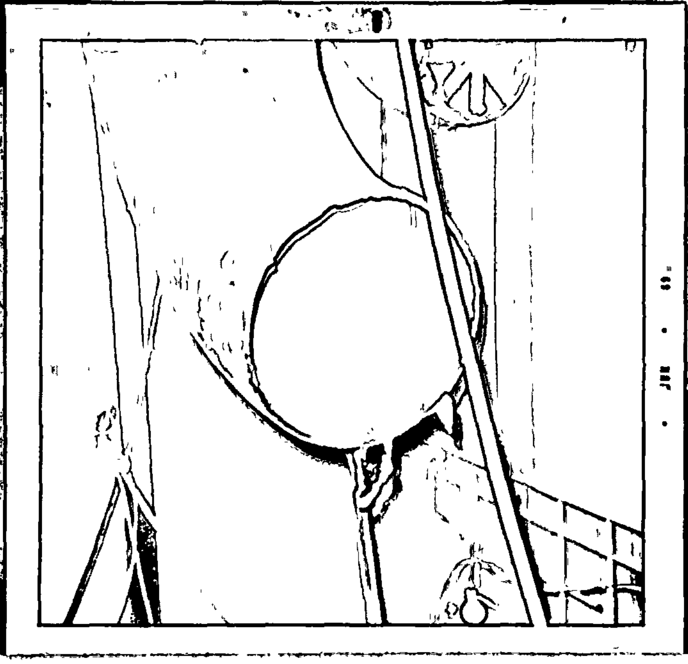
BROWNS FERRY UNIT NO. 1



20-INCH ALUMINUM HEADER FOR CONDENSATE STORAGE AND SUPPLY SYSTEM - shows rough mitre weld and lack of insulation between aluminum pipe and carbon steel hanger.



20-INCH VERTICAL RISER BELOW EXCHANGER RESIDUAL HEAT REMOVAL HEAT SYSTEM - shows contamination and mill scale.

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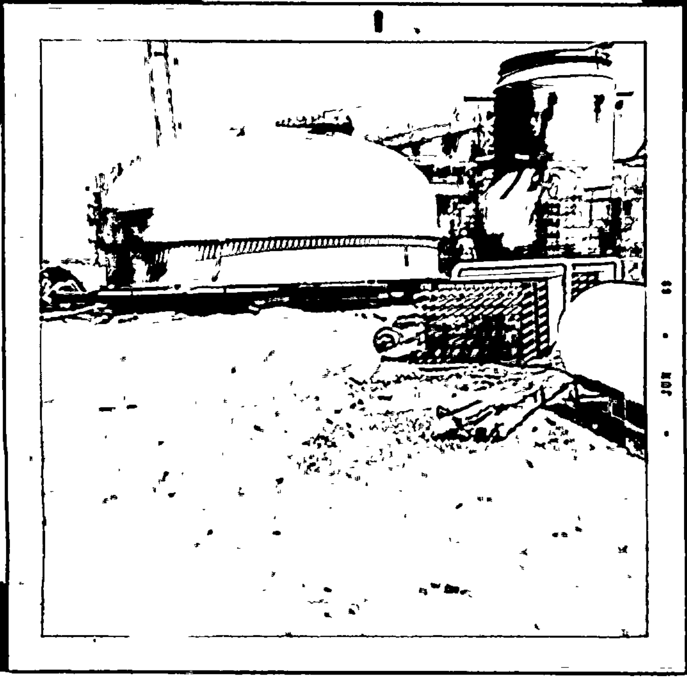
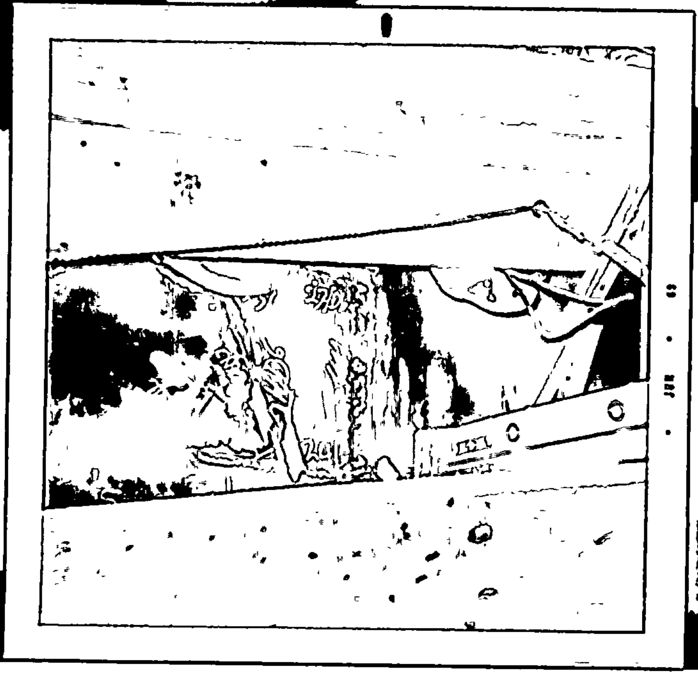
BRO FERRY UNIT NO. 3



