THE NRC ANALYSIS PROGRAM FOR SEVERE ACCIDENTS IN LWR'S

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I. Introduction

The analysis of severe accident consequences for light water reactors has been a project of major importance in the NRC prior to and, even more vigorously, after the event at Three-Mile Island Unit 2. In the development of a methodology for ascertaining the consequences of such events for use in risk analysis and source term studies, one must appreciate beforehand that the nature and complexity of the phenomena involved limit the extent to which an exact analysis capability is possible. An accurate analysis capability would require a set of experimentally validated models capable of treating in great detail all the phenomena occurring in the multi-phase, multi-component, nuclear steam supply system (NSSS) over temperature ranges from 300C to 2800C, pressure ranges from 15 psi to 2350 psi, and time periods over days, weeks, and possibly months. As a practical matter, the capabilities of an analysis technique of this type must be limited since:

1. Fully integral tests to validate all the models under all possible conditions for all types of plants would be prohibitively expensive, take decades to accomplish, and be difficult to evaluate.

2. Risk analyses and the necessary sensitivity and uncertainty analyses which are part of them require fast-running analysis codes so that many sequences for many plants can be made in a reasonable time frame. Therefore, fast-running computer codes require extensive model simplification to achieve this goal.

3. An exact quantification of the technology of all the processes expected to occur is not possible since uncertainties will always exist in experimental data for material properties, physical and chemical parameters, the models themselves, and the sequences predicted to occur.

Therefore, in order to make the goal of quantifying the consequences of severe accidents tractable, one must approach the problem in a more realistic manner. That is, recognizing that (1) uncertainties will always exist, (2) complete experimental validation is not feasible, and (3) analysis codes for risk studies must be fairly fast (i.e., less than a few hours on a CRAY for a given sequence and plant), a methodology must be developed that will result in a fast-running analysis code that evaluates the entire nuclear steam supply system and is composed of phenomenological models whose uncertainty can be quantified. Such a goal requires the accomplishment of two major prerequisite tasks; namely:

(1) The establishment of a data base to the extent practically possible in the temperature and pressure ranges of interest and,

(2) The development of specialized detailed analysis codes for specific components of the NSSS such as the primary system T/H, the core, and the containment. These codes should be validated using item (1) above.

Given the above two elements, the final goal of a fast-running overall NSSS code containing quantified uncertainties in its output can be achieved by benchmarking its simplified models against the corresponding models in the more mechanistic analysis codes. The uncertainty in the latter codes will be quantified -- to the extent possible -- by the established data base. It is to be expected that the uncertainty of a given model in the overall code will be greater than or equal to that in the more mechanistic code because of the for need fast-running times. However, whatever the magnitude of the uncertainty, <u>it will be quantified and the results can therefore be used in decision making.</u>

II. Current Program of the NRC

In order to achieve its goal of developing a fast-running NSSS analysis code with quantifiable uncertainties for analyzing the consequences of severe accidents in LWR's, the Office of Research established the Severe Accident Research Program (SARP) as part of its long-range plan shortly after the TMI-2 event. This program consists of approximately seventy research contracts to develop data bases for core behavior; primary system behavior; containment behavior; fission product release, transport, deposition, and re-evolution; hydrogen behavior; and detailed analysis codes for all of the above including severe accident sequence analysis (SASA); detailed risk analysis; and risk reduction studies. The current program utilizes approximately one-third of the annual Research budget, and is scheduled for completion in the 1986/87 time frame. The object analysis code mentioned above (i.e., the fast-running NSSS computer code with quantified uncertainties) is under development and has been given the name MELCOR. The initial, unvalidated version, is scheduled for completion by the end of FY 1984.

