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SCDAP/RELAP5
Independent Peer Review

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by

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ABSTRACT

The SCDAP/RELAP5 code has been developed for best-estimate transient simulation of light-water-reactor coolant systems during severe accidents. The newest version of the code is SCDAP/RELAP5/MOD3. The US Nuclear Regulatory Commission (NRC) decided that there was a need for a broad technical review of the code by recognized experts to determine overall technical adequacy, even though the code is still under development. For this purpose, an eight-member SCDAP/RELAP5 Peer Review Committee was organized, and the outcome of the review should help the NRC prioritize future code-development activity. Because the code is designed to be mechanistic, the Committee used a higher standard for technical adequacy than was employed in the peer review of the parametric MELCOR code. The Committee completed its review of the SCDAP/RELAP5 code, and the findings are documented in this report. Based on these findings, recommendations in five areas are provided: (1) phenomenological models, (2) code-design objectives, (3) code-targeted applications, (4) other findings, and (5) additional recommendations.

EXECUTIVE SUMMARY

The SCDAP/RELAP5 code has been developed for best-estimate transient simulation of light-water-reactor (LWR) coolant systems during severe accidents. The newest version of the code is SCDAP/RELAP5/MOD3, which is intended to model both boiling-water-reactor (BWR) and pressurized-water-reactor (PWR) plants. A number of organizations, both foreign and domestic, are using or planning to use the current version. Although the quality control and validation efforts are progressing, there was a need to have a broad technical review by recognized experts to determine or confirm the technical adequacy of the code for the integrated and complex analyses it is expected to perform.

The objective of this report is to document the findings of the SCDAP/RELAP5 Peer Review Committee, which was formed to fulfill the charter described in the following section.

Committee Charter

The SCDAP/RELAP5 Peer Review Committee was created to (1) provide an independent assessment of the SCDAP/RELAP5 code through a peer review process, (2) determine the technical adequacy of SCDAP/RELAP5 for the complex analyses it is expected to perform, and (3) issue a final summary report describing the technical findings of the Committee.

Peer Review Process

The Committee developed and followed a multistep process for the SCDAP/RELAP5 Peer Review similar to the structured approach used in an earlier peer review of the MELCOR code. The steps in the process are to:

1. Identify design objectives for the code,
2. Identify targeted applications of the code,
3. Identify the code version to be reviewed (MOD3/Version 7X),
4. Identify and distribute the code document database to Committee members,
5. Review plant severe-accident scenarios available,
6. Develop a common perspective regarding technical adequacy,
7. Identify dominant phenomena for the plants and scenarios,
8. Define a "Standard of Technical Adequacy" to be used in developing findings,
9. Define a process for reviewing for technical adequacy,
10. Assess technical adequacy of individual phenomenological models and/or correlations within the code ("bottom-up" review),
11. Assess technical adequacy of the integral code against the design objectives and targeted applications ("top-down" review), and

12. Document findings in a summary report.

Detailed descriptions of each step in the review process are provided in Section 1, Introduction.

Major Findings

Background

The Idaho National Engineering Laboratory (INEL) staff provided strong support to the Committee and was very professional in the manner in which it presented information needed for the review. Once information was available and a document database was assembled, the Committee conducted a detailed review of the phenomenological models in the code to determine the technical adequacy of the individual models. To provide a process in which the overall technical adequacy of the SCDAP/RELAP5 code could be reviewed, three key elements were specified by the US Nuclear Regulatory Commission (NRC):

1. Code-design objectives,
2. Code-targeted applications, and
3. Success criteria associated with each design objective and targeted application.

The Committee used a higher standard for assessing technical adequacy than was employed in the peer review of the MELCOR code because the eventual goal for the SCDAP/RELAP5 code is to perform best-estimate, mechanistic simulations of the physical processes that might occur during a severe accident. If the Committee had used the same scale to measure the technical adequacy that had been used for the MELCOR code, several more phenomenological models would have been found adequate.

The NRC does not currently employ SCDAP/RELAP5 alone to assess severe-accident issues, but uses instead a combination of integral-code simulations coupled with supporting calculations from more detailed codes and expert consultation to analyze severe-accident issues.

To determine technical adequacy, the Committee examined the overall performance of the code relative to its design objectives and targeted applications. The findings were divided by the four intervals of a severe accident:

1. Initial Transient, Coolant Depletion, and Heatup Interval (before core uncovery);
2. Core Uncovery Interval (before the start of core damage);
3. Melt Relocation and Slump Interval (substantial damage); and
4. Core-Debris Material Inside Lower-Plenum Interval (possible lower-head failure).

On the basis of the success criteria for design objectives and targeted applications, the Committee determined overall technical adequacy of the code and provided recommendations for improvements.

Major Committee Findings

The Committee recognizes that considerable accomplishments have been made over the years in the development of computer codes for the analysis of severe accidents. This is true even though resources have been limited over the duration of the code development effort. Significant advances continue to be made in simulation capability and code improvements even as these findings are being published. The findings presented by the Committee should be interpreted in the context of the NRC's efforts to provide a quality computational tool for reactor safety research.

The NRC requested this peer review, and the NRC also has set high standards against which the technical adequacy of the SCDAP/RELAP5 code must be measured. Against these high standards, the code fared reasonably well considering the state of knowledge with respect to severe-accident phenomena present in Intervals 3 and 4. However, the Committee's challenge was to uncover as many deficiencies as possible that would prevent the code from meeting the NRC's high standards, and the findings in this section are the result of that challenge.

Major Findings on the Technical Adequacy of the Code Models

The Committee found that the technical adequacy of the code phenomenological models strongly depended on the interval of the severe accident. Many code models were technically adequate during the early intervals of an accident but were deemed inadequate as an accident progressed into the later intervals where core degradation, relocation, and possible vessel failure might occur. Only a few of the models that were deemed technically inadequate were also judged relatively unimportant in predicting core damage or the magnitude of the source term. Many key models were deemed technically inadequate and were also judged to be highly important to predicting core damage or the magnitude of the source term—specifically:

1. Fuel-rod liquefaction, flow, and solidification (Sections 2.9, 2.13);
2. Fission-product release, transport, and deposition* (Sections 2.10, 2.17, and 2.18);
3. Control rod and core structure, including grids (Section 2.11);
4. Debris heatup, heat transfer, fragmentation, and quenching in the core and lower plenum (Sections 2.13, 2.14, 2.15, and 2.25);
5. Molten pool formation, crust behavior, and convection in molten pools (Sections 2.13, 2.15, and 2.25); and
6. Heat transfer to lower head and vessel-head response (Sections 2.15, 2.16, 2.25, and 2.26).

