PSAT 04000U.04

Attachment 3

PSAT Calculation 04011H.01

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

CALCULATION TITLE PAGE

CALCULATION NUMBER: PSAT 04011H.01

CALCULATION TITLE:

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

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0 - Initial Issue	N/A
1 - To add Appendix B and Appendix C and to correct cover sheet for Appendix A	N/A
2 - To update Reference 3 and expand Reference 9	N/A
3	
4	

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Purpose

The purpose of this calculation is to specify the volumetric exchange rates between the Browns Ferry drywell and the torus during two periods of the problem: during the fission product release (gap release phase from 30 seconds to 1830 seconds and early in-vessel release phase from 1830 seconds to 7230 seconds - see Table 3.6 of Reference 1) and after the fission product release phase (7230 seconds until 30 days which is the end of the dose calculation interval from Reference 2). During (and immediately after) the fission product release phase the flow is only from the drywell to the torus and may be referred to as the "sweep-out" rate.

Methodology

In order to specify the volumetric sweep-out rate, it is necessary to know the quantity of water remaining in the vessel after the DBA blowdown, the thermodynamic state in the drywell, and the rate at which steam is produced from the core debris in-vessel up to and including the point in time where the core-debris quench is complete (assuming that to be shortly after 7230 seconds, the end of the in-vessel release phase). Beyond 7230 seconds + the reflood time, the containment is assumed to be well-mixed, but a mixing rate must be specified to reflect that assumption.

A manual calculation is shown below which:

- Quantifies the minimum water mass remaining in the vessel after DBA blowdown,
- Determines a minimum steaming rate for that remaining water, and
- Calculates the volumetric flowrate rate (drywell to torus) that corresponds to that steaming rate and to the final quench of the core debris.

Assumptions

Assumption 1: Reactor vessel reflood occurs at 7230 seconds, terminating the release and quenching the core debris.

Justification: This assumption reflects the position that Reference 3 takes with respect to the release phases of Reference 1. Reference 3 references an NRC position taken on the advanced light water reactors in Reference 4, which is:

> "In a forthcoming paper, the NRC staff will indicate that for evaluation of design basis accidents (DBA) for evolutionary and passive light-water reactor designs, only the releases associated with the gap and early in-vessel release phases will be used. The inclusion of the ex-vessel and late in-vessel releases are considered to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions."

This NRC position, as extended to operating reactors by Reference 3, means that vessel failure is not to be included in the DBA. This position also implies, then, that debris coolability must be re-established at about the time of the end of the in-vessel release phase; otherwise, reactor vessel failure would likely follow.

Assumption 2: Containment is well-mixed following the core debris quench at 7230 seconds + the time to reflood.

Justification: Once the core debris is quenched in-vessel, the production of steam and non-condensible hydrogen will cease. Steam condensation in the drywell (in particular, if drywell sprays are actuated) will cause a return of non-condensibles and radioactivity from the torus airspace to the drywell. Since the details of the primary containment thermal-hydraulic conditions during the remainder of the thirty day dose calculation interval are not known precisely, it is reasonable to effectively consider a "one control volume" containment; i.e., a containment that is well-mixed. This is consistent with current practice.

Assumption 3: Following the DBA (recirculation suction large break LOCA) the water mass remaining in the vessel is that corresponding to coolant at operating conditions in the volume below the bottom of active fuel, depressurized at constant enthalpy to atmospheric pressure with steam being released from the vessel.

Justification: This assumption yields a conservatively small value for the water mass remaining in the bottom of the vessel after blowdown. All water above the bottom of the core is assumed to be removed at its operating state with no change in phase and no liquid remaining. Then, the remaining coolant is assumed to flash all the way down to atmospheric pressure. In reality, coolant would flash throughout the vessel as the vessel depressurizes, leaving more liquid in the bottom of the vessel then the above assumption would permit. Moreover, the coolant would only flash down to a pressure corresponding to that of the containment which would be greater than atmospheric pressure.

While it is true that the volume described above includes some of the annulus between the vessel and the lower shroud that would be drained by the recirculation break, it does <u>not</u> include the jet pumps up to their inlets and the corresponding volume within the core. Therefore, it is a conservative estimate of the volume that could remain water-filled with a recirculation suction line broken.