The development of the supporting mechanistic analysis codes for benchmarking and quantifying MELCOR has been underway since 1980. This set of codes consists of a few newly developed codes, previously developed codes (such as TRAC and RELAP5), and linked packages of new and previously-developed codes. An extensive effort has been made to utilize previously developed analysis packages to avoid "re-inventing the wheel and, of course, to minimize expenditures. In order to clearly show how all these codes (there are 25 of them) fit into the general scheme outlined in the Introduction, it will be necessary to graphically illustrate their function in relation to the NSSS. In order to do this we must first separate those codes intended for detailed analysis for benchmarking MELCOR from the current "risk" codes that MELCOR is intended to replace.

A. Current NRC Risk and Source Term Analysis Codes

This group of codes is currently being used in the current NRC evaluation of the source term for selected plants and sequences. Earlier versions of some of the models were used in the WASH-1400 study. It must be remembered that these codes are not "detailed" in the sense mentioned above; i.e., they are fast-running codes which necessarily requires considerable use of simplifying assumptions, empirically-based correlations, and user input options. They also have not had the benefit of an adequate data base from which to validate and assess their simplified models.

However, they represent our current best integrated analysis capability for source term analyses until the SARP program is completed and MELCOR has been fully validated and quantified. Table I gives a list of the codes and their application to the NSSS. Note that this family of codes is commonly referred to today as the "Battelle Suite of Codes" since the Battelle-Columbus laboratory has been prime contractor for their application. It should be noted that most of the models in this code series will be used directly in MELCOR with only slight changes. These models are ORIGEN, TRAP/MELT, VANESA, SPARC, CORCON, and ICEDF. Major improvements are expected for the MARCH, MERGE, CORSOR, and containment aerosol applications. MELCOR will, however, contain additional models for containment temperature and pressure response as well as ex-plant consequence models.

B. "Mechanistic Specialized Codes for Benchmarking MELCOR and Special Applications

This group of codes represent best-effort modeling with little emphasis on speed, but great emphasis on model accuracy. In other words, these codes are intended to represent the best state of technology for specialized phenomena and for applications to specific areas of interest. These codes, when completed, will be maintained as a best-estimate base of modeling expertise to be used when the necessarily simplified models in MELCOR are judged inadequate for highly specialized applications. The general philosophy being applied here is that the NRC staff must maintain "state of knowledge" expertise to be able to do in-depth studies of important phenomena whenever the need arises. However, as stated above, the primary application of this code group is to benchmark and

quantify the modeling uncertainties that will inherently be present in MELCOR, and, incidentally, to the Battelle Suite of Codes. Table II presents a summary of this group and where they are applied in the NSSS. With regard to the availability dates given in Table II, it should be understood that these codes will be continually updated and improved beyond that date as new experimental data for validation become available.

C. Linkages of Mechanistic Codes for Special Applications Where "Feedback" is Important

There are many situations where the input period from one specialized code is affected by the output of the receiving code over a given time period. For such cases, more accurate modeling is accomplished by "linking" the codes together so that input/output information can be interchanged during the run. An example of this is a TRAP/MELT calculation which relocates a significant part of the decay heat source into the upper plenum. This relocated heat source will affect the MERGE input on flows and structures in the upper plenum and other parts of the primary system and, therefore, future TRAP/MELT computations. Therefore, it is essential to develop hard links between the mechanistic fuel behavior modules (SCDAP, MELPROG) and system T/H codes (RELAP, TRAC) because of the intimate coupling between fuel degradation and the T/H behavior of the reactor coolant system. Note that MELCOR will be a fully integrated code package with inter-model feedback included as an inherent part of the programming.

For the detailed code series a few major code links are planned and are currently being implemented. They are shown in Table III. Completion dates for these linkages have not been firmly established, but they are expected to be available by mid 1985. The major reason for the SCDAP/RELAP5 link in addition to the TRAC/MIMAS/MELPROG link is that detailed modeling of the fuel pins are necessary for attentuated accidents but may be necessary for "core on the floor" accidents. Therefore, using the less detailed MIMAS code in place of SCDAP will increase computational speed dramatically for non-attenuated accidents scenarios. Moreover, the SCDAP/RELAP5 link will give early capability to analyze both PWR and BWR systems for these less drastic events. It should be noted that the above

procedure is very cost-effective since the system codes have already been developed and validated as part of the ECCS research program, and the linkages can be accomplished in a very short time to give essentially universal applicability. That is, capability for PWR's and BWR's for "core on the floor events" and for attenuated events such as the accident in TMI-2. Finally, the linked codes can be used to benchmark a wide range of MELCOR's integrated package and quantify the effects of feedback in severe accident analyses.

