AUDIT CALCULATIONS FOR A STATION BLACKOUT SEQUENCE IN THE SURRY FACILITY

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January 10, 1985

Prepared for

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under Contract No. DE-AC02-76CH00016
FIN A-

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ABSTRACT

Calculations have been performed for a station blackout sequence in the Surry Plant using the suite of severe accident phenomenology codes developed for the Accident Source Term Program Office (ASTPO), RES/NRC. This calculational effort has demonstrated that the ASTPO suite of codes can be exported to an independent organization. A station blackout sequence at Surry was originally calculated by staff at BCL and SNL using the ASTPO codes and reported in Volume V of BMI-2104. The purpose of the present BNL calculations is to provide an independent audit of the BMI-2104 results. Hence, the full suite of ASTPO codes have been obtained from BCL and SNL and made operational on the BNL computing system. No modifications were made to the codes other than those necessary to make them operational at BNL. Code input and output parameters used in BMI-2104 were reviewed in detail and several inconsistencies related to data transfer between codes were found. Some difficulties were experienced in the use of certain codes. However, a self-consistent station blackout sequence was calculated, and good agreement was obtained between BMI-2104 and the audit calculation.

ACKNOWLEDGEMENTS

The quality assurance team is indebted to R. A. Bari (BNL), J. Rosenthal (NRC), J. Mitchell (NRC), and J. Read (NRC) for many helpful discussions and guidance while carrying out this work and preparing the document. The expertise and help of Y. Sanborn (BNL) and W. Bornstein (BNL) in carrying out the computer calculations is gratefully acknowledged. Thanks are due to T. Rowland for her preparation of the document on short notice.

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1. INTRODUCTION

1.1 Background

During the last several years the NRC has sponsored research related to severe accidents in Light Water Reactors (LWRs). In particular, the Accident Source Term Program Office (ASTPO) RES/NRC sponsored the development of a suite of severe accident phenomenology codes at BCL and SNL. These codes are intended to describe how a nuclear reactor core might degrade without adequate cooling and hence release radioactive fission products. The codes also follow the subsequent transport of the fission products from the damaged core to the environment if the containment fails or is bypassed. These codes therefore focus on the release and transport of fission products and were applied to model selected severe accidents for six representative reactor designs. The results of this code application effort is reported in BMI-2104.[1] Note that other activities sponsored by the NRC related to determining containment loads and containment performance during severe accidents are reported separately[2,3] and were not taken into account in Reference (1).

The methodology described in Reference (1) has received extensive peer review over the last several months and is also under review by the American Physical Society (APS). As a result of questions raised during the APS review a meeting[4] was held at NRC to develop an appropriate response to the questions. Part of this response was to demonstrate that the ASTPO suite of severe accident codes could be exported to an independent organization and that (by using similar input parameters and intercode data transfer) similar results to those reported in BMI-2104 could be obtained. BNL was selected by the NRC to be the independent organization and this informal report documents the results of this effort.

1.2 Objective and Scope of Audit Calculation

This effort was performed under a very severe time constraint. The meeting which initiated this effort was held on November 20, 1984. A station blackout sequence at Surry was analyzed using the full suite of ASTPO codes. The objective of the effort is limited to demonstrating that each of the codes used in BMI-2104 can be made operational at BNL and that the results in BMI-2104 can be reproduced. However, we did have sufficient time to verify the appropriateness of code input parameters and intercode data transfer. This is therefore a limited quality assurance (QA) audit of the results in BMI-2104. We have not had sufficient time to QA the codes and in fact we were specifically requested to make no modifications to them and to set the internal options (related to alternative models) as in BMI-2104. Validation efforts are described elsewhere. A number of inconsistencies related to data transfer between codes were found and these are discussed in detail later in this informal report.

1.3 Calculational Methods

The codes used to analyze the station blackout sequence are identified in Figure 1.1, which indicates the relationship between the codes and also provides a brief description of the purpose of each code. The MARCH code calculates in-vessel core degradation (BOIL subroutine), vessel failure (HEAD subroutine), ex-vessel core debris/water interactions (INTER subroutine). All of these subroutines feed the containment building response model (MACE subroutine). The core temperature history, as predicted by BOIL, is fed to the CORSOR code to calculate in-vessel fission product release. Core outlet gas and steam flow rates, as predicted by BOIL, are fed to the MERGE code to

calculate primary system thermal hydraulics. Output from CORSOR and MERGE are used in the TRAP-MELT code to predict primary system transport of fission products. Fission product output from TRAP-MELT feeds the NAUA code, which calculates fission product transport in the containment building. Note that NAUA gets thermal hydraulic data from the MACE subroutine of MARCH. Inconsistency between these codes was noted as part of the initial peer review of BMI-2104. The flow rate predicted by PRIMP of MARCH is not consistent with the flow rate predicted by MERGE.

After head failure (HEAD subroutine) MARCH can model core debris/water (HOTDROP) or core debris/concrete (INTER) interactions depending on conditions in the reactor cavity beneath the reactor vessel. The thermal/hydraulic conditions in containment depend strongly on the mode of ex-vessel core debris interactions. If extensive core/concrete interactions occur then the associated fission product release must also be calculated, which is done by using the CORCON code (core debris/concrete interactions) together with the VANESA code (fission product release from ex-vessel interactions). The user of the codes should explicitly ensure that the start of core/concrete interactions predicted in the MARCH code is consistent with start of interactions assumed in the CORCON/VANESA codes. In addition, the user must be mindful of the inconsistency of using INTER to calculate the containment response (MACE) and CORCON/VANESA to calculate fission product release. This was also noted as part of the peer review of BMI-2104.