It should be noted that a finding of technical inadequacy does not necessarily mean that all elements of a code model are inadequate. The Committee's measure of technical adequacy was so stringent that if only one part of a model was found to be inadequate, the entire model was deemed inadequate. The Committee recognizes that this does not give a measure of how close a model is to becoming technically adequate and suggests that future peer reviews attempt to provide another measure of adequacy.

* It was brought to the Committee's attention recently that the NRC does not intend to use the SCDAP/RELAP5 code for best-estimate, source-term prediction. The VICTORIA code being developed under NRC sponsorship is intended to fulfill this role.

Major Findings Relative to Code-Design Objectives

In general, the code did not meet all of its design objectives. However, until the code reaches the level where it can satisfy all of its design objectives and be used for all targeted applications, the NRC is employing the code in the following manner:

1. Simulating a severe accident to develop a better understanding of specific technical issues;
2. Performing a range of sensitivity studies to identify a range of results (the code is not being used in a predictive mode at the present time); and
3. Using expert opinion to supplement any lack of code models and enhance the knowledge and understanding of the phenomenological behavior.

The Committee did not review thermal-hydraulic models in RELAP5/MOD3 and has a significant concern that the RELAP5/MOD3 correlations and models will not be appropriate for geometries and flow configurations encountered in severe-accident scenarios.

The Committee found that the code generally had sufficient modeling detail to represent the key and important phenomena during the first two intervals of a severe accident. Modeling detail was less sufficient for the last two intervals of a severe accident, particularly for BWR plants; assessment calculations, especially for fission-product behavior, were very limited. The degree of parameterization increases as simulation moves into later intervals of the severe accident. In the final intervals of a severe accident, the code models were quite deficient and were capable of predicting only bounding estimates of expected behavior.

The code capability to predict important parameters within experimental uncertainty has not been fully demonstrated, and assessment activities must continue to systematically identify and evaluate uncertainties in the code models and database. The code has several user-defined parameters, some of which are deemed to have a major impact on the prediction of accident progression. The Committee also found that the code assessment previously undertaken has employed a wide range of code versions during the course of individual assessment calculations.

When code performance was examined, the Committee found that runtimes were acceptable for the analysis required and only became excessive if fission-product models were used. It was found that the setup time for a full-plant calculation was very long but commensurate with the complexity of the models and the facility.

A code-configuration-control procedure exists that allows complete tracking of code changes, testing, and documentation. However, adequate documentation has lagged significantly behind the release of the test versions of the code, creating substantial confusion from users and persons reviewing the technical adequacy of the code. The Committee is aware that the NRC is planning to modify the documentation to reflect the comments of the Committee, code users, and other sources.

To meet code-design objectives, code assessment should be expanded, code robustness should be improved, noding sensitivities should be identified, code documents should reflect the latest code, and assessment reports should accompany each code released.

Major Findings Relative to Code-Targeted Applications

The Committee found that the code was technically adequate for most simulation requirements in severe-accident Intervals 1 and 2, i.e., through core uncovery, and technically inadequate in general for simulation requirements in severe-accident Intervals 3 and 4 for both experimental and full-plant analysis (see Section 4 for more details). It is recognized that the database for later intervals is limited. These findings are based on a review of available publications and consideration (from the bottom-up review) of the technical adequacy of the code models that are required for these simulations. It should be noted, however, that the published information available for review relevant to the latest version of the code was very limited, and work is presently under way to provide additional documentation of code assessment and full-plant simulations.

The Committee also found that the code is not technically adequate to be employed by itself in its present state solely for detailed analysis or resolution of specific technical issues. It is recognized that the use of SCDAP/RELAP5 within a larger technical framework for integration of a severe-accident knowledge base into regulatory decision making is probably adequate. The code has not been shown to predict the dominant phenomena associated with specific technical issues by assessment against sufficient experimental results.

It was also determined that the code has not been used to benchmark and assess the MELCOR code, although there have been a limited number of simulation exercises where both SCDAP/RELAP5 and MELCOR results were available. This effort is hampered by a lack of sufficient data.

For the capability of the code to simulate the Three-Mile-Island-2 (TMI-2) accident with reasonable prediction of the dominant phenomena, the Committee found that present code models were adequate to allow prediction of phenomena in Intervals 1 and 2 (mostly the thermal-hydraulic portion) of an accident, but predictions in later intervals could not adequately be treated with the current version of the code (particularly melt relocation into the lower plenum).

Other Major Findings

The Committee found that there were inconsistencies between what is in the code documentation and what is in the code, and it is the Committee's understanding that work is presently under way to provide up-to-date code documentation with code releases. INEL has provided a good basis in the form of an assessment matrix, and because relatively limited attention has been paid to assessing later intervals of a severe accident, this matrix represents a good starting point (details are presented in Section 5).

The Committee found several difficulties with an assessment scheme that attempts to assess a code that is still under development, including a lag of code documentation behind code releases and an accurate determination of the exact code version used for various assessment activities. Finally, the Committee found that the image of the code in the user community generally needs some improvement, particularly with input/output specifications and external user support. The Committee recognizes that efforts are under way to alleviate some of these concerns. Furthermore, even though SCDAP/RELAP5 has some problems, it is nonetheless being used extensively in the international community because it reflects the state of the art.

Recommendations

The Committee recommends that improvements be made to the technically inadequate models that are important to predicting core damage and the source term, specifically:

1. Fuel-rod liquefaction, flow, and solidification (Sections 2.9 and 2.13);
2. Fission-product release, transport, and deposition* (Sections 2.10, 2.17, and 2.18);
3. Control rod and core structure, including grids (Section 2.11);
4. Debris heatup, heat transfer, fragmentation, and quenching in the core and lower plenum (Sections 2.13, 2.14, 2.15, and 2.25);
5. Molten pool formation, crust behavior, and convection in molten pools (Sections 2.13, 2.15, and 2.25); and
6. Heat transfer to lower head and vessel-head response (Sections 2.15, 2.16, 2.25, and 2.26).

To meet code-design objectives, code assessment should be expanded, code robustness should be improved, spatial and temporal noding sensitivity calculations should be performed, code documents should reflect the latest code version, assessment reports should accompany each code released, and code maintenance should continue.

To meet code-targeted applications, more full-plant calculations should be performed (such as additional accident sequences and reactor facilities, including BWRs), the Westinghouse transient natural circulation tests should be analyzed, and improvements should be made to models that affect reflood of a degraded core and lower-head-failure analysis. To address other Committee findings, code input/output should be streamlined, and code assessment goals should be made clear. A final recommendation is that code development continue to receive periodic independent code peer reviews so that the NRC can continue to ensure the quality of the tools being developed for safety analysis. Detailed explanations of Committee recommendations are provided in Section 6.