Assumption 4: In order to calculate the steaming rate from the core debris, it is assumed that the fraction of the core participating in the boil-off of the water mass remaining in the bottom of the vessel increases uniformly from zero at 1830 seconds (end of the gap release phase) to 50% at 7230 seconds (end of the in-vessel release phase).

Justification: This assumption is based in part on Assumption 1. At the end of the in-vessel release all of the core debris will be quenched, both that which has relocated to the lower part of the vessel and that remaining in the original core region. For conservatism, the debris remaining in the core region is neglected in the calculation of the steaming rate during core degradation; only the assumed 50% of the core debris which relocates to the lower part of the vessel and its interaction with the residual water (Assumption 3) is included in the quantification of the steam production during the in-vessel release phase.

Assumption 5: The exchange rate between the drywell and the torus is assumed to be constant during the release phase (up to 7230 seconds).

Justification: This assumption is slightly non-conservative because it overestimates the removal rate from the drywell early in the release phase. However, it does simplify the analysis; and for relatively low removal rates (of the order of one per hour) the underestimate of the late removal compensates nearly completely for the overestimate of the early removal. A further demonstration of the adequacy of this assumption is presented in Appendix A.

Assumption 6: The final core debris quench requires the time it takes minimum ECCS (one core spray pump) to refill the core region, and it involves only the energy stored in the one-half of the core debris assumed not to relocate to the lower part of the vessel.

Justification: Leaving one-half the core uncovered for a period of 7230 seconds (less the blowdown/core uncovery time) results in core debris left in the core region with significant stored energy. The restoration of minimum ECCS will remove this stored energy at a rate determined by the coolant injection rate (drawn from the suppression pool) and the rising water level (reflood rate). To determine the reflood rate, the ECCS injection rate must be reduced by the rate of steam production. The rate of steam production in this analysis corresponds to a low estimate of stored energy in only one-half of the core debris.

Reference 5 indicates that the sweep-out rate corresponding to the final core debris quench would be expected to be of the order of 10 drywell volumes per hour.

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: Leaver, D. E. and Metcalf, J. E., "Generic Framework for Application of Revised Source Term to Operating Plants", EPRI Interim Report TR-105909, EPRI Research Project 4080-2, November 1995

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

Reference 5: Leaver, D. E., et al., "Licensing Design Basis Source Term Update for the Evolutionary Advanced Light Water Reactor", DOE/ID-10298, September, 1990

Reference 6: 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 7: Babcock and Wilcox, Steam, Its Generation and Use, New York, 1963

Reference 8: McAdams, Heat Transmission, McGraw-Hill, New York, 1942

Reference 9: NRC Generic Letter 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities - 10CFR50.54(f)", November 23, 1988

Calculation

Minimum mass of water remaining in vessel post-DBA blowdown

Reference 6 provides the following:

- Volume in-vessel, below BAF - 4100 ft^3 (Item 3.26)
- Reference pressure for determination of coolant mass - 1015 psia (Item 8.9)
- Liquid specific volume at reference pressure - 0.02166 ft^3/lbm (Item 8.10)

The mass assumed to remain in the vessel prior to the flash is:

Mass = 4100 ft^3 / 0.02166 ft^3/lbm = 1.89E5 lbm

The enthalpy for saturated water at 1015 psia =

h_f @ 1000 psia + 0.15 (h_f @ 1100 psia - h_f @ 1000 psia) =

542.4 BTU/lbm + 0.15(15 BTU/lbm) = 544.7 BTU/lbm   based on Exhibit 1 data from Reference 7

Using this enthalpy, the fraction flashed to steam at constant enthalpy, x, can be determined from the following expression (evaluated using data from Exhibit 1):

x(h_g @ 14.7 psia) + (1-x)(h_f @ 14.7 psia) = 544.7 BTU/lbm

1150.4x + 180.1 - 180.1x = 544.7

970.3x = 364.6

x = 0.38

The minimum mass of water remaining in the vessel after the assumed flash is:

(1-x)(1.89E5 lbm) = (0.62)(1.89E5) = 1.17E5 lbm

which is based on Assumption 3.