III. Summary

The code development plan outlined above will provide the NRC with the analysis capability required for decision-making on severe accidents in LWR's. An integrated risk code is provided - MELCOR - for large-scale PRA and source term studies as well as special-application, more mechanistic codes for less broad, more specific decision-making processes. The plan utilizes to the broadest extent possible the codes developed for other purposes as well as the extensive new data base being developed under SARP for NSSS behavior under severe accident conditions.

Table IV summarizes how all these codes are used in NSSS analyses by classifying them by NSSS component application and by phenomenological categories. Finally, Figure 1 summarizes the codes graphically to show how they will "fit" together to accomplish their intended purpose. Note that some of the current "Battelle Suite" of codes will be used directly in the mechanistic set and will not be "re-invented". Note also that the CONTAIN code consists of many models from current codes used in a subroutine capacity such as CORCON, HECTR, etc.

TABLE I

BATTELLE SUITE OF CODES

FOR RISK AND SOURCE-TERM STUDIES

NAME OF CODE	APPLICATION IN THE NUCLEAR STEAM SUPPLY SYSTEM
ORIGEN	MODELS FISSION PRODUCT INVENTORY IN THE CORE PRIOR TO SCRAM
MARCH 2.0	PRIMARY SYSTEM T/H CORE PHENOMENA (CONTAINMENT T&P)
MERGE	IN-VESSEL GAS FLOW AND HEAT TRANSFER TO STRUCTURES. USED AS AN INTERFACE BETWEEN MARCH & TRAP/MELT
TRAP/MELT	MODELS FISSION PRODUCT AND AEROSOL TRANSPORT AND DEPOSITION WITHIN THE REACTOR COOLANT SYSTEM
CORSOR	MODELS FISSION PRODUCT AND AEROSOL RELEASE FROM THE CORE. AN EMPIRICAL CODE BASED UPON EX-PILE, FISSION-PRODUCT RELEASE EXPTS
CORCON	MODELS EX-VESSEL MOLTEN CORE INTERACTION WITH REACTOR CAVITY BASEMAT MATERIAL
VANESA	MODELS FISSION PRODUCT AND AEROSOL RELEASE DURING MOLTEN CORE/BASEMAT INTERACTION
NAUA-4	MODELS AEROSOL BEHAVIOR IN THE CONTAINMENT
SPARC	MODELS AEROSOL RETENTION IN SUPPRESSION POOLS (BWR ONLY)
ICEDF	MODELS AEROSOL RETENTION IN PWR ICE-CONDENSER CONTAINMENT SYSTEMS