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* It was brought to the Committee's attention recently that the NRC does not intend to use the SCDAP/RELAP5 code for best-estimate, source-term prediction. The VICTORIA code being developed under NRC sponsorship is intended to fulfill this role.

1. INTRODUCTION

The SCDAP/RELAP5 code (Ref. 1-1) has been developed for best-estimate transient simulation of LWR coolant systems during large-break (LB) and small-break (SB), loss-of-coolant accidents (LOCAs), operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow, as well as severe accidents. The code has been designed as a detailed mechanistic code; numerically it solves mass, momentum, and energy conservation equations.

First-principle modeling is used in the code to the extent practical—not only in thermal hydraulics, but also in core-melt progression, fission-product release and transport, and hydrogen generation. The newest version of the code is SCDAP/RELAP5/MOD3, which can model both BWR and PWR plants. In the present two-tier analysis code structure used by the NRC, SCDAP/RELAP5 is a lower-tier, mechanistic, severe-accident code in contrast to a fast-running integral code of the upper tier.

A number of organizations, both foreign and domestic, are using or planning to use the current version. Although the quality control, model development, and validation efforts are seen to be proceeding, there is a need to have a broad technical review by recognized experts to determine or confirm the technical adequacy of the code for the serious and complex analyses it is expected to perform.

A peer review committee has been organized using recognized experts from the national laboratories, universities, SCDAP/RELAP5 user community, and independent contractors. Specific design objectives and targeted applications for the code, along with associated success criteria, have been provided by the NRC (Refs. 1-2 and 1-3). Using the NRC information, documentation, presentations provided by code developers and other technical sources, and a basic understanding of the underlying severe-accident phenomena, the Committee developed a consensus on the overall technical adequacy of the SCDAP/RELAP5 code.

The objective of this report is to document the findings of the SCDAP/RELAP5 Peer Review Committee.

1.1. Committee Charter

The charter of the SCDAP/RELAP5 Peer Review Committee was to (1) provide an independent assessment of the SCDAP/RELAP5 code through a peer review process, (2) determine the technical adequacy of SCDAP/RELAP5 for the complex analyses it is expected to perform, and (3) issue a final report describing the technical findings of the Committee.

1.2. Committee Membership

The Committee membership was selected to conduct a broad peer review of the SCDAP/RELAP5 code using recognized technical experts from universities, the national laboratories, the community of SCDAP/RELAP5 code users, and other technical organizations. The Committee membership selection was based on the expertise required to evaluate the specific code models, as well as the overall code performance. Brief experience summaries of each Committee member are presented in Appendix A. The Committee members are as follows:

Dr. Michael L. Corradini, University of Wisconsin, Madison
Dr. Vijay K. Dhir, University of California at Los Angeles
Dr. Tim J. Haste, AEA Technology, Winfrith, UK
Mr. Terry J. Heames, Science Applications International Corporation, Albuquerque
Mr. Richard P. Jenks (Committee Chair), Los Alamos National Laboratory
Dr. John E. Kelly, Sandia National Laboratories, New Mexico
Dr. Mohsen Khatib-Rahbar, Energy Research, Incorporated
Dr. Raymond Viskanta, Purdue University

1.3. Peer Review Process

The Committee implemented a process for conducting the peer review of the SCDAP/RELAP5 code. A substantial effort was undertaken previously to develop a multistep process for the MELCOR Peer Review (Ref. 1-4). Because that process proved very workable for reviewing the MELCOR code for technical acceptability, a nearly identical procedure was adopted by the SCDAP/RELAP5 Peer Review Committee.

1.3.1. Identify the Design Objectives

Both the design objectives and the associated success criteria for the SCDAP/RELAP5 code were provided by the NRC (Refs. 1-2 and 1-3):

1. Modeling detail shall be capable of representing key and important phenomena of severe-accident experiments, the TMI-2 accident, and anticipated plant accidents and transients.

General Success Criterion: The code can model the PWR and BWR coolant systems and operator actions plus experimental facilities used for code assessment.

a. Expected modeling uncertainties should be comparable to uncertainties in integral severe-accident experiments and TMI-2 accident conditions and results.

Success Criterion: Uncertainties in important parameters calculated by the code should be less than or equal to measured values. For example, if the uncertainties in the measured bundle temperatures and associated boundary conditions are +/-20%, then the success criteria for the code would be +/-20% for calculated temperatures.

b. User-defined parameters, other than those needed to define experiment or plant-unique features, should be eliminated where experimental or other credible bases exist to define those parameters.

Success Criterion: The code will not contain any user-defined parameters other than those noted.

2. The code should provide reasonable predictions of the in-vessel melt progression phenomena during the course of a severe accident. It should permit estimates of the uncertainties of severe core-damage predictions without requiring modifications to the code.

Success Criterion: The code predicts major trends for dominant phenomena on the basis of assessment against integral facility data. The code also predicts values of important parameters associated with dominant phenomena within measurement uncertainty when assessed against integral facility data. The code employed for these assessments would be the frozen-released code without any code modifications made during the period of application.

3. The code should be applicable for severe core-damage studies under various accident sequences for both PWRs and BWRs.

Success Criterion: The code can predict the core damage resulting from risk-dominant accident sequences identified by probabilistic risk assessment (PRA) studies for both PWRs and BWRs. Physical models, as well as component models, exist sufficiently to accurately predict dominant phenomena.

4. The code should be robust, portable, and fast running.

General Success Criterion 1: While runtime is machine dependent, the following general expectation can be used: Runtime should be reasonable so as to not handicap the user's ability to perform sensitivity/uncertainty analyses for the phenomena/conditions the code is designed to model. Runtime should be a small fraction of the time required to perform the entire analysis.

General Success Criterion 2: On the basis of user's guidelines and lessons-learned information in the code manual, code users shall be able to set up a plant model (e.g., input deck) to truly represent a full-scale LWR plant and successfully perform plant calculations for various severe-accident scenarios, which are in the domain of targeted applications of the code.

a. The code should not abort prematurely because of user-input errors or numerical nonconvergence but should exit with sufficient diagnostic messages for users.

Success Criterion: The code performs as noted.

b. Numerical precision should be compatible with modeling precision. Spatial convergence should be compatible with the modeling scale. Timestep control should be automatic.

Success Criterion: The code performs as noted.

c. The code should be transportable for mainframe and workstation computing machines.

Success Criterion: The code is transportable.

5. The maintenance of the code should follow accepted quality-assurance (QA) standards for configuration control, testing, and documentation.