Exhibit 1

TABLE 2. SATURATION: PRESSURES

Abs Press. Lb/Sq In. p	Temp F t	Specific Volume Sat. Liquid v_f	Specific Volume Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat. Vapor h_g	Entropy Sat. Liquid s_f	Entropy Evap s_{fg}	Entropy Sat. Vapor s_g	Internal Energy Sat. Liquid u_f	Internal Energy Evap u_{fg}	Internal Energy Sat. Vapor u_g	Abs Press. Lb/Sq In. p
1.0	101.74	0.01614	333.6	69.70	1036.3	1106.0	0.1326	1.8456	1.9782	69.70	974.6	1044.3	1.0
2.0	126.08	0.01623	173.73	93.99	1022.2	1116.2	0.1749	1.7451	1.9200	93.98	957.9	1051.9	2.0
3.0	141.48	0.01630	118.71	109.37	1013.2	1122.6	0.2008	1.6855	1.8863	109.36	947.3	1056.7	3.0
4.0	152.97	0.01636	90.63	120.86	1006.4	1127.3	0.2198	1.6427	1.8625	120.85	939.3	1060.2	4.0
5.0	162.24	0.01640	73.52	130.13	1001.0	1131.1	0.2347	1.6094	1.8441	130.12	933.0	1063.1	5.0
6.0	170.06	0.01645	61.98	137.96	996.2	1134.2	0.2472	1.5820	1.8292	137.94	927.5	1065.4	6.0
7.0	176.85	0.01649	53.64	144.76	992.1	1136.9	0.2581	1.5586	1.8167	144.74	922.7	1067.4	7.0
8.0	182.86	0.01653	47.34	150.79	988.5	1139.3	0.2674	1.5383	1.8057	150.77	918.4	1069.2	8.0
9.0	188.28	0.01656	42.40	156.22	985.2	1141.4	0.2759	1.5203	1.7962	156.19	914.6	1070.8	9.0
10	193.21	0.01659	38.42	161.17	982.1	1143.3	0.2835	1.5041	1.7876	161.14	911.1	1072.2	10
14.696	212.00	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566	180.02	897.5	1077.5	14.696
15	213.03	0.01672	26.29	181.11	969.7	1150.8	0.3135	1.4415	1.7549	181.06	896.7	1077.8	15
20	227.96	0.01683	20.089	196.16	960.1	1156.3	0.3356	1.3962	1.7319	196.10	885.8	1081.9	20
30	250.33	0.01701	13.746	218.82	945.3	1164.1	0.3680	1.3313	1.6993	218.73	869.1	1087.8	30
40	267.25	0.01715	10.498	236.03	933.7	1169.7	0.3919	1.2844	1.6763	235.90	856.1	1092.0	40
50	281.01	0.01727	8.515	250.09	924.0	1174.1	0.4110	1.2474	1.6585	249.93	845.4	1095.3	50
60	292.71	0.01738	7.175	262.09	915.5	1177.6	0.4270	1.2168	1.6438	261.90	836.0	1097.9	60
70	302.92	0.01748	6.206	272.61	907.9	1180.6	0.4409	1.1906	1.6315	272.38	827.8	1100.2	70
80	312.03	0.01757	5.472	282.02	901.1	1183.1	0.4531	1.1676	1.6207	281.76	820.3	1102.1	80
90	320.27	0.01766	4.896	290.56	894.7	1185.3	0.4641	1.1471	1.6112	290.27	813.4	1103.7	90
100	327.81	0.01774	4.432	298.40	888.8	1187.2	0.4740	1.1286	1.6026	298.08	807.1	1105.2	100