CONTAINMENT DOME - stored on intermittent supports.

STAINLESS STEEL PENETRATION BUNDLES - pipe ends uncovered and open to contamination.



UNIT NO. 3 DRYWELL - lamination repair, second course @ approximate azimuth 220°.

Exhibit A
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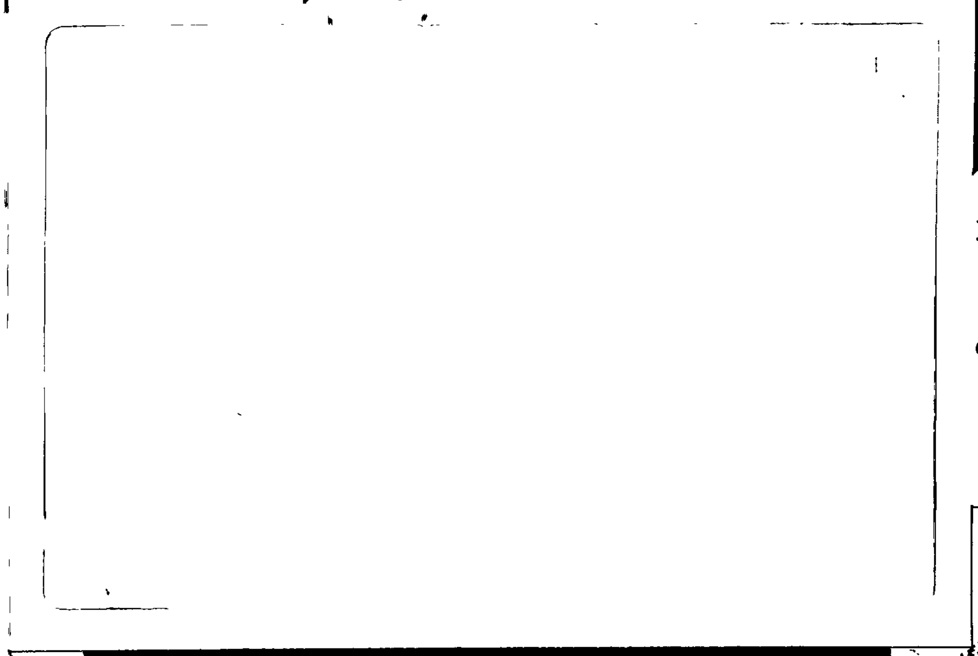

Photo #7

Mechanical Group field roll out showing all field welds of recirculation system. Chart at upper right (not fully shown) provides check off control for each weld.