General Success Criterion: The code QA procedures and associated documentation should be sufficient to allow the certification of the code for ANSI/ASME NQA-1 or the equivalent where required for NRC applications.

a. All code changes should be controlled and verified by redundant means.

Success Criterion: The code changes should be made as noted.

b. Testing standards and benchmarks should be defined for all versions released for production applications.

Success Criterion: The code performs as noted.

c. Documentation should define the theoretical basis, limits of applicability, and testing or assessment results of the code.

Success Criterion: The documentation should satisfy the stated criteria.

1.3.2. Identify Targeted Applications

Both the targeted applications and the associated success criteria for the SCDAP/RELAP5 code were identified by the NRC (Refs. 1-2 and 1-3). The targeted applications for the SCDAP/RELAP5 code are:

1. Experimental analysis and support for in-vessel, severe-accident experimental programs such as CORA, PBF, LOFT, and NRU.

Success Criterion: The code has been or can be used to analyze these facilities and can provide reasonable predictions of associated dominant phenomena. (Compared with experimental data, the calculated results will be within the experimental uncertainty bands.)

2. LWR plant analysis with and without water addition.

Success Criterion: The code has been or can be used to analyze LWRs and can provide reasonable predictions of associated dominant phenomena. The code must have been shown to predict the dominant phenomena associated with these two actions (with or without water addition) by assessment against sufficient experimental results.

3. A selected detailed analysis for specific technical issues includes lower-head-failure analysis, influence of water addition, natural circulation, hydrogen generation upon reflood, and accident management evaluations.

Success Criterion: The code has been or can be used to provide a detailed analysis of these specific technical issues predicting the associated dominant phenomena with reasonable agreement. The code must have been shown to predict the dominant phenomena associated with these specific technical issues by assessment against sufficient experimental results.

4. The MELCOR benchmarking and assessment.

Success Criterion: The code has been used to benchmark and assess the MELCOR code in-vessel behavior, at least for integral experiments.

5. TMI-2 accident evaluation.

Success Criterion: The code has been or can be used to evaluate the TMI-2 accident with reasonable prediction of the dominant phenomena. (Compared with the TMI-2 data, the calculated results will be within the measured uncertainty bands.)

1.3.3. Identify the Code Version to Be Reviewed

The Committee primarily focused on Code Version SCDAP/RELAP5-MOD3[7X].* However, the Committee considered and acknowledged the planned development program as part of its review effort.

1.3.4. Identify and Distribute the "Document Database" to Committee Members

A SCDAP/RELAP5 Document Database was compiled and continually updated as new information was identified. Included in the Document Database are: (1) reference reports, (2) published INEL papers, (3) external (to INEL) reports and papers, (4) correspondence and memoranda, (5) other information, and (6) committee documents and findings.

1.3.5. Select Plant Severe-Accident Scenarios

INEL's Surry PWR model has been exercised more than any other at INEL. An Oconee PWR model also exists, but this model has not been exercised as much. INEL has only limited BWR SCDAP/RELAP5 experience. Oak Ridge National Laboratory (ORNL) has developed BWR models for incorporation into the code. A set of plant models (PWR and BWR) were identified, and available simulations were reviewed by the Committee. Other plant calculations have also been performed (see Section 5.6) but not reviewed by the Committee.

1.3.6. Develop a Common Perspective

The Committee members hold a variety of perspectives in deciding the technical acceptability of SCDAP/RELAP5. To develop a common perspective on "how good is good enough," three important factors were considered related to severe-accident phenomena:

1. Knowledge of Physical Processes. The current level of scientific knowledge about severe-accident processes varies. The physics of some physical processes are well understood, while the physics of other physical processes are partially or poorly understood.

* Note that 7X refers to Versions 7a through 7af, although most calculations used only 3 or 4 of these 32 versions.

Therefore, the Committee cannot expect SCDAP/RELAP5 to accurately model any phenomena that the scientific community has not yet fully understood. The SCDAP/RELAP5 code should therefore permit the consideration of uncertain physical processes by other means (e.g., sensitivity studies).

2. SCDAP/RELAP5 Physics. The objective of any modeling effort should be to mathematically represent the physical processes correctly, because incorrect mathematical representations cannot be expected to accurately predict reality. SCDAP/RELAP5 is a best-estimate, mechanistic code, and one of the design objectives (from Section 1.3) is that it should have "modeling uncertainties comparable to uncertainties in integral severe-accident experiments." Therefore, sufficiently accurate mathematical representations of physical phenomena should be present in the model to accurately reproduce the physical behavior.

A model that captures the major features or trends of dominant phenomena and leads to the physically correct end state will be defined to be a "zeroeth-order" model. A first-order model improves the characterization of features or trends that affect event timing and magnitude and more accurately represent the passage to the end state. All models that are important to the prediction of dominant phenomena in severe-accident simulations must be of first order for the code to be considered best estimate and mechanistic.

3. Importance. Some models are more important than others in determining the course of a severe-accident simulation. Importance reflects the code's ability to predict (1) the source term to the containment, and (2) the amount of core damage, which includes hydrogen generation, melt ejection characteristics, and the failure of the reactor coolant system. For those more-important models, either the physics must be well understood and modeled or the sensitivity of results must be well quantified. However, for models of less importance (i.e., having less impact on the outcome of the simulation), the physics and modeling are less important, as are the sensitivity studies.

An importance value will therefore be determined for each model reviewed. The technical acceptability of the more important models will carry the most weight in determining overall technical acceptability of the code. Models that are inadequate, but are of less importance, should be flagged for future code improvement but should not impact on the Committee's statement of overall code technical adequacy. Appendices C, D, and E provide additional details regarding importance and ranking of severe-accident phenomena.

1.3.7. Identify Dominant Phenomena for the Plants and Scenarios

Dominant phenomena for both PWR and BWR plants have been developed on the basis of the following "four-interval" scenario given in Appendix C of Ref. 1-1.

1. Initial Transient, Coolant Depletion, and Heatup Interval (before Core Uncovery);
2. Core Uncovery Interval (T < 1500 K);
3. Melt Relocation and Slump Interval (Substantial Damage); and
4. Core-Debris Material Inside Lower-Plenum Interval (possible lower-head failure).

The objectives of this approach were to (1) identify and list severe-accident phenomena, (2) identify important or dominant severe-accident phenomena, and (3) check for the existence of SCDAP/RELAP5 models for the dominant severe-accident phenomena. The goals of this step are important, and the Committee has attempted to fully implement it. The in-vessel phenomena identified in Ref. 1-1 were used as a starting point by the Committee members to determine if a complete and technically acceptable set of models is present in the code.