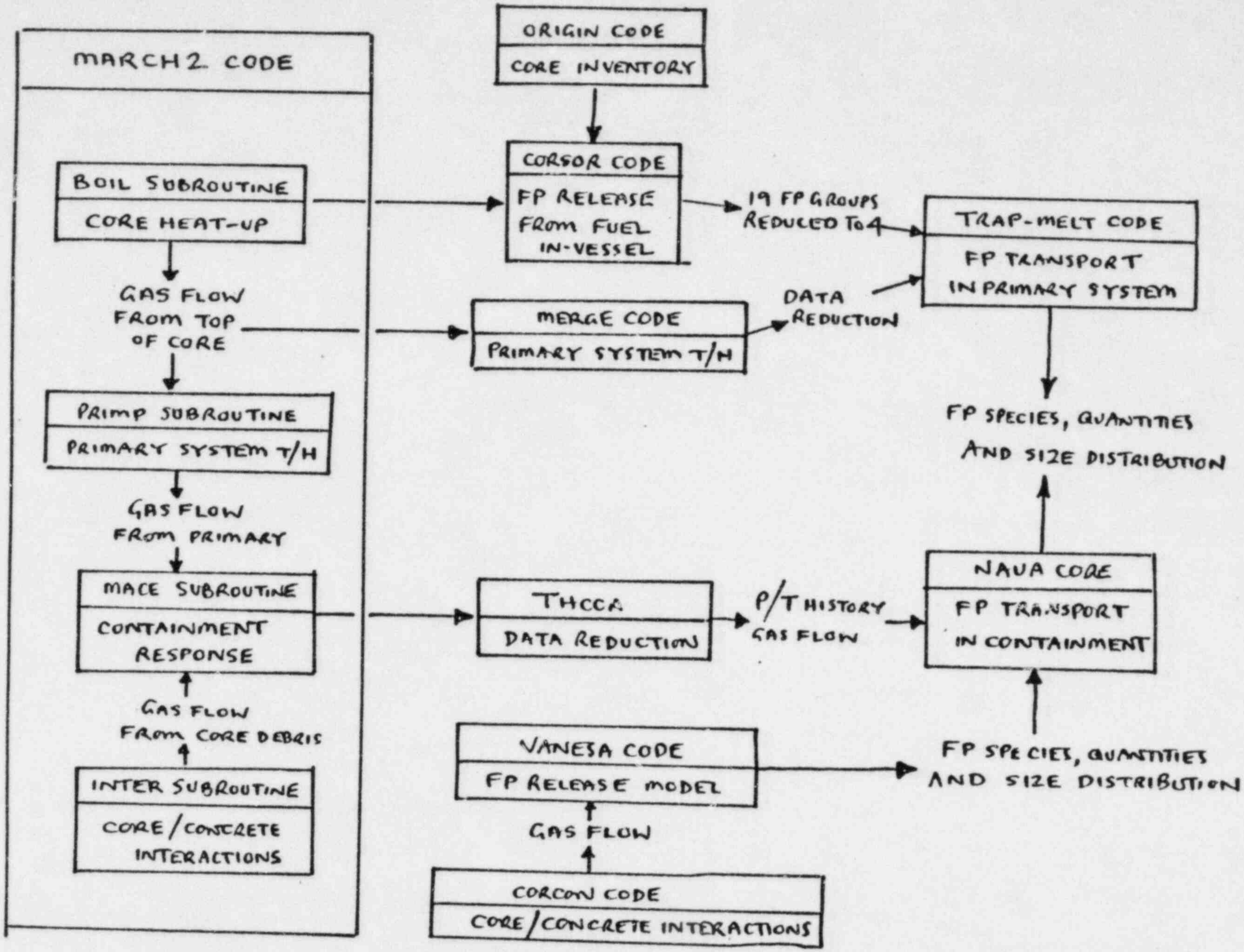

Figure 1.1 Flow diagram of ASTPO codes for application to Surry.

1.4 Calculational Procedure and Team

The audit calculations were performed using the suite of codes described in Section 1.3. BNL staff involved in the audit calculation are identified in Table 1.1. The audit calculation was performed by three BNL staff members (refer to "Analyst" column) familiar with running the respective ASTPO codes. These calculations were then in turn checked by other BNL staff members (see "QA" column) also familiar with the various codes. Integration of the codes is an extremely important process and this was ensured by a BNL staff member separate from the analyst team and this process was also subjected to a QA review.

A station blackout sequence (TMLB') in the Surry plant was selected at the November 20 meeting as the basis of the audit calculation. It was decided[5] after the meeting to analyze a basemat (ε) failure mode at BNL to ensure comparison with a calculation in Volume V of BMI-2104 that used a consistent set of in-vessel and ex-vessel input parameters. The overpressurization failure mode (δ) in BMI-2104 used primary system behavior (MARCH, CORSOR, MERGE, and TRAP-MELT) consistent with an ε failure mode rather than the appropriate δ failure mode. Differences between the primary system behavior for the basemat and overpressurization failure modes in BMI-2104 relate primarily to the amount of cladding that oxidizes in-vessel (59% clad reaction for the ε failure mode vs. 93% reaction modeled for the δ failure mode). The amount of clad reaction principally influences the Te release. The fraction of clad reacted is controlled by MARCH input parameters and a 93% in-vessel reaction is certainly an upper estimate. Consequently, the overpressurization failure mode is based on an upper bound in-vessel clad oxidation but uses fission product releases based on 59% clad reaction assumed for the ε failure mode. The

influence of this inconsistency on the quantities of fission products released to the environment is not great. However, it was decided to base the audit calculations on the basemat failure mode because this was calculated in a consistent manner in BMI-2104.

In Section 2 we describe the audit calculation and each individual code is briefly discussed. Finally in Section 3 the results of the audit calculation are compared with the BMI-2104 results and some comments are provided.

Table 1.1 BNL staff participating in audit calculation

Code	Analyst	QA
MARCH2	R. Jaung	J. Yang
CORSOR	R. Jaung	H. Ludewig
MERGE	R. Jaung	K. Perkins
TRAP-MELT	R. Jaung	H. Ludewig/W. Yu
NAUA	W. Yu	H. Ludewig
CORCON	G. Greene	M. Khatib-Rahbar
VANESA	G. Greene	W. Yu
Code Integration	H. Ludewig	W. T. Pratt

2. AUDIT CALCULATIONS

A station blackout sequence resulting in failure of containment via basemat penetration (TMLB'ε) was selected as the basis of the audit calculation. This sequence results in failure of all active emergency core cooling systems and containment heat removal systems. Initially, the secondary side heat sink (steam generators) boils dry. After the ultimate heat sink is lost the primary system begins to boil at the set point of the relief valves and the primary system water inventory is lost. Eventually, the reactor core is uncovered and it begins to heat up and degrade. During this process the in-vessel release of fission products begins. Without coolant injection the core will melt and slump into the bottom of the reactor vessel where it will thermally attack the lower vessel head. When the reactor vessel fails any residual primary system water is released to containment while the core materials are released from the vessel. For this sequence, as the primary system depressurizes, the accumulators will inject water. The subsequent accident progression depends on the amount of water in the reactor cavity and on the mode of contact between the core debris, water and concrete.

The codes described in Section 1.3 were applied to the above sequence in a manner similar to that assumed in Volume V of BMI-2104. Input parameters and intercode data transfer have been carefully checked and an independent audit calculation performed. Each code used to model the above accident sequence is described in the following sections. The sections also give a discussion on the input model assumptions used in BMI-2104 and in some cases suggest alternative assumptions. During the audit calculation an inconsistency was found for the TMLB'-ε sequence as reported in BMI-2104. That analysis (Section 2.1) assumes quenching of the core debris immediately after vessel

failure and does not predict core/concrete interactions to begin until the cavity boils dry (297 minutes after reactor scram). However, fission product release due to core/concrete interactions is modeled separately from MARCH using the CORCON/VANESA codes and this process was started at vessel failure (157 minutes after reactor scram). The airborne fission product masses in containment are modeled using NAUA and the fission products generated by core/concrete interactions were input at 297 minutes, which is consistent with the MARCH calculation (but inconsistent with the CORCON/VANESA calculation). This inconsistency was corrected in our audit calculation and its impact on the predicted released of fission products is not great for the late containment failure mode (refer to Section 3).

2.1 MARCH2

The version of the MARCH code used in BMI-2104 Volume V for the station blackout sequence was 1.98. This version of the code is not available at BNL. At BNL version 2.111 was initially used. This was the most up-to-date version of MARCH operational at BNL at the beginning of the effort and also judged by BCL staff to be the most appropriate. However, difficulties were experienced with this version of the code since it became clear that the core/concrete interaction models used in versions 1.98 and 2.111 were quite different. A later version of MARCH was delivered to BNL (namely, 2.151) which (when used with the same input) gave results compatible with MARCH 1.98. All the calculations reported in this document are therefore based on MARCH 2.151.