TABLE II

DETAILED NRC SEVERE ACCIDENT COMPUTER CODES TO BENCHMARK AND QUANTIFY MELCOR

NAME OF CODE	APPLICATION IN THE NUCLEAR STEAM SUPPLY SYSTEM	AVAILABILITY DATE
SCDAP	DETAILED CORE BEHAVIOR TO LOSS OF ROD GEOMETRY, I.E., TO ABOUT 2400K	OCTOBER 1984
MELPROG	DETAILED CORE BEHAVIOR THROUGH MELTDOWN AND EXIT THE REACTOR PRESSURE VESSEL	DECEMBER 1984
MIMAS	SIMILAR TO SCDAP-BUT MUCH LESS DETAILED--USED AS INPUT TO MELPROG FOR UNATTENUATED "CORE ON THE FLOOR" EVENTS	1-D VERSION COMPLETE 2-D VERSION IN DECEMBER 1984
FASTGRASS	DETAILED FISSION PRODUCT RELEASE FROM INTACT FUEL USED IN SCDAP & MELPROG	OCTOBER 1984
TRAP/MELT	SEE TABLE I - THESE MODELS WILL BE MODIFIED & USED IN SCDAP, MIMAS, AND MELPROG	COMPLETED
VICTORIA	DETAILED FISSION PRODUCT AND AEROSOL RELEASE FROM MOLTEN FUEL FOR USE IN MELPROG	OCTOBER 1985
CONTAIN	INTEGRATED DETAILED CONTAINMENT MODEL: I.E. IT USES SUBMODELS FOR T/H, AEROSOL AND FISSION PRODUCTS (MAEROS), CAVITY MODELS (CORCON, MEDICI, VANESA), HYDROGEN BURNING (HECTR) & ESF MODELS	JUNE 1984
HECTR	SEE CONTAIN CODE ABOVE. HECTR MODELS HYDROGEN BEHAVIOR IN CONTAINMENT. TREATS DEFLAGRATIONS, SPRAYS, HEAT TRANSFER, IGNITERS, SUPPRESSION POOLS, SUMPS, AND FANS	DECEMBER 1984

TABLE III

DETAILED NRC CODE LINKS

FOR SEVERE ACCIDENT ANALYSES*

CODES LINKED	APPLICATION TO NSSS
SCDAP/RELAP5	INTEGRATED PRIMARY SYSTEM/PARTIALLY DEGRADED CORE BEHAVIOR FOR ATTENUATED ACCIDENTS SUCH AS TMI-2
TRAC/MIMAS/MELPROG	INTEGRATED PRIMARY SYSTEM/MOLTEN CORE BEHAVIOR FOR "CORE ON THE FLOOR" EVENTS.

*NOTE: EACH PACKAGE WILL CONTAIN FULLY INTEGRATED FISSION PRODUCT AND AEROSOL RELEASE, TRANSPORT AND DEPOSITION MODELS FROM FASTGRASS, VICTORIA, TRAP/MELT, AND VANESA

TABLE IV
SUMMARY OF NRC SEVERE ACCIDENTS COMPUTER CODES
(MELCOR MODELS ALL)

NSSS COMPONENT	RISK CODES	MECHANISTIC CODES
PRIMARY SYSTEM	MARCH, MERGE TRAP/MELT	TRAC, RELAP5, TRAP/MELT
CORE	MARCH, CORSOR	SCDAP, MELPROG, MIMAS, FASTGRASS, VICTORIA, TRAP/MELT
REACTOR SUMP	CORCON, VANESA	CORCON, VANESA, MEDICI
CONTAINMENT	NAUA-4, SPARC, ICEDF, MARCH	CONTAIN (INCLUDES SUMP CODES)