1.3.8. Define a "Standard of Technical Adequacy" to Be Used in Developing Findings

A two-stage approach to defining a standard for technical acceptability was used by the Committee. Stage 1 is applicable to the detailed models in SCDAP/RELAP5 (the bottom-up view). Stage 2 is applicable to the total or integrated SCDAP/RELAP5 code (the top-down view).

<u>Stage 1:</u> The Committee's standards for technical adequacy of the individual SCDAP/RELAP5 models are that (1) the model pedigree is known, documented, and acceptable, (2) the model is used appropriately or stated in another manner, and the application of the model is acceptable, and (3) the prediction of, or fidelity to, the dominant phenomena modeled is acceptable.

Stage 2: The Committee's standards for technical adequacy of the total or integral code are that (1) the total code is applicable, and (2) the prediction of integral phenomena is acceptable.

1.3.9. Define a Process for Reviewing for Technical Adequacy

Having defined the Standard of Technical Adequacy, the Committee performed two review actions.

A **bottom-up review** was conducted by examining the pedigree, applicability, and fidelity of the many individual models and closure relationships in SCDAP/RELAP5. This was accomplished by reviewing the Models and Correlations document and other related literature from the SCDAP/RELAP5 Document Database, once this database was fully developed. A classification scheme was proposed, assigning a category number to each model consistent with the methods used in the MELCOR Peer Review.

A **top-down review** was conducted on the applicability and fidelity of the total code by examining benchmarking efforts available. The top-down review was documented for the specific version of the code that was reviewed. In the absence of such information, code-to-code comparisons and expert opinions became stronger elements of the top-down stage of the review.

1.3.10. Assess Technical Adequacy of Individual Models and/or Correlations Within the SCDAP/RELAP5 Phenomenological Models (Bottom-Up Review)

A bottom-up review was conducted by examining the pedigree, applicability, and fidelity of the individual models in SCDAP/RELAP5. The code documentation database provided most of the necessary model information. The Committee employed a set of classifications and definitions as depicted in Fig. 1-1 and defined in Table 1-I. Findings of Categories 1, 2, or 3 indicate that the code model is technically adequate. Findings of Categories 4, 5, and 6 indicate its technical inadequacy. A finding of Category 7 indicates the code model's technical inadequacy but low importance. (Highlighted regions in Fig. 1-1 show technically acceptable categories for SCDAP/RELAP5, as well as the related regions for the MELCOR code, for comparison purposes.)

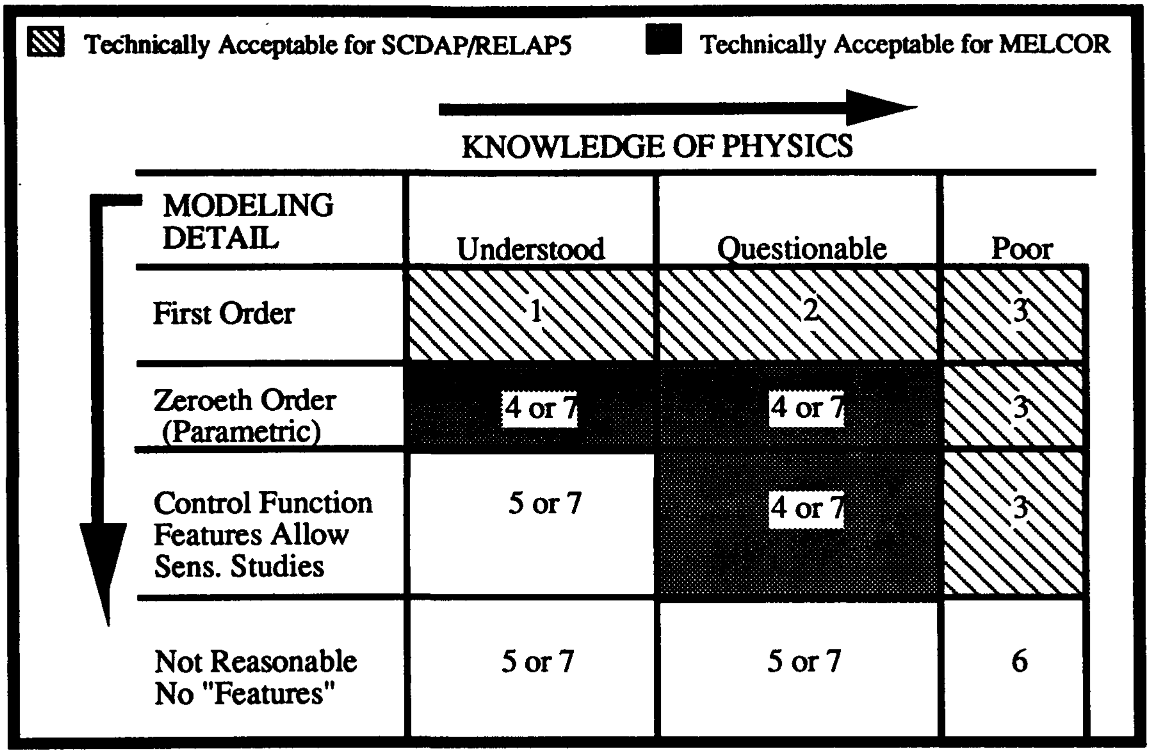

Fig. 1-1.
SCDAP/RELAP5 bottom-up review matrix of technical adequacy findings.

The Committee conducted reviews of the individual models and correlations and provided documentation of their results in the format shown in Table 1-II. The model review summaries (provided as subsections to Section 2) provide a qualitative overview of the technical adequacy of the code using the standard-of-model pedigree, model applicability, and model fidelity. The detailed reviews (provided as sections to Appendix E) give additional information to support the overall qualitative assessment.

1.3.11. Assess Technical Adequacy of the Integral Code Against the SCDAP/RELAP5 Design Objectives and Targeted Applications (Top-Down Review)

SCDAP/RELAP5 is designed to describe the response of the primary-reactor coolant system during a severe accident up to and including the point of reactor vessel or system failure (Ref. 1-2). System thermal hydraulics, core-damage progression, hydrogen generation, and fission-product behavior are described.