The input to the MARCH 2.151 calculation is identical to that used in BMI-2104 Volume V. A comparison of the results is shown on Tables 2.1 through 2.3. It is seen that the event times (Table 2.1) and most of the primary system and containment response parameters agree well. Minor disagreement occurs for the core temperature, time at temperature and the fraction of zircaloy oxidized. These differences (although small) play a role in determining the amount of tellurium (Te) released during the melting phase of the accident. The implication of this will be pointed out in the CORSOR discussion in Section 2.3

The input parameters to MARCH were reviewed and found to be consistent with plant construction details. In addition, the input modeling assumptions are reasonable given our current understanding of the progression of a core meltdown accident. The assumption that approximately 60% of the clad will oxidize in-vessel is rather higher than the current "industry" estimates but

is certainly feasible. The rapid head failure time after core slump is also to be expected given the high primary system pressure for this accident sequence. It should also be noted that mechanisms exist to depressurize the primary system (due to high temperature degradation of the primary system pressure boundary) prior to direct corium attack on the bottom of the vessel. Such a possibility can be investigated using different MARCH input assumptions.

The subsequent ex-vessel interactions of the core debris depends on the configuration of the region below the reactor vessel. In Surry, for a station blackout sequence, the reactor cavity would be relatively dry and hence core debris released from the bottom of the reactor vessel would not immediately contact water. However, as the primary system depressurizes (below 665 psig) the accumulators will inject water, which will eventually reach the core debris in the reactor cavity. At this point MARCH can model two possible modes of core debris/water/concrete interactions.

The HOTDROP subroutine in MARCH assumes that the core materials, on contact with the water, form fine particles which mix homogenously with the water. This configuration allows rapid heat transfer between the core debris and water, which rapidly cools the core debris and generates significant quantities of steam. This rapid steam generation causes a rapid pressure rise in containment. Eventually, if water is not supplied to the reactor cavity, the fission product decay heat will dry out the core debris. The core debris is then assumed to reheat (adiabatically in HOTDROP) until it melts, which allows transfer (in MARCH) from the HOTDROP to the INTER (core/concrete interaction) subroutine. After INTER begins to calculate core/concrete interactions fission product release must also be calculated (refer to Sections 2.5 and 2.6).

Note that this adiabatic reheating of the core debris is a conservative assumption and if heat losses were modeled during this period the core debris may not reach its melting point. This assumption is built into the MARCH code and it is beyond the scope of the audit calculation to alter internal code structure. However, the above is the mode of core debris/water/concrete interactions assumed in BMI-2104 and some of the conservatisms inherent in the calculations should be noted. The second mode of ex-vessel core debris interaction is discussed below.

There is some experimental evidence that if water is poured onto molten core materials that a porous crust will form and thus prevent mixing of the water and core materials. The stability of such a crust across the large surface area of a reactor cavity has been questioned, however, the possibility can be modeled in MARCH by bypassing HOTDROP and assuming core/concrete interactions being immediately after vessel failure with an overlying water pool. INTER is used to model core/concrete interactions and heat transfer from the core materials to the water is limited by film boiling. This alternative mode of core debris/water/concrete interactions was not assumed in BMI-2104 but is feasible and it would give different containment loads and ex-vessel fission product release characteristics than given by the HOTDROP assumption.

Table 2.1 Accident event times (minutes)

	Audit (MARCH 2.151)	BMI-2104
Steam Generator Dry	69.0	67.5
Core Uncover	97.25	95.5
Start Melt	118.5	118.3
Core Slump	143.5	146.3
Bottom Head Fail	155.0	157.3
Cavity Dry	213.3	214.9
Start Concrete Attack	287.3	289.9
Containment Fail	738.0	738.2

Table 2.2 Primary system response

Event	Time (min) A*	Time (min) B**	Pressure (psia) A	Pressure (psia) B	Average Core Temp. (°F) A	Average Core Temp. (°F) B	Peak Core Temp. (F) A	Peak Core Temp. (F) B	Fraction Core Melted A	Fraction Core Melted B	Fraction Clad Reacted A	Fraction Clad Reacted B
Core Uncover	97.2	95.5	2369	2369	669	669	674	675	0.0	0.0	0.0	0.0
Start Melt	118.5	118.3	2366	2366	1926	1990	4130	4130	0.0	0.0	.05	.06
Core Slump	143.5	146.3	2363	2362	3762	3709	4130	4147	.580	.55	.38	.33
Bottom Head Failure	155.0	157.3	2366	2368	3509	3820	-	-	.87	-	.61	.59

*Audit Calculation
**BMI-2104 Results

Table 2.3 Containment system response

Event	Time (min) A*	Time (min) B**	Compartment Pressure (psia) A	Compartment Pressure (psia) B	Compartment Temp. (°F) A	Compartment Temp. (°F) B	Sump Water (lbm) A	Sump Water (lbm) B	Sump Water Temp. (°F) A	Sump Water Temp. (°F) B	Reactor Cavity Water (lbm) A	Reactor Cavity Water (lbm) B	Steam Cond. on Walls (lb/min) A	Steam Cond. on Walls (lb/min) B
Steam Generator Dry.	69	67.5	13.1	13.0	137	136	3.43(4)***	3.39(4)	138	138	0.0	0.0	1070	1051
Core Uncovery	97.25	95.5	28.7	28.8	219	219	2.07(5)	2.07(5)	196	197	0.0	0.0	2446	2498
Start Melt	118.5	118.3	26.2	25.7	211	209	2.44(5)	2.47(5)	200	200	0.0	0.0	1039	994
Core Slump	143.5	146.3	23.0	22.5	199	197	2.65(5)	2.69(5)	198	198	0.0	0.0	754	684
Bottom Head Failure	155.0	157.3	46.0	45.9	253	253	2.80(5)	2.82(5)	199	198	171(5)	171(5)	11649	11810
Cavity Dry	213.31	214.9	58.9	58.6	273	272	4.01(5)	4.06(5)	222	221	0.0	0.0	1407	1405
Start Core/ Concrete Interaction	281.31	289.9	46.2	46.0	253	253	4.52(5)	4.54(5)	226	225	0.0	0.0	634	653
Containment Failure	738.03	738.2	52.9	53.7	248	249	4.86(5)	4.85(5)	228	227	0.0	0.0	.74	0.0

*Audit Calculations
**BMI-2104 Results
***3.48(4) = 3.48×10^4

2.2 MERGE

A revised version of the MERGE code was received at BNL in September 1984 and this version was used in this analysis. The advantage of using this version of the code over earlier versions is primarily in the capability of representing the primary system by seven control volumes. In the interests of consistency these same seven control volumes were used in the primary system fission product transport calculation to be discussed in Section 2.4. The seven volumes are connected in series and represent the following structures.

Volume 1 - Core

Volume 2 - Core plate

Volume 3 - Thin metal structure upper vessel internals

Volume 4 - Thick metal structure upper vessel internals

Volume 5 - Piping (pressure vessel to pressurizer)

Volume 6 - Pressurizer

Volume 7 - Containment (sink)

MERGE input consists of volume dimensions and connections to adjacent volumes. The thermal/hydraulic input data was obtained from the MARCH 2.151 calculation. Direct comparisons with BMI-2104 calculations become difficult at this stage because some of the above volumes were combined for the MERGE step. The results of the four-volume MERGE calculation were extrapolated in BMI-2104 to the seven-volume model used in TRAP-MELT (refer to Section 2.4). We simply used the same seven control volumes in both MERGE and TRAP-MELT.