120	341.25	0.01789	3.728	312.44	877.9	1190.4	0.4916	1.0962	1.5878	312.05	795.6	1107.6	120
140	353.02	0.01802	3.220	324.82	868.2	1193.0	0.5069	1.0682	1.5751	324.35	785.2	1109.6	140
160	363.53	0.01815	2.834	335.93	859.2	1195.1	0.5204	1.0436	1.5640	335.39	775.8	1111.2	160
180	373.06	0.01827	2.532	346.03	850.8	1196.9	0.5325	1.0217	1.5542	345.42	767.1	1112.5	180
200	381.79	0.01839	2.288	355.36	843.0	1198.4	0.5435	1.0018	1.5453	354.68	759.0	1113.7	200
250	400.95	0.01865	1.8438	376.00	825.1	1201.1	0.5675	0.9588	1.5263	375.14	740.7	1115.8	250
300	417.33	0.01890	1.5433	393.84	809.0	1202.8	0.5879	0.9225	1.5104	392.79	724.3	1117.1	300
350	431.72	0.01913	1.3260	409.69	794.2	1203.9	0.6056	0.8910	1.4966	408.45	709.6	1118.0	350
400	444.59	0.0193	1.1613	424.0	780.5	1204.5	0.6214	0.8630	1.4844	422.6	695.9	1118.5	400
450	456.28	0.0195	1.0320	437.2	767.4	1204.6	0.6356	0.8378	1.4734	435.5	683.2	1118.7	450
500	467.01	0.0197	0.9278	449.4	755.0	1204.4	0.6487	0.8147	1.4634	447.6	671.0	1118.6	500
550	476.93	0.0199	0.8422	460.8	743.1	1203.9	0.6608	0.7934	1.4542	458.8	659.4	1118.2	550
600	486.21	0.0201	0.7698	471.6	731.6	1203.2	0.6720	0.7734	1.4454	469.4	648.3	1117.7	600
700	503.10	0.0205	0.6554	491.5	709.7	1201.2	0.6925	0.7371	1.4296	488.8	627.5	1116.3	700
800	518.23	0.0209	0.5687	509.7	688.9	1198.6	0.7108	0.7045	1.4153	506.6	607.8	1114.4	800
900	531.98	0.0212	0.5006	526.6	668.8	1195.4	0.7275	0.6744	1.4020	523.1	589.0	1112.1	900
1000	544.61	0.0216	0.4456	542.4	649.4	1191.8	0.7430	0.6467	1.3897	538.4	571.0	1109.4	1000
1100	556.31	0.0220	0.4001	557.4	630.4	1187.8	0.7575	0.6205	1.3780	552.9	553.5	1106.4	1100
1200	567.22	0.0223	0.3619	571.7	611.7	1183.4	0.7711	0.5956	1.3667	566.7	536.3	1103.0	1200
1300	577.46	0.0227	0.3293	585.4	593.2	1178.6	0.7840	0.5719	1.3559	580.0	519.4	1099.4	1300
1400	587.10	0.0231	0.3012	598.7	574.7	1173.4	0.7963	0.5491	1.3454	592.7	502.7	1095.4	1400
1500	596.23	0.0235	0.2765	611.6	556.3	1167.9	0.8082	0.5269	1.3351	605.1	486.1	1091.2	1500
2000	635.82	0.0257	0.1878	671.7	463.4	1135.1	0.8619	0.4230	1.2849	662.2	403.4	1065.6	2000
2500	668.13	0.0287	0.1307	730.6	360.5	1091.1	0.9126	0.3197	1.2322	717.3	313.3	1030.6	2500
3000	695.36	0.0346	0.0858	802.5	217.8	1020.3	0.9731	0.1885	1.1615	783.4	189.3	972.7	3000
3206.2	705.40	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580	872.9	0	872.9	3206.2