Table 1-I
SCDAP/RELAP5 Bottom-Up Review
Classification of Findings

Finding	**Definition**
Category 1	The severe-accident phenomena are generally understood, and the physics are correctly represented with at least a first-order model.
Category 2	Questions exist regarding the severe-accident phenomena, but reasonable physics (in the context of expert judgment) are represented with at least a first-order model.
Category 3	A poor understanding exists of the severe-accident phenomena. A model may have been provided in the code, or features are available in the code to perform sensitivity studies over the currently understood ranges of phenomenological behavior.
Category 4	The severe-accident phenomena may be generally understood, but the physics are represented at best by a zeroeth-order model, or features are available in the code to perform sensitivity studies over the currently understood ranges of phenomenological behavior.
Category 5	The severe-accident phenomena may be generally understood, but (1) the physics are represented only by control function features that allow the user to perform sensitivity studies, (2) the model is not reasonable, or (3) no modeling features exist at all.
Category 6	A poor understanding exists of the severe-accident phenomena, and no model is provided at all. Features are not available in the code to perform sensitivity studies over the currently understood ranges of phenomenological behavior.
Category 7	A finding of either Category 4, 5, or 6 would apply, but the phenomena being modeled are not of sufficient importance to markedly influence either the major features or trends of the severe accident. (NOTE: This category relates more to prioritizing the correction of defects than condoning their continuance.)

Table 1-II

Format for SCDAP/RELAP5 Bottom-Up Detailed Review Documentation

	Section 2. Model Review Summary Format
1.	Phenomenological Description
2.	Qualitative Perspective
3.	Technical Adequacy
	Appendix E. Detailed Review Format
1.	Model Description and Pedigree
2.	Applicability/Physical Reasonableness of Model to Predict Dominant Phenomena
3.	Implementation Within the Code
4.	Results of Model Sensitivity Studies
5.	Results of Benchmarking/Validation Studies
6.	Identified Deficiencies and Recommendations for Model Improvements
7.	Importance of Model to Predict Dominant Phenomena
8.	Technical Adequacy of Model

A top-down review was conducted by examining the applicability and fidelity of the total code. Previous benchmarking efforts were reviewed. In the absence of this information, plant calculations and expert opinion were employed.

The Committee has documented its findings in relation to the code objectives and targeted applications. For each code objective and targeted application, the Committee defined areas to be examined to adjudge the technical acceptability of the integral code.

1.3.12. Document Findings in a Summary Report

This report documents the findings of the SCDAP/RELAP5 Peer Review Committee and the process used to develop those findings.

1.4. Presentation of Committee Findings

The Committee decided to present its findings in a format similar to that employed for the previously cited MELCOR review. Summarized findings for the detailed review of code phenomenological models are given in Section 2. More detailed findings for these bottom-up reviews are given in Appendix E. The findings relative to the code objectives are presented in Section 3; findings relative to the targeted applications are presented in Section 4; other findings are presented in Section 5. SCDAP/RELAP5 peer review assignments for individual Committee members are presented in Table I-III.

Table 1-III
SCDAP/RELAP5 Peer Review Assignments

Description	Individual A	Individual B
Detailed Models (Bottom-Up)		
Structural Behavior Models	**VKD and MKR**	TJH1 and RV
• Material oxidation model	VKD and MKR	TJH1 and RV
• Nuclear-heat model	VKD and MKR	TJH1 and RV
• Effective materials properties	VKD and MKR	TJH1 and RV
• Fuel-state models	VKD and MKR	TJH1 and RV
• Heat-conduction model	VKD and MKR	TJH1 and RV
• Cladding-deformation models	VKD and MKR	TJH1 and RV
• Fuel-rod, internal-gas pressure	VKD and MKR	TJH1 and RV
Core-Degradation/Relocation Models	**MKR and VKD**	RV and TJH1
• Liquefaction, flow, and solidification	MKR and VKD MKR and VKD	RV and TJH1 RV and TJH1
• Fuel-fission-product release	MKR and VKD	RV and TJH1
• Control-rod and core structure	MKR and VKD	RV and TJH1
• Radiation heat-transfer model	MKR and VKD	RV and TJH1
• Core-region debris modeling	MKR and VKD	MLC, RV, and TJH1
• Core-slumping models	MKR and VKD	MLC, RV, and TJH1
• Lower-plenum debris heatup	MKR and VKD	MLC, RV, and TJH1
• Structural creep-rupture model	MKR and VKD	RV and TJH1
Aerosol and Fission Products	**TJH2**	MKR
• Aerosol agglomeration models	TJH2	MKR
• Aerosol particle deposition	TJH2	MKR
• Vapor evaporation/condensation	TJH2	MKR
• Heterogeneous chemical reaction between chemical species and walls	TJH2	MKR
Materials Properties	**TJH2**	TJH1 and JEK
Decay-heat distributions for volatiles released following fuel disruption	**RV**	TJH2
• Fission-product decay heat		
Decay-Heat Deposition	**RV**	TJH2
• Energy deposition model	RV	TJH2
• Gamma-attenuation, complete-absorption model	RV	TJH2
Severe-Accident Thermal Hydraulics	**JEK**	RV

Table 1-III
SCDAP/RELAP5 Peer Review Assignments
(cont.)

Description	Individual A	Individual B
Integral Code Performance (Top-Down)		
Code Objectives Issues	**JEK**	TJH1
• Systems code uncertainty	JEK	TJH1
• Systems code architecture	JEK	TJH1
• Systems code numerics	JEK	TJH1
• Systems code portability	JEK	TJH1
• Code speed/robustness	JEK	TJH1
• Code QA/configuration control	JEK	TJH1
• Systems code documentation	JEK	TJH1
• PWR severe-accident phenomena	JEK	All Committee
• BWR severe-accident phenomena	JEK	All Committee
• Ranking of dominant severe-accident phenomena	All Committee	All Committee
Code Application Issues	**MLC**	JEK
• CORA	MLC	JEK
• PBF	MLC	JEK
• LOFT	MLC	JEK
• NRU	MLC	JEK
• Lower-head-failure analysis	MLC	JEK and VKD
• Water addition to degraded core	MLC	JEK and VKD
• Natural circulation	MLC	JEK and RV
• Reflood H_2 generation	MLC	JEK
• Accident management	MLC	JEK and VKD
• MELCOR benchmarking and assessment	MLC	JEK
• TMI-2 accident evaluation	MLC	JEK
Other Findings	**TJH1**	MLC

<u>Committee Members' Initials:</u>

Corradini	MLC	Kelly	JEK
Dhir	VKD	Khatib-Rahbar	MKR
Haste	TJH1	Viskanta	RV
Heames	TJH2		

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1-2. F. Eltawila, US Nuclear Regulatory Commission, defining SCDAP/RELAP5 "Design Objectives" and "Targeted Applications," letter to R. Jenks, Los Alamos National Laboratory (September 10, 1991).

1-3. F. Eltawila, US Nuclear Regulatory Commission, defining SCDAP/RELAP5 "Success Criteria for Each Design Objective and Targeted Application," letter to R. Jenks, Los Alamos National Laboratory (October 22, 1991).

1-4. B. E. Boyack, V. K. Dhir, J. A. Gieseke, T. J. Haste, M. A. Kenton, M. Khatib-Rahbar, M. T. Leonard, and R. Viskanta, "MELCOR Peer Review," Los Alamos National Laboratory report LA-12240 (March 1992).