BY PHENOMENOLOGICAL CATEGORY

T/H CODES: MARCH, TRAC, MERGE, RELAP5

FISSION PRODUCT RELEASE CODES: CORSOR, FASTGRASS, VICTORIA, VANESA

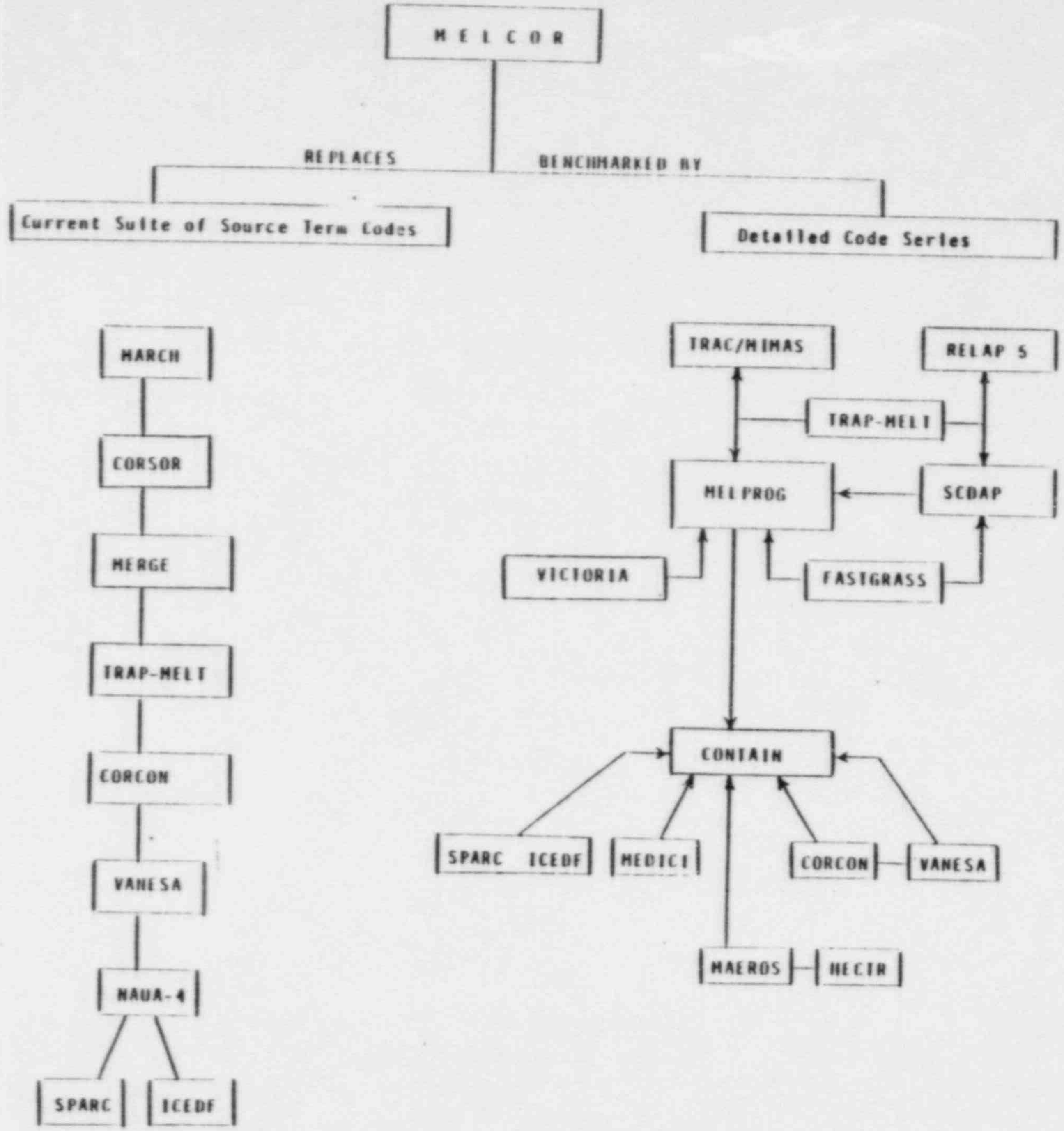
FISSION PRODUCT TRANSPORT AND DEPOSITION CODES: TRAP/MELT

AEROSOL BEHAVIOR CODES: NAUA-4, MAEROS (IN CONTAIN)

REACTOR CAVITY MODELS: CORCON, MEDICI

HYDROGEN BEHAVIOR CODES: HECTR (IN CONTAIN)

ESF CODES: SPARC, ICEDF (BOTH IN CONTAIN)



MELCOR
REPLACES
BENCHMARKED BY
Current Suite of Source Term Codes
Detailed Code Series
MARCH
CORSOR
MERGE
TRAP-MELT
CORCON
VANESA
NAUA-4
SPARC
ICEDF
TRAC/MIMAS
RELAP 5
TRAP-MELT
MELPROG
SCDAP
VICTORIA
FASTGRASS
CONTAIN
SPARC ICEDF
MEDICI
CORCON
VANESA
MAEROS
HECTR