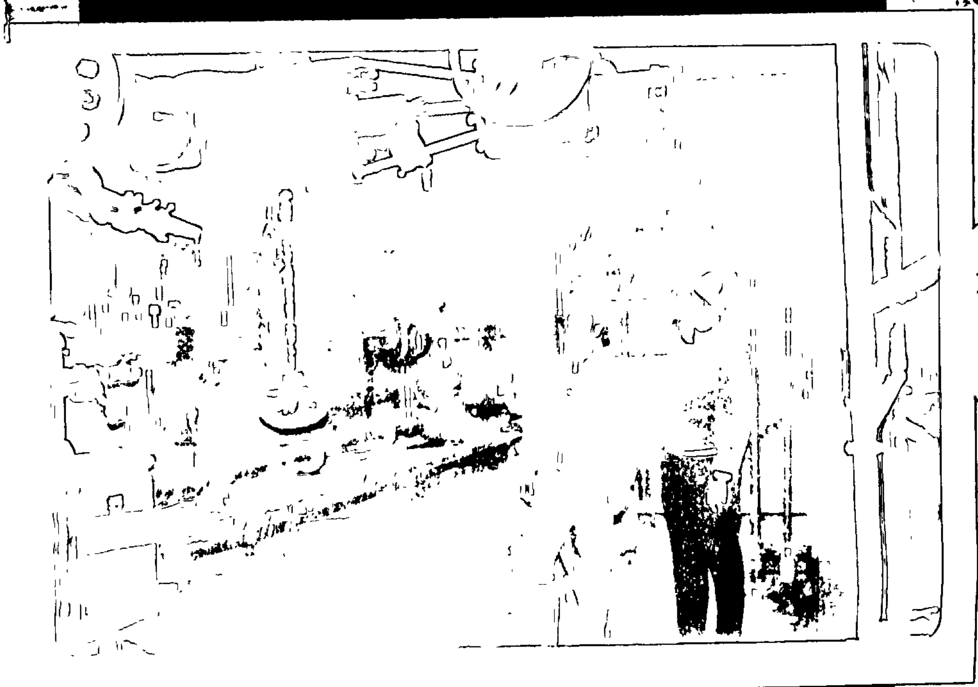

Photo #8

HPCI pump and turbine driver installation.

Photo #9

RCIC pump and turbine driver installation.

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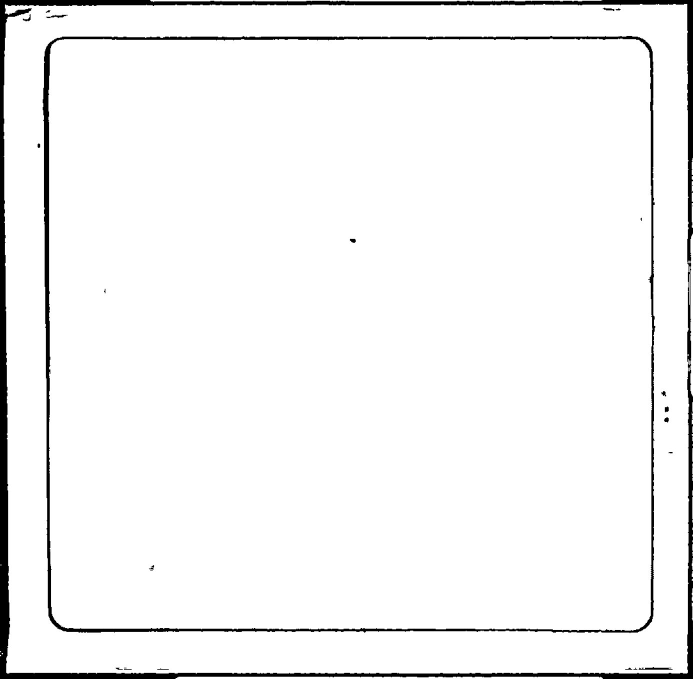

Photo #13

Crack vertical to axis of fillet weld at deflector support base, Unit #2.

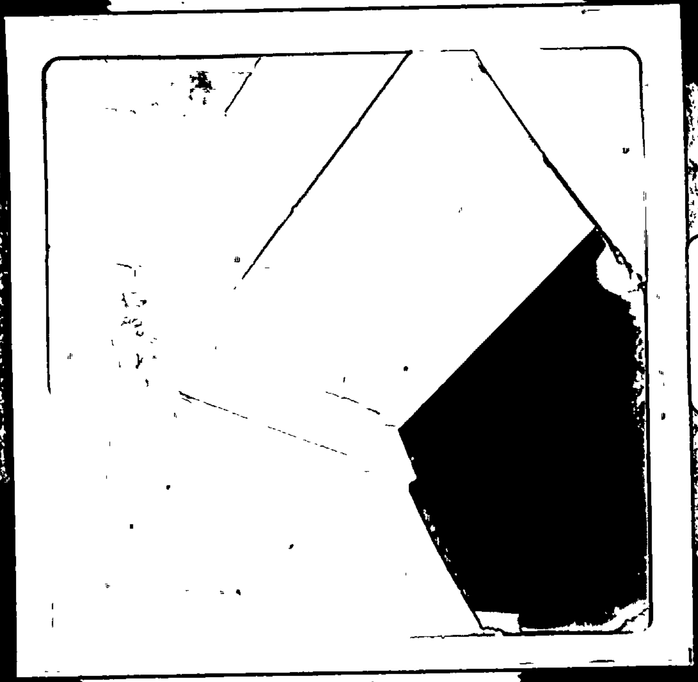

Photo #14

Two cracks, vertical to axis, in fillet weld at support base, Unit #2.