2. BOTTOM-UP REVIEW OF SCDAP/RELAP5 PHENOMENOLOGICAL MODELS

The objective of the bottom-up review process was to determine the technical adequacy of individual phenomenological models in the SCDAP/RELAP5 code. To accomplish this objective, the Committee used category numbers that identified to what degree a given model was technically adequate (see Step 10 of the Peer Review Process described in Section 1.3). The category numbers are depicted and described in Fig. 1-1 and Table 1-I, respectively.

A model that was determined to have a category number of 1, 2, or 3 was deemed to be technically acceptable; otherwise, the model was deemed unacceptable and not necessarily adequate (Categories 4, 5, or 6) or unimportant (Category 7).

Detailed reviews of each SCDAP/RELAP5 phenomenological model are presented in Appendix E. However, the Committee decided that a summary of the phenomenological package reviews was advisable, and the objective of this section is to present such reviews of each phenomenological model.

Each summary review is divided into three elements: (1) a brief description of the phenomenological package; (2) a qualitative perspective that considers, for example, factors such as the level of modeling detail and whether the models were developed or imported for SCDAP/RELAP5; and (3) a discussion of the technical adequacy of each model. Table 2-I summarizes the findings for technical adequacy by interval for each of the code models reviewed and identifies the order in which the model reviews will be presented in this section.

Table 2-I
Technical Adequacy Categories by Interval for SCDAP/RELAP5 Phenomenological Models[a]

Section/Model	Interval 1	Interval 2	Interval 3	Interval 4
2.1. Material Oxidation	7	1	5	5
2.2. Nuclear Heat	7	7	5	-[b]
2.3. Electric Fuel Rod	1	1	1	-
2.4. Effective Materials Properties	1	1	1	-
2.5. Fuel State	7	4	4	-
2.6. Heat Conduction	1	1	4	4
2.7. Clad Deformation	1	4	4	-
2.8. Fuel-Rod Internal Gas	1	4	-	-
2.9. Fuel-Rod Liquefaction, Flow, and Solidification	-	-	3	4
2.10. Fuel Fission-Product Release	-	1	4	7
2.11. Control Rod and Core Structure:				
control rods	1	4	4	-
flow shroud	1	1	1	-
grid	-	-	4	-
2.12. Radiation Heat Transfer	-	1	5	4
2.13. Core Debris	-	-	3	3
2.14. Core Slump	-	-	3	6
2.15. Lower-Plenum Debris	-	4	3	5
2.16. Structural Creep:				
creep	-	-	-	4
lower-head penetration	-	-	-	4
2.17. Aerosol Agglomeration	-		1[c]	1[c]
2.18. Aerosol Particle Deposition	-	-	1	1
2.19. Condensation/ Evaporation	-	7	7	7
2.20. Chemical Reactions	-	-	7	7
2.21. Materials Properties:				
liquid	1	1	1	2
gas	-	-	5	5
2.22. Fission-Product Decay	-	4	4	-
2.23. Decay-Heat Energy	-	1[d]	7	-
2.24. Decay-Heat Gamma Attenuation	-	1[d]	7	-
2.25. Severe-Accident Thermal Hydraulics	1	1	4	4

[a]Applies to MOD3/Version 7X that was reviewed by Committee.
[b]Not applicable to this interval.
[c]Not implemented correctly.
[d]Not strictly first order but best practical approach considering the state of the art.

A severe accident is typically divided into four intervals:

INTERVAL	DEFINITION
Thermal-Hydraulic and Neutronic Transient	Time from accident initiation until superheat in core
Core Uncovery	Time from superheat in core until maximum core temperature is 1500 K
Core Heatup/Oxidation/ Relocation and Slumping in Core	Time from when core is at 1500 K, through the oxidation transient, and up to formation of molten pool
Lower Plenum and Vessel Failure	Time from when molten material relocates into lower plenum until vessel failure

While these intervals are somewhat arbitrary, the Committee decided that this partitioning was a reasonable approach for the review process. A complete description of these separate intervals is given in Section 3.

The Committee found that the technical adequacy of each model was dependent on the interval of the accident. An example of interval dependency (see Table 2-I) is that the material oxidation modeling was unimportant (Category 7) during Interval 1 (the thermal-hydraulic and neutronic part of the accident). Material oxidation modeling was deemed to be technically adequate (Category 1) for the core uncovery interval but technically inadequate (Category 5) for later intervals of the accident (core heatup through vessel failure). In addition to the models identified in Table 2-I, Section 2.26 presents options for additional models currently being developed or upgraded.

2.1. Material Oxidation Model

2.1.1. Phenomenological Package Description

The oxidation of zircaloy in a high-temperature steam environment is treated using parabolic rate equations with parameters taken from well-known literature sources. Hydrogen production and steam removal are considered, and oxidation is limited by the availability of oxidizable metal or of steam. The heat generation is taken into account in the heat-conduction models. Material layers are assumed to oxidize in sequence, and completely oxidized layers are assumed to present no obstacle to the oxidation of layers underneath. Oxidation of the inner surface of ballooned, ruptured cladding is considered by simply doubling the calculated oxidation rate. There is no model for the oxidation of material during relocation, in porous or cohesive debris beds, nor in the plena of intact fuel rods.

2.1.2. Qualitative Perspective

The model is basically reasonable for oxidation of fuel cladding in intact geometries, although there can be minor inconsistencies in the results. The hydrogen production for structures such as the upper plenum, which can be significant, is not tracked by the code. The transition from parabolic to cubic kinetics seen for zircaloy below 1000°C is not modeled. However, the most serious deficiencies lie in the modeling of oxidation in the later phases of fuel-rod degradation during and after significant relocation of zirconium-rich eutectics, for which materials oxidation kinetics data are sparse. In these later stages, hydrogen production is generally underpredicted by the model.

Additionally, the sharp increase of oxidation and hydrogen production often seen in the reflood interval cannot be modeled by the code; a possible mechanism for this increase is the quench-induced shattering of existing oxide shells followed by rapid oxidation of the newly exposed metal. This underprediction of hydrogen production has accident management implications. It is understood that recent work by INEL is addressing this deficiency, but documentation of this work was not available at the time of the peer review.

2.1.3. Technical Adequacy

The modeling is technically adequate for intact geometry, with only minor improvements recommended to the kinetics model to account for cubic time dependence where appropriate. Oxidation of cladding in fuel-rod plena and oxidation of intact steel-core structures need to be modeled by the code. The modeling is not technically adequate for degraded geometry. The processes governing oxidation in the later intervals of accident sequences need more detailed study, especially for reflood situations, and improved models should be introduced as a matter of high priority.