2.3 CORSOR

CORSOR input in the form of a core temperature history was obtained from the BOIL subroutine in MARCH, and initial fission product inventories and power peaking factors were taken from BMI-2104 Volume V. Table 2.4 shows the results of the CORSOR calculation. For Cesium (Cs) and Iodine (I) it is seen that essentially all the material is released during the melting phase of the accident. In the case of Tellurium (Te) 12.8 kg are released out of an initial inventory of 25 kg. In comparison, the values quoted in BMI-2104 show the same result for Cs and I. However, the BMI-2104 calculation released 9 kg of Te. This difference is to be expected since there are differences in the temperature history of the core and the fraction of zircaloy oxidized between our Audit calculations and BMI-2104. This implies that different quantities of Te will participate in the next steps of the calculation, i.e., transport within the primary system and during ex-vessel interactions of the corium with concrete following vessel failure.

Table 2.4 In-vessel mass balance (CORSOR)

Element	Initial Inventory (kg)	Mass Released (kg) During Core Melt
Cs	146	144.0
I	12.15	12.0
Xe	260	256.21
Kr	13	13.20
Te	25.4	12.80
Ba	61	10.73
Sn	262	98.25
Ru	215	1.53
Zr	179	.03
Mo	155	12.89
Sr	48	3.44
Ag	2750	1275.90
Cd	173	134.24
In	505	71.71
UO_2	79650	18.01
Zr (clad)	16454.	1.88
Fe	6486.	73.47

2.4 TRAP-MELT

The version of TRAP-MELT used in this analysis was implemented at BNL in April 1984. It is written in FORTRAN 5 and allows for five states. The same seven-volume representation of the reactor coolant system used in MERGE and as described in Section 2.2 was used in TRAP-MELT. Thermal/hydraulic data determined by MERGE was used as input for the various volumes and flows between volumes. The fission product release rate, which acts as the source in this calculation, was determined by CORSOR.

This step in the calculation proved to be quite frustrating, in that the TRAP-MELT solution was unstable and diverged toward the end of the transient. Different choices of time steps for thermal/hydraulic input and volume nodalization did not solve this problem. This numerical instability remains as an area of concern. It was decided to use only the stable part of the solution, since the instability only occurred at the end (approximately five minutes from the end) of the time frame of interest. The TRAP-MELT solution spans the time frame from beginning of core melt to bottom head failure. In order to obtain input for NAUA from TRAP-MELT, it was necessary to stop the calculation prior to vessel failure and release all the remaining fission products, which had been calculated to be released during the last five minutes by CORSOR, as a puff at the end of the TRAP-MELT calculation. In BMI-2104 this problem was handled by extrapolating the TRAP-MELT calculation to the time of vessel failure. In this way mass conservation was assured, but the quantity of fission products deposited in the primary system will be slightly underestimated. It was felt that for the particular sequence being considered, this approximation would have a small effect on the mass of fission products leaked to the environment, since the containment failure time is almost 10 hours after vessel

failure. This allows a large amount of time for agglomeration and settling in the containment, and the details of the release at vessel failure are less important. In other accident sequences where the containment may fail earlier, this approximation would be even less valid.

By inspecting the results presented on Table 2.5 it is seen that this shortcoming in the calculation does not affect CsI and CsOH, since their entire mass is emitted by the time the calculation diverges. In the case of Te and aerosols approximately 2 kg will be emitted to the containment at the time of bottom head failure. The slight difference between the releases calculated by CORSOR and those calculated by TRAP-MELT for CsI and CsOH are due to different time nodalizations used in representing the release rate and subsequently carrying out a sum to determine the total released mass.

Table 2.5 In-vessel mass balance during fission product transport (TRAP-MELT)

Fission Product Species	Released During Melting (CORSOR)	Released in Transport Code (TRAP-MELT)	Mass Retained (Audit)	Mass** Retained (BMI-2104)
CsI	24.61	24.67	21	21.8
CsOH	147.78	148.22	126	128
Te	12.80	10.70*	9.50	7.6
Other (Aerosol)	1702.06	1700.03	1605	1612

*2.1 Kg of Te release as a puff at vessel failure.
**Reproduced from Table 7.3 of BMI-2104.

2.5 CORCON

The CORCON code (Muir,[6] et al., 1981) was used to calculate the ex-vessel attack of molten core debris on the basemat and this calculation drives the ex-vessel aerosol and fission product source term calculation. The CORCON code is the state-of-the-art model for the analysis of the interaction between molten fuel and structural materials with concrete. Typical output of CORCON calculations are the concrete erosion rate, generation rates of concrete decomposition gases, and the core melt/temperature history. These three quantities are necessary input to the VANESA code for the calculation of the ex-vessel release of fission products and aerosols into the containment during a core melt accident.

The version of the CORCON code used in the all BMI-2104 accident sequence source term calculations was CORCON/MOD1 with two official Fortran update packages documenting changes made to the original code by Sandia National Laboratory. This code is heretofore referred to as CORCON/MOD1-C2. A third update which has not been widely distributed to date was identified during this audit calculation and was used in the BNL calculations.

The input to the CORCON code consists primarily of concrete composition data (user input), core melt composition (MARCH output), cavity geometry (plant specifications/FSAR), surroundings temperature history (user input or MARCH), initial melt temperature and time after SCRAM at start of core/concrete interaction (MARCH), and melt/concrete surroundings radiative emissivity vs. time (user input). Those input variables which are "user input" are left to the discretion and scientific judgement of the user. In addition, there are no internal parametric model variations possible through input. The input for the Surry TMLB' CORCON/MOD1-C2 calculation is identical to that used in

BMI-2104 Volume V, with the following exceptions. The SNL CORCON calculation was started at 157 minutes after SCRAM at an initial core debris temperature of 1807K and a time-invariant surroundings temperature of 500K. The BNL core/concrete interaction was started at 287 minutes after SCRAM with a time-invariant surrounding temperature of 1373K and initial melt temperature of 1777K. This was done to be consistent with the BNL MARCH calculation results of initial melt temperature, surroundings temperature and start of core/concrete interactions at 287 minutes after SCRAM (corresponding to the time INTER allowed core/concrete interaction to occur after adiabatic reheating from quenching). Both of the above core/concrete interaction calculations were performed at BNL and compared for differences. It was found that only minor differences were calculated in erosion rate, gas generation rates, and melt temperature vs. time.

The results of the CORCON/MOD1-C2 calculations are shown in Figures 2.1-2.4 as melt temperature history vs. time, integrated gas generation rates in kg and moles vs. time, and vertical and lateral erosion depths vs. time, respectively.