Minimum steaming rate for the remaining water

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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:

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

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

$$v_g = 10.2\ ft^3/lbm$$

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.

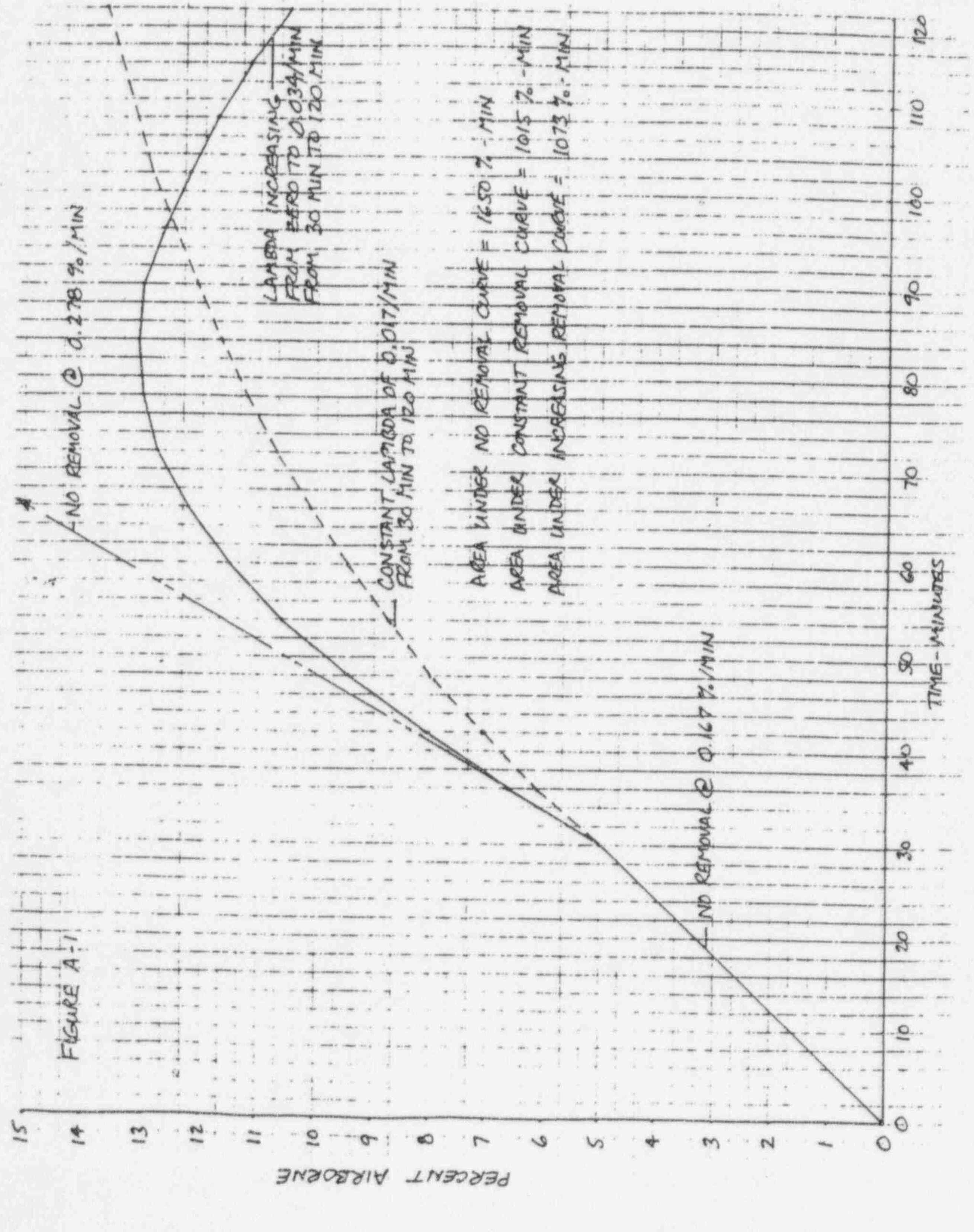
FIGURE A-1
NO REMOVAL @ 0.278 %/MIN
CONSTANT LAMBDA OF 0.017/MIN FROM 30 MIN TO 120 MIN
LAMBDA INCREASING FROM ZERO TO 0.034/MIN FROM 30 MIN TO 120 MIN
AREA UNDER NO REMOVAL CURVE = 1650 %-MIN
AREA UNDER CONSTANT REMOVAL CURVE = 1015 %-MIN
AREA UNDER INCREASING REMOVAL CURVE = 1073 %-MIN
NO REMOVAL @ 0.167 %/MIN
TIME-MINUTES
PERCENT AIRBORNE

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)
θ = 0.5 hours (30 minutes - time before start of bulk of fission product release)
α = 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.