PHOTO NO. 7

E 7018 Electrodes - Low hydrogen electrodes open to atmosphere. Several bundles also observed to be like unprotected.

NOTE: 5% moisture may result in underbead cracking.

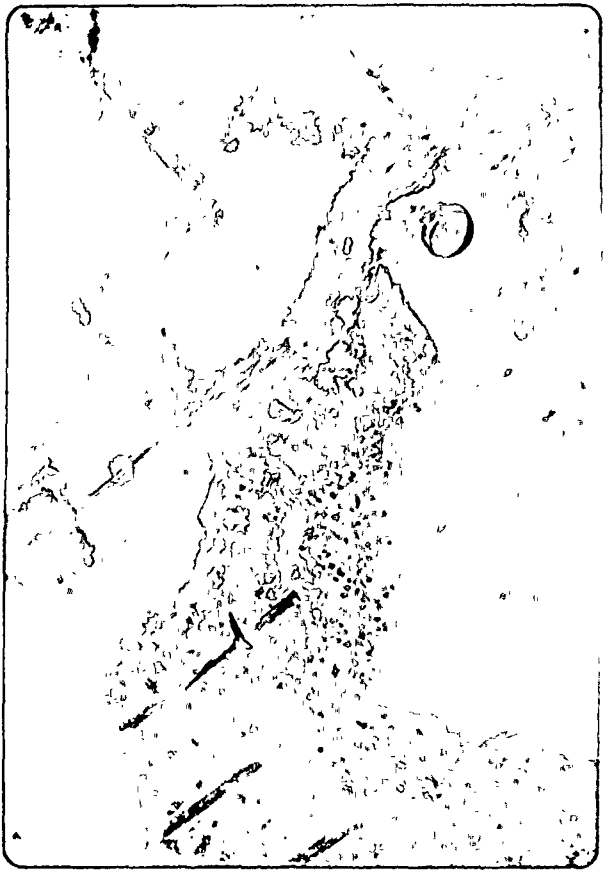

PHOTO NO. 8

Ferrophosphorus Concrete - View of ceiling and floor above where gas pressure resulted in explosion and concrete failure to depth of reinforcing steel. Cored hole for examination disclosed continued gas evolution. New break is area near hole.

Photo #10

Rough transition down stream of venturi meter in recirculation loop.

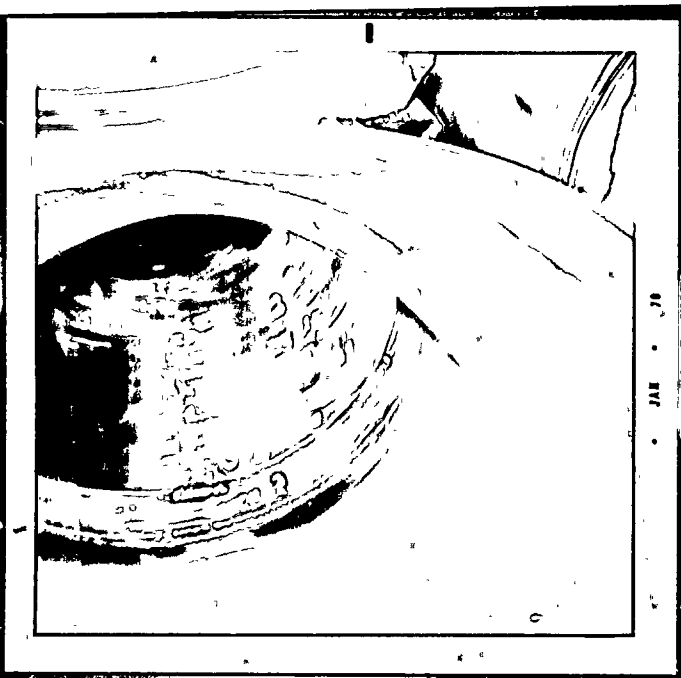

Photo #11

View of permutit venturi fitting installation from top of tee.