A more complete assessment of this computer code may be found in ORNL/TM-8842, Chapter V.[7]

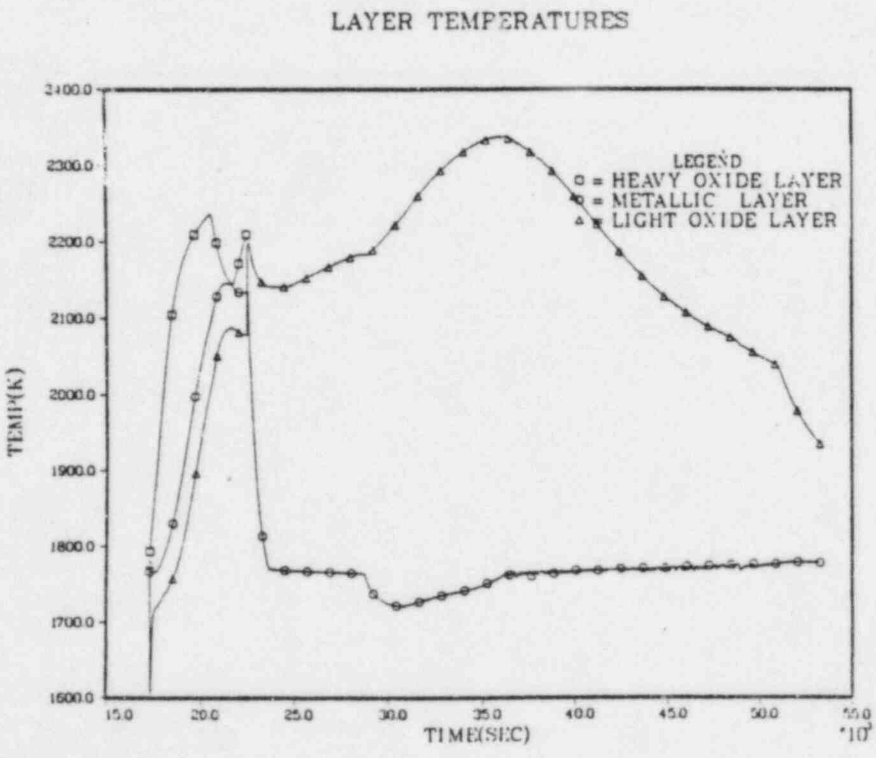

Figure 2.1 Melt temperature history vs. time: Surry TMLB' sequence.

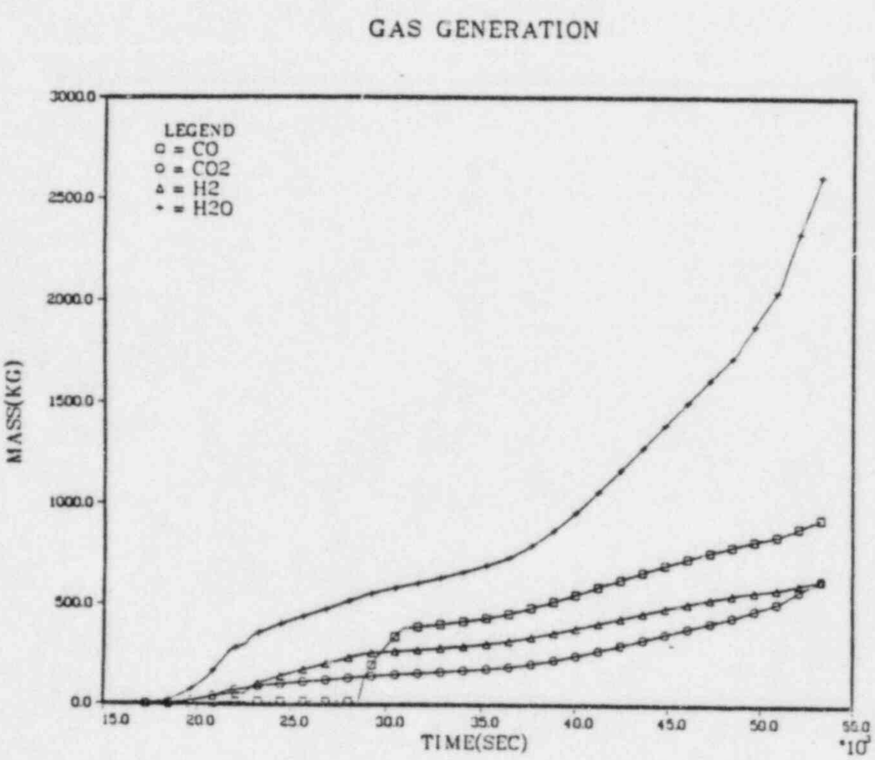

Figure 2.2 Integrated gas generation dates (kg/) vs. time: Surry TMLB' sequence.

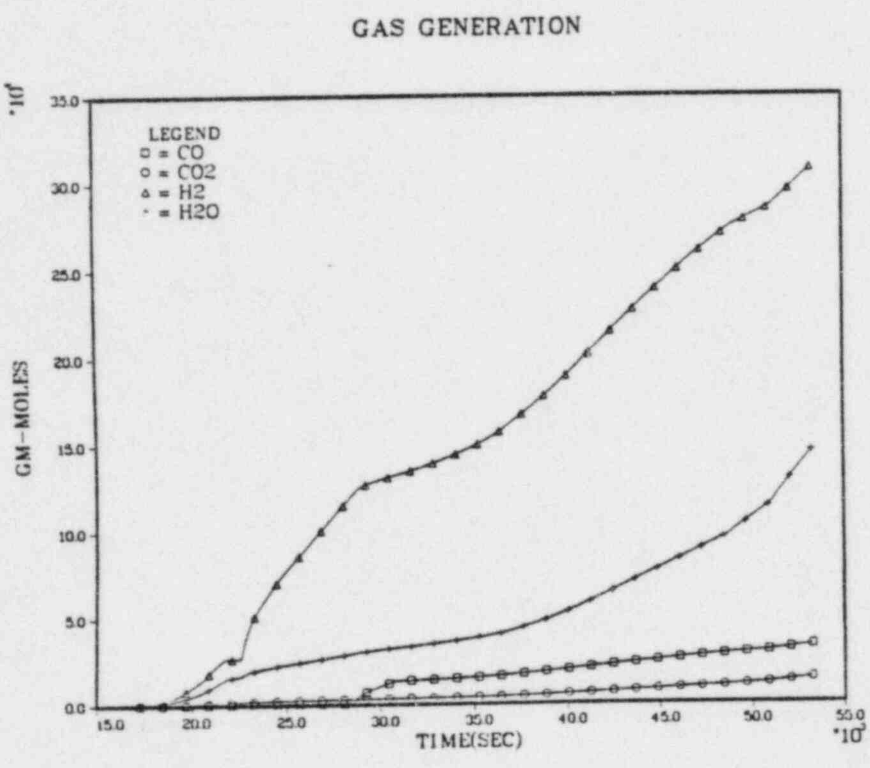

Figure 2.3 Integrated gas generation rates (gm-moles) vs. time: Surry TMLB' sequence.