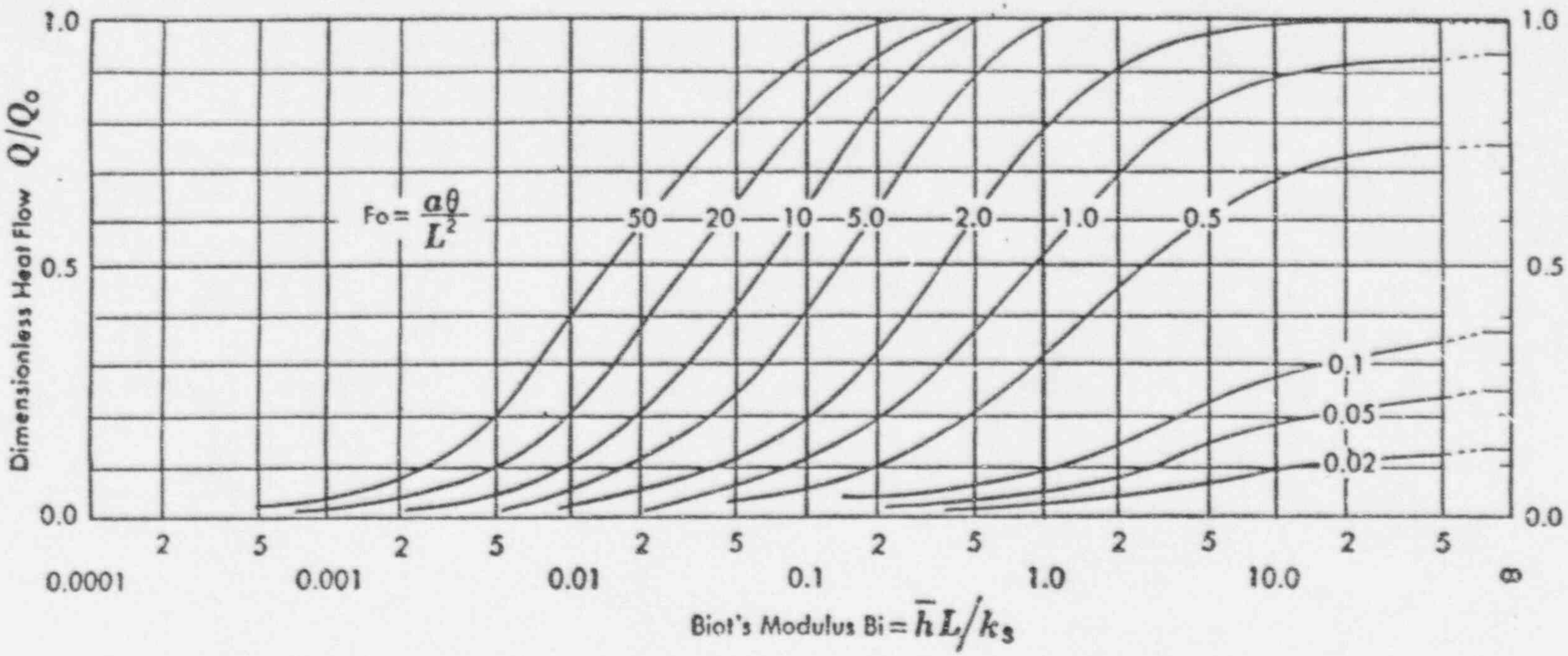

FIG. 4-9. Dimensionless heat flow to or from a wall subjected to a sudden change in environmental temperature.

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
Reactor Core		
Thermal Power (Mwt)	3,293	3,293
Number of Fuel Assemblies	764	764
Number of Control Rods	185	185
Reactor Vessel		
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
RHR System		
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
RHR Service Water System		
Number of Pumps	3	8
Flow Rate per Pump (gpm)	4,666	4,500
Core Cooling Systems		
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
Containment		
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 Type of Reactor Type of Containment	Peach Bottom BWR/4 Mark I	Browns Ferry BWR/4 Mark I
Secondary Containment		
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
Turbine Building		
Volume (ft^3)	2,100,000	5,700,000

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