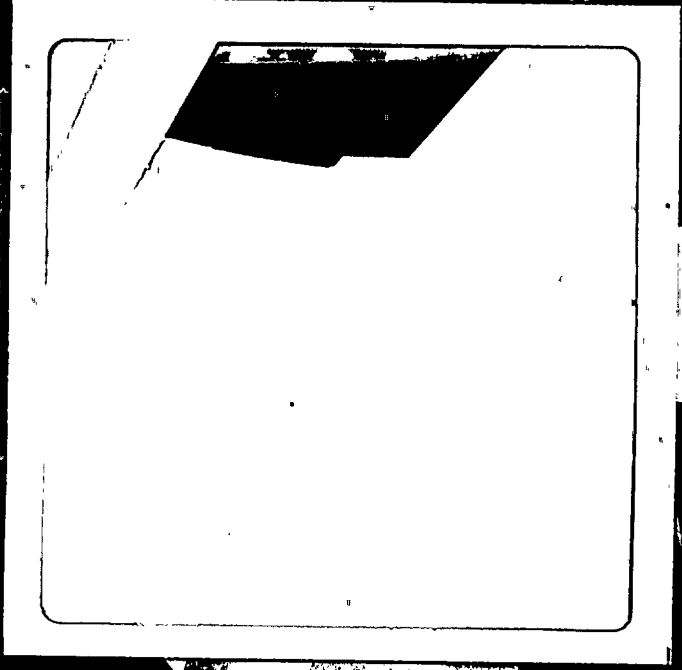

Photo #12

Crack in fillet weld at base of deflector plate support at entrance to suppression chamber duct, unit #1, Brown's Ferry.

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SINGLE-CONDUCTOR DATA AND CURRENT CARRYING CAPACITY POLYETHYLENE-INSULATED CABLE
(75°C MAXIMUM COPPER TEMPERATURE)
100 PERCENT LOAD FACTOR - 40°C AMBIENT AIR

CONDUCTOR SIZE AWG OR MCM	VOLTAGE RANGE												NOMINAL O.D.			
	0-5000			5001-8000						0-5000	5001-8000					
	NUMBER OF CABLES IN ONE CONDUIT									SINGLE CONDUCTOR IN STILL AIR*						
	1	2	3	1	2	3							600V	5000V	8000V	
14			19			23				26			.137			
12			24			28				33			.156			
10			33			37				44			.187			
8			43			46				61			.34			
6			57			65				83			.378			
4			76			85				110			.43			
2			101			111				150			.485			
1			115			127				172			.58			
1/0	167	146	133	179	161	145				202	214		.62	.870	1.05	
2/0	194	169	152	205	185	165				235	245		.66	.915	1.122	
3/0	222	194	175	235	212	187				273	284		.70	.985	1.17	
4/0	256	222	203	270	240	213				315	329		.77	1.04	1.23	
250	285	247	225	298	264	235				352	362		.88			
300	323	279	251	333	294	263				393	403		.94	1.145	1.35	
350	352	304	274	367	322	288				443	445		.99			
400	380	327	296	398	348	312				481	486		1.03	1.23	1.42	
500	432	374	336	453	395	352				546	554		1.13	1.30	1.60	
750	546	467	418	569	487	432				691	716		1.42	1.50	1.73	
1000	640	532	478	670						824	852		1.58			
1250	728			765						938	980		1.75			
1500	808			848						1032	1084					
1750	880			922						1130	1187					
2000	950			987						1220	1290					

Notes:

1. For motor circuits, use cables with ratings 125 percent of motor ratings.

2. Ratings in above table are based on AIEE-IPCEA ampacities and: (a) load factor of 100 percent which is continuous loading for a period of 8 hours or longer; (b) maximum copper temperature of 75°C with ambient temperature of 40°C; (c) 60 cycles a-c; (d) skin and proximity effect included; (e) single-conductor cable nonleaded or with open-circuited sheath.

3. For cable sizes larger than 500 mcm consult technical engineer.

4. Cable sizes shown above are determined by current carrying capacity only and should be increased where necessary to maintain suitable voltage regulation.

5. Control cables made up of No. 14, No. 12, or No. 10 wire are not ordinarily subject to temperature rise limitations. Maximum continuous current per conductor should not exceed 12 amperes for No. 14, 15 amperes for No. 12, or 22 amperes for No. 10.

CORRECTION FACTORS FOR AMBIENT TEMPERATURES

For 20°C multiply table values by 1.25
For 30°C multiply table values by 1.13
For 50°C multiply table values by 0.85

*Correction factor for cables in trays - 0.80.