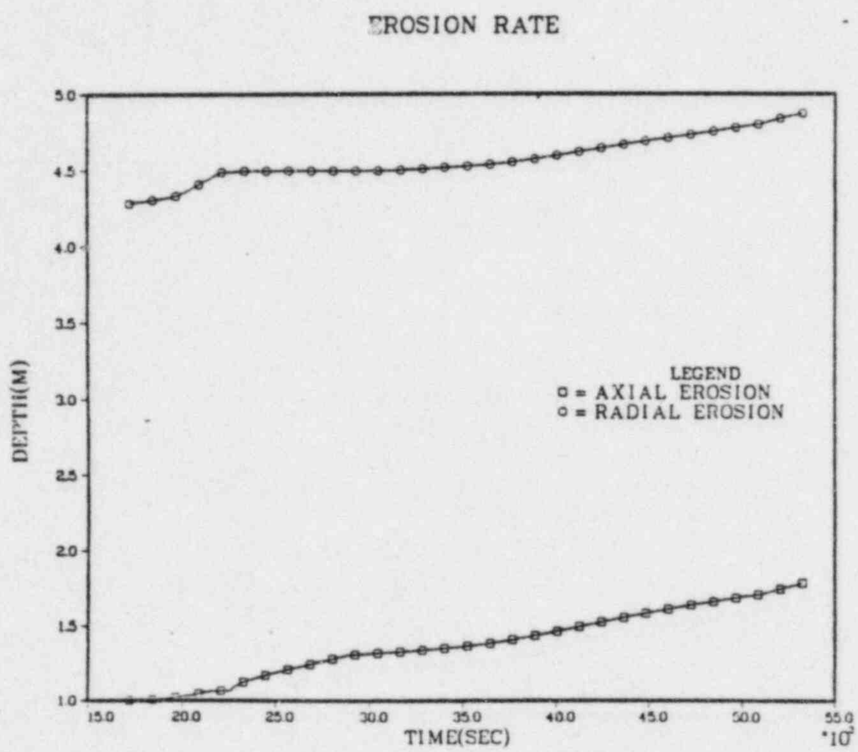

Figure 2.4 Concrete erosion depths vs. time: Surry TMLB' sequence.

2.6 VANESA

In order to account for the ex-vessel release of aerosols and fission products into the containment during the thermal interaction between molten core debris and structural concrete, the VANESA model was used. The VANESA model is a mechanistic description of the aerosol generation and fission product release during ex-vessel core-concrete interactions. A more complete description of the VANESA model may be found in ORNL/TM-8842, Chapter VI.[8] The VANESA model accepts as input the output of the CORSOR and CORCON computer codes.

From CORSOR, VANESA will get the mass of all species in the core melt inventory. At the present state of development, VANESA can accept explicitly 32 species; other species are surrogates of one of the 32 species. Examples of this are as follows:

Gd, Eu, Pm $\rightarrow$ La on molar basis

Np $\rightarrow$ Ce on molar basis

Cd $\rightarrow$ Cs on molar basis

In $\rightarrow$ Ag on mass basis

In addition, it is required to input the inventory of Nb as the equivalent mass of Nb_2O_5 directly into VANESA in Subroutine ASSEMB, line 41. These variations on melt species input reflect the ongoing development of the VANESA model as new species were identified as important during the Accident Source Term Reassessment Study.

From CORCON, VANESA receives the following information in time step intervals as specified:

oxide melt temperature (K)
integrated gas release rates (kg)

H_2
H_2O
CO
CO_2

maximum radial erosion radius (m)
SiO_2 content of melt (kg)

The arrays of these seven input variables were transferred automatically between CORCON and VANESA.

A detailed description of the output from VANESA has been documented elsewhere (Powers,[8] 1983) and will not be repeated here (see Table 2.6). The most important of the output quantities for subsequent use in the calculation of the ex-vessel source term are:

1. aerosol mass generation rate,
2. chemical composition of aerosolized mass,
3. aerosol mean particle size and,
4. aerosol density

The version of VANESA used in this audit calculation of the Surry TMLB' sequence as reported in BMI-2104 Volume V was the first exportable Fortran version. This version was adapted from the original non-Fortran version written at SNL and used in the BMI-2104 study. The code was first received at BNL on November 15, 1984. It was successfully compiled and a sample problem executed on November 19, 1984. In the process of familiarization with the code, several minor transcription errors were identified in the Fortran version by the SNL and BNL staff and were corrected.

The Surry TMLB' audit calculation was then run with the validated input from MARCH, CORSOR, and CORCON as previously specified. The results of this calculation were tabulated for each time step, listing the species in the aerosol, the source rate, oxide melt temperature, aerosol density, and aerosol mean particle size. One exception that should be explicitly mentioned is the Cs_2O group. Recall that this species actually consists of Cs_2O and Cd. A breakdown of this group for the first four time steps is given in Table 2.7 on the basis that Cd actually comprises 98.7% of this category on a mass basis. Also indicated in Table 2.7 are the source rates in gm/s for the times during which a water pool existed over the core melt. In these cases, SNL calculated a decontamination factor (DF) by which the source rate was diminished. BMI-2104 Volume V reports the source rates with the DF included; Table 2.7 lists both sets of results for the BNL audit calculations. Comparison of the results obtained in this calculation to those reported in BMI-2104 Volume V demonstrates excellent agreement on a quantitative basis.

Table 2.6 Output from the VANESA model (ORNL/TM-8842)

Aerosol Properties

1. Density of aerosol material (g/cm^3)
2. Mean aerosol particle size (μm)
3. Mass flux of aerosols (g/s)
4. Aerosol concentration at STP (g/m^3)
5. Aerosol concentration in cavity (g/m^3)

Aerosol Composition

1. Fission Products (mass percent Cs, I, Te, Ru, Sb, Mo, Sr, Ba, U, Pu, Ce, La, Nb)
2. Concrete Constituents (mass percent Na, K, Al, Si, Ca, Fe)
3. Fuel and Structural Materials (mass percent Fe, Ni, Cr, Mn, Sn, Zr, U)

Kinetics Data

1. Source of Release (sparging, evaporation, mechanical)
2. Rate limitation (surface area, time, mass transport, or chemical kinetics)
3. Vapor phase speciation

Melt Composition

1. Change caused by aerosol formation
2. Change caused by metal oxidation
3. Change caused by concrete melting

Permanent Gas Characteristics

1. Composition (volume percent CO, CO_2, H_2, H_2O, OH, O, H)
2. Flux (moles/s)
3. Superficial velocity (m/s)

Table 2.7 Breakdown of Cs_2O grouping: Surry TMLB'

t (sec)	0	1200	2400	3600
Cs_2O (%)	1.26	.52	.41	.36
Cd (%)	95.56	39.34	30.92	27.21
Source rate (gm/s) w/DF*	1.5	11.2	40.2	66.5
Source rate w/o DF	4.4	21.3	53.6	72.4
DF	2.947	1.903	1.334	1.088

*DF - Decontamination Factor

2.7 NAUA

The NAUA code used in this calculation was transmitted to BNL in December 1984, and includes models for diffusio-phoresis and homogenous nucleation. This final step of the calculation takes input from TRAP-MELT and VANESA in the form of fission product aerosol source, and from MARCH 2.151 for the containment building temperature, steam addition rate and leak rate. The in-vessel release is supplied by TRAP-MELT and it begins soon after the start of core melting and continues through to the time of bottom head failure, which consists of a puff release of Te and aerosol as described in Section 2.4. At the time of bottom head failure a large amount of water and steam is also added to the containment (Table 2.3). The core debris, which is on the cavity floor is assumed to be cooled to the water temperature at this time. During the next phase of the accident the water in the cavity boils away and the core debris eventually drys out. No fission products or aerosols are released during this re-heating phase. The core debris is assumed to become sufficiently hot to attack the concrete basemat at 287 minutes (approximately 2 hours following bottom head failure). This attack is computed to continue until the core debris penetrates the concrete basemat and containment failure occurs.

The aerosol source term description is represented by two modes. The first mode represents the source from the primary system into the containment and the second mode represents the source from core/concrete interaction. The first mode stops after 8945 seconds, which corresponds to the time when the TRAP-MELT calculation diverged, and is six minutes before the bottom head fails. The second release mode at that time represents the puff release corresponding to the unreleased fission products during the melting phase. There is no aerosol source from the time of bottom head failure until the cavity

dries out (17200 seconds). At this time the core/concrete interaction source starts and continues up to basemat failure. The release rate (gm/sec), fraction of aerosol composition and aerosol size description are determined by CORCON/VANESA. The fission product input to VANESA corresponds to the mass leaving the Reactor Pressure Vessel following bottom head failure. In the calculation described in Section 2.6 this mass was based on fission product masses reported in BMI-2104 Volume V. The masses released in the Audit calculation are different (see Table 2.8) and this difference was accounted for by adjusting the release fraction in the calculation described in Section 2.6. These adjusted release fractions, together with the release rates and aerosol size description from Section 2.6, form the input to NAUA for the core/concrete interaction.

Table 2.8 Input to VANESA (kg)

Species	Audit	BMI-2104
I	.15	.1
Cs	1.8	.7
Te	12.6	16.4

Table 2.9 shows a mass balance for the containment calculation. It is seen that the bulk of the material is deposited on horizontal surfaces (sedimented deposition) with a smaller fraction on vertical surfaces (diffusion

deposition). "Other" in this table refers to aerosols leaking out of the primary system and those generated in the core/concrete interaction, where as in Table 2.5 "other" refers to aerosols generated only during the in-vessel release phase.

Table 2.9 Ex-vessel mass balance (NAUA)

Fission Product Species	Sedimented Deposit (kg)	Diffusive Deposit (kg)	Airborne (kg)	Total in Containment (kg)	Leaked (kg)	Total (kg)
CsI	3.397	0.628	0.0019	4.027	0.0403	4.067
CsOH	17.735	3.264	0.0012	21.0	0.0261	21.026
Te	4.971	0.541	0.389	5.901	1.790	7.691
Other	663.526	67.719	43.415	774.660	275.0	1049.660

3. SUMMARY

Shown on Tables 3.1 and 3.2 is a summary of the results of the audit calculations. The tables show the final distribution of fission products after the accident. A direct comparison with values taken from BMI-2104 is made, and for CsI and CsOH the agreement is good. In both calculations approximately 15% of the CsI and CsOH remain in the containment, 85% in the reactor coolant system and a small fraction leaks to the environment. The estimated fractions leaked to the environment agree closely.

In the case of Te the distribution of fission products is somewhat different between the two calculations. The audit calculations predict a larger fraction of fission products retained in the primary system that is calculated in BMI-2104. This is due to the higher Te release in the audit calculation during the in-vessel core degradation and thus more Te is available for retention in the primary system. During ex-vessel corium/concrete interactions the split between sedimented retention on containment surfaces and retention in the melt is also different between the two calculations. A larger fraction is predicted to be retained on surfaces in the audit calculation and a smaller fraction is retained in the melt than in BMI-2104. These differences are possibly due to differences in the CORCON/VANESA calculation. It has been pointed out in earlier sections of this report that there are inconsistencies between the NAUA and CORCON/VANESA calculations in BMI-2104. These inconsistencies were eliminated in our audit calculations resulting in the differences noted below:

	Audit		BMI-2104	
	CORCON/VANESA	NAUA	CORCON/VANESA	NAUA
Start of core/concrete interaction (min)	287	287	157.3	282
Initial melt temperature (°K)	1777	1777	1807	1807
Temperature of surroundings (°K)	1373	1373	500	500
Initial Te mass (kg)	12.6	12.6	16.4	16.4

From the above table it is clear that the CORCON/VANESA calculations in BMI-2104 were started at the point of vessel failure rather than at the point of debris dryout. In the audit calculation, the CORCON/VANESA calculations was started at the time of debris dryout. However, both calculations predict that essentially the same fraction of Te leaks to the environment. This agreement is due primarily to the long time before containment failure for this sequence. For a sequence in which the containment function is compromised at an earlier time the agreement may not be as close.

A final check between these two calculations is a comparison between the airborne and leaked mass as a function of time. Figures 1 and 2 illustrate this comparison. The audit calculation (Figure 1) and the BMI-2104 calculation (Figure 2) show the same trends and to a large extent the same magnitudes. The airborne mass peaks following vessel failure due to steam in the atmosphere. The peak values agree well between the calculations. A valley develops as the steam condenses, and aerosols due to core/concrete interactions have not been generated yet. The valley in the audit calculation is

deeper than in the BMI-2104 calculation. However, the remainder of the curve starting at the beginning of core/concrete interactions agrees well with the BMI-2104 curve. Following containment failure the BNL curve predicts a sharp decrease.

The audit calculation has further shown that some inconsistencies exist in BMI-2104 regarding inter code data transfer. For this particular suite of codes this is a significant difficulty and great care must be taken to ensure code compatibility. However, for the particular sequence and failure mode considered in the audit calculation the inconsistencies did not strongly influence the predicted release of fission products. Obviously for any future use of the ASTPO codes to a specific reactor plant code compatibility should be ensured and also the other input model assumptions noted in this report (but not exercised in BMI-2104) should also be factored into a comprehensive analysis.

In summary, this short-term audit calculation has demonstrated that the suite of codes developed under ASTPO sponsorship can be successfully exported to an independent organization and that the results in BMI-2104 can be reproduced.