REV NO	DATE	MADE	CHKD	SUPV	INSP	SUBM	RECM

DSGN. J.D.
DRWN.
CHKD.
TRCD.
COMP.
SUPV.
INSP. J.D.
ENGINEER

ELECTRICAL STANDARDS

CONDUCTOR DATA-CUR. CARRYING
CAPACITY POLYETHYLENE INSL
CABLE 0-8,000V

DESIGN AND DRAFTING STANDARDS
TENNESSEE VALLEY AUTHORITY
DIVISION OF DESIGN

SUBMITTED L R Sellers
RECOMMENDED
APPROVED C. McCord

KNOXVILLE | 9-26-63 | G | E | 0 | 30A1020R0.

Exhibit C
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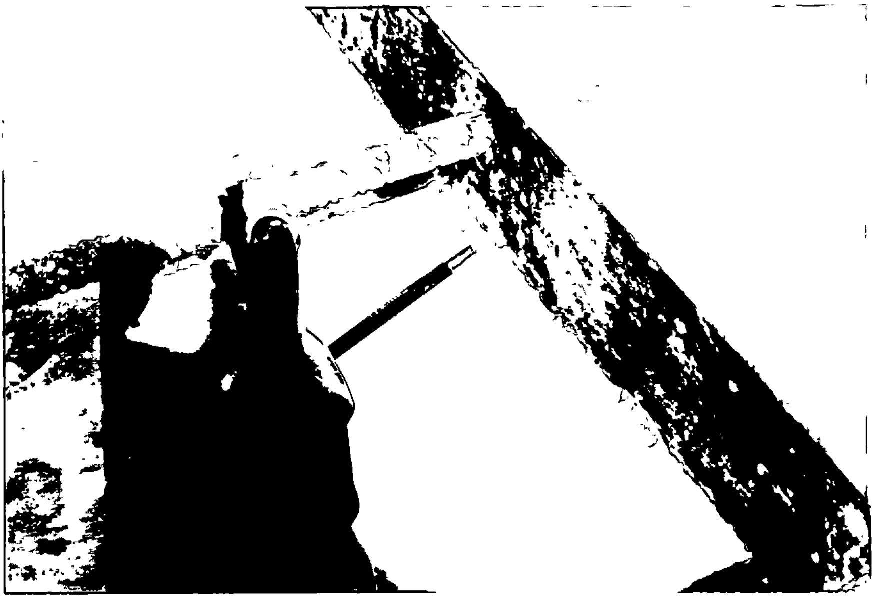

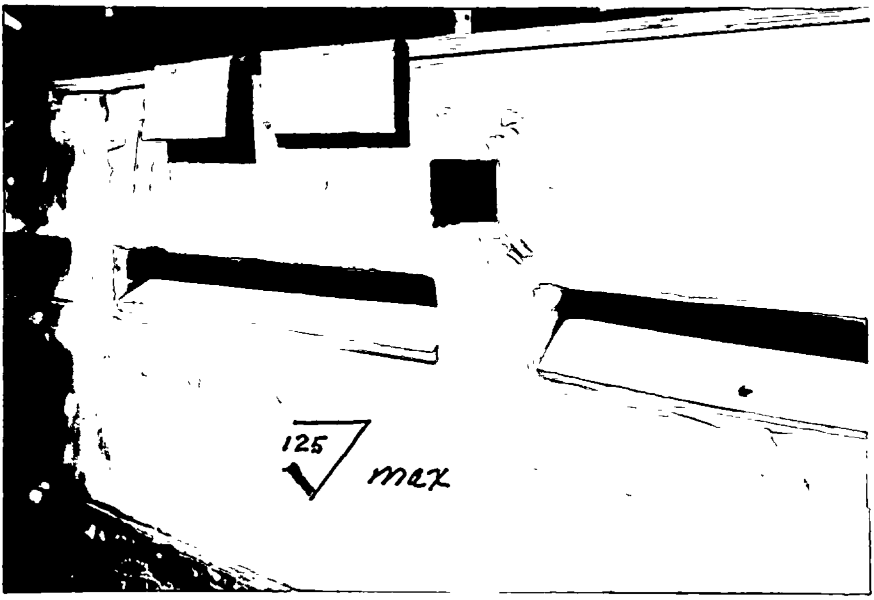
125
max