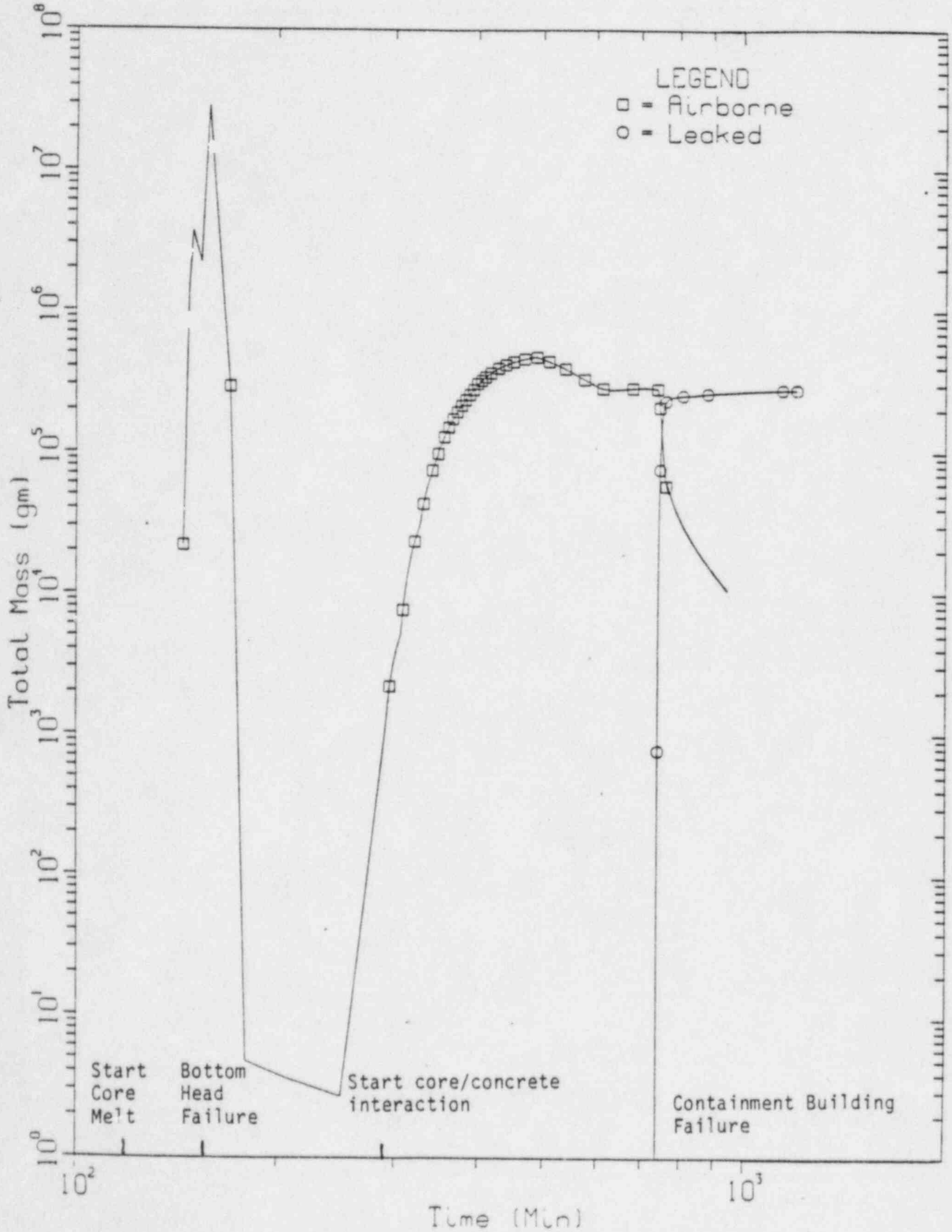

Figure 3.1 Airborne and leaked masses (Audit calculations).

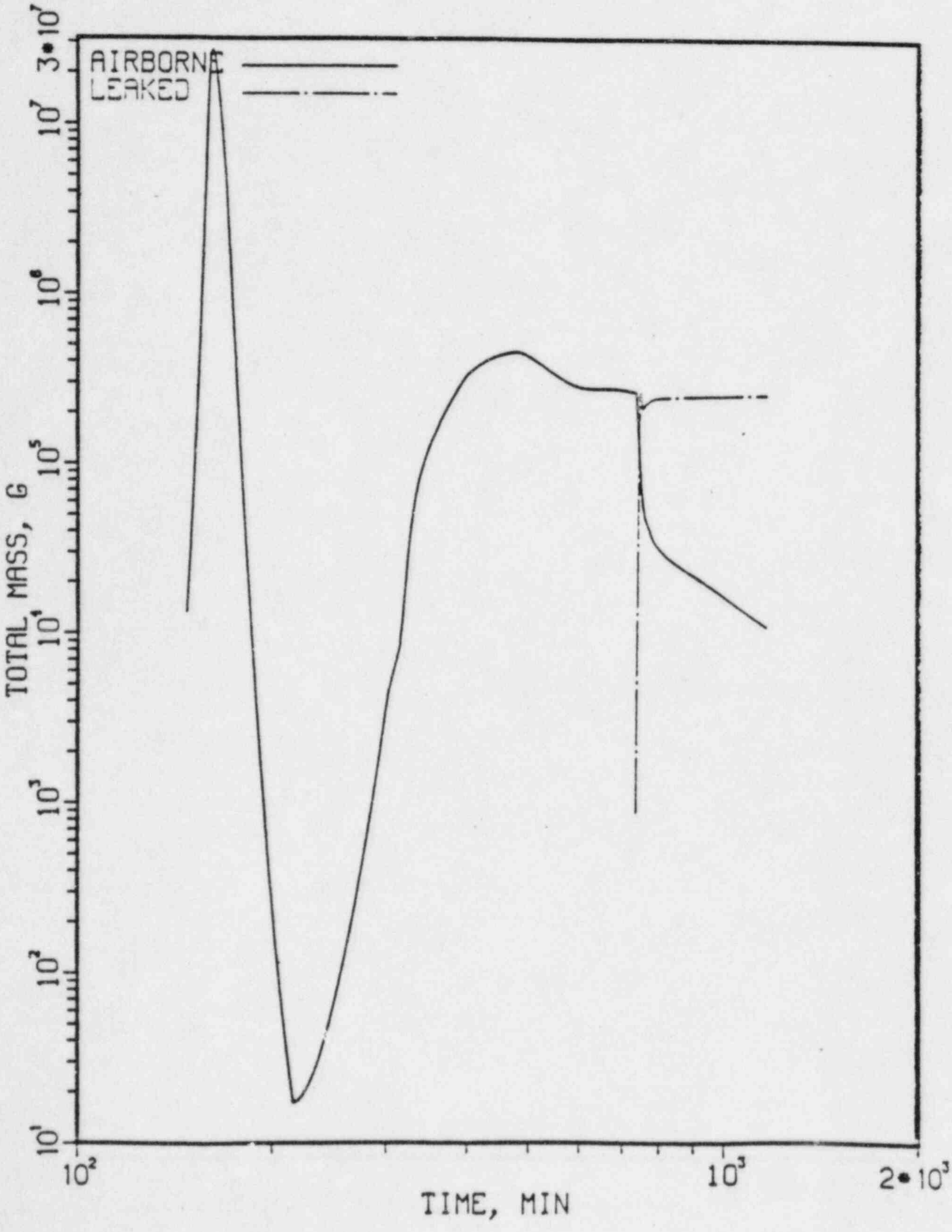

Figure 3.2 Airborne and leaked masses (BMI-2104 Volume V)
(Reproduced from Figure 7.34 of Volume V of BMI-2104)

Table 3.1 Overall mass balance (kg)

Species	Total in Containment	Leaked Mass	Mass Retained in Primary System	Mass Retained in Melt	Total Mass
CsI	4.027	0.0403	20.72	-	24.79
CsOH	21.00	0.0261	125.99	-	147.02
Te	5.901	1.79	9.51	8.2	25.4

Table 3.2 Distribution of species after accident (BNL fractions based on total mass from Table 3.1)

Species	Fraction of Core Inventory							
	In Containment		In Melt Ex-Vessel		In Primary System		Leaked	
	Audit	BMI-2104	Audit	BMI-2104	Audit	BMI-2104	Audit	BMI-2104
CsI	.16	.15	-	-	.84	.85	1.6(-3)*	2.8(-3)
CsOH	.14	.14	-	-	.86	.86	1.8(-4)	1.7(-4)
Te	.24	.19	.31	.43	.38	.30	7.2(-2)	8.1(-2)

*1.63(-3) = 1.63×10^{-3}

4. REFERENCES

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