2.2. Nuclear-Heat, Fuel-Rod Model

2.2.1. Phenomenological Package Description

The total power is assumed to be separable in time and space and is a sum of prompt fission power and the decay-heat power inside the reactor core.

The time-dependent prompt neutron power amplitude is provided as input to the code. The decay-heat amplitude is determined on the basis of the ANSI/ANS-5.1-1979 decay-heat standards, considering fission products with corrections to neutron capture (on the basis of an empirical relation) and contributions to decay heat due to the decay of ^{239}U and ^{239}Np.

For prompt neutrons, the axial power-peaking factors can also vary as a function of time on the basis of user-supplied distributions. On the other hand, the decay-heat, axial power-peaking factors are allowed either to adjust themselves exponentially to the prompt power-peaking factors or are supplied through separate user-input values. In SCDAP/RELAP5, it is assumed that radial power distributions remain time invariant.

The decay-heat, radial power-peaking factor is assumed to be identical to that of prompt neutron power for nonfuel components, while for the fuel components, it is assumed that gamma-ray energy is ~one-half of the decay power within fuel components. The radial power-peaking factor is then adjusted accordingly.

2.2.2. Qualitative Perspective

The assumed recoverable energy of 195.33 MeV/fission is incorrect. Typically, there are ~6.8 MeV/fission more due to activation of the structural material. This approximation is not justified.

SCDAP neglected the effects of delayed neutrons after reactor shutdown. The approach of forcing the prompt neutron power amplitude as a means of simulating delayed neutrons is wrong.

The SCDAP nuclear-heat model is inadequate for applications to high-burnup-fuel LWRs because it does not include the fission of ^{238}U and ^{239}Pu.

Neutron capture correction factors (G factors) are not applied by RELAP5. This creates inconsistencies between SCDAP and RELAP5 parts of the package.

During severe accidents, reduction factors are applied to the fission-product decay heat due to loss of volatile fission products from the fuel matrix. Justifications are not provided for neglecting the effect on decay heat due to the loss of lower-volatility species (i.e., tellurium, barium, strontium, etc.). Adjustments to both prompt and decay-power contributions are made to correct for a significant movement of mass as a result of fuel-rod degradation and relocation. The decay-heat term is corrected for changes of fuel mass and density. The prompt power amplitude is multiplied by the ratio of current material density to initial material density as a way to account for any fuel disruption or phase change. This approximation is not expected to be valid during core disruption and is therefore not justified.

RELAP5 provides an alternative point kinetics and one-dimensional diffusion capability for use with reactor scenarios. However, the RELAP5 kinetics models were not evaluated as part of this peer review effort.

2.2.3. Technical Adequacy

The present models for the most part rely on user-specified inputs for the determination of spatial power-generation rates during severe-accident conditions. This is consistent with SCDAP/RELAP5 objectives and targeted applications. Several approximations have been noted that are not fully justified; these include:

1. No application of G factors by RELAP5. This creates inconsistencies between SCDAP and RELAP5 parts of the package.

2. The assumed recoverable energy neglects ~3% of the energy due to activation of the structural material.

3. SCDAP neglected the effects of delayed neutrons after reactor shutdown. The statement that "user can force consideration of . . . as an additional prompt neutron heat source" is wrong.

4. The SCDAP nuclear-heat model is inadequate for applications to high-burnup-fuel LWRs because it neglects the fission of ^{238}U and ^{239}Pu.

Overall, the approach is judged to be reasonable. However, the above-noted inadequacies must be addressed.

2.3. Electrically Heated Fuel-Rod Model

2.3.1. Phenomenological Package Description

A model is provided for the electrically heated fuel-rod simulators of the type used in the CORA facility, which is used for experimental investigation of early-interval melt progression. The model calculates the partition of power generation between the main tungsten heating element and the lead-in electrodes, treats axial and radial heat conduction (in the tungsten-heated section only), and deals with oxidation and ballooning of the zircaloy cladding in the same way as for standard fuel rods.

2.3.2. Qualitative Perspective

The model provides a physically reasonable first-order description of the specific features of the CORA fuel-rod simulators. In particular, the strong temperature dependence of the axial partition of electrical energy is adequately modeled. However, the two-dimensional, heat-conduction model is not extended into the plenum regions, so the true axial boundary conditions are not precisely simulated, and oxidation in these regions is not treated. There is the potential for geometrical inconsistencies if nonstandard tungsten-core diameters are input. The presence of the tungsten cores is not taken into account in the calculation of the UO_2 inventory nor in the fuel dissolution model. Thus, the models employed for simulation in later intervals cannot be used in a CORA environment, owing to limitations in the heat-conduction model.

2.3.3. Technical Adequacy

The modeling is basically sound and deals with the significant phenomena specific to the CORA simulator design. Some minor changes are required to extend the heat-conduction and generation models into the plenum regions and to ensure consistency internally and with fuel relocation models. The fact that the later-interval phenomenon (debris formation) cannot be modeled is only really significant in CORA tests where quenching is employed. It should be noted that different heater designs will require production of different experiment-specific models.

2.4. Effective Materials Properties

2.4.1. Phenomenological Package Description

This model provides volume-averaged thermophysical properties (density and specific heat product and thermal conductivity) of the fuel rod, control rod, or flow shroud. Effective volumetric heat generation for the heat-conduction element of volume is calculated, and integral transformation for treating phase change is also described.

Parallel and Series Resistance. The effective materials properties and effective volumetric heat generation for heat-conduction element volume are approximated by volumetric averaging on the basis of the parallel and series-resistance method. The averaging is for one-dimensional Cartesian and cylindrical geometries. If the thermophysical properties or heat generation are functions of temperature, the properties are averaged over the temperature range of each element. Layers specified by geometry-independent and geometry-dependent properties are considered to exist. The former include unirradiated fuel, zircaloy cladding, ZrO_2, liquid Zr-U-O_2, frozen Zr-U-O_2, and structural and absorber materials. The latter include cracked fuel, relocated fuel, and gaps and voids.

Integral Transformation. The concept of effective heat capacity is introduced to account for phase transformation at the phase-change front. This is a well-established procedure. To account for the enthalpy change during phase transformation, a product of effective volumetric heat capacity and a small temperature jump are introduced. This product is equivalent to the latent heat of fusion. The effective volumetric heat capacity is then used in the corresponding heat-conduction element in computing the temperature when a phase change occurs within an element.

Effective Volumetric Heat Generation. Volumetric heat generation within a heat-conducting component is due to both volumetric and surface (i.e., oxidation and dissolution) contributions. The effective volumetric heat generation in a heat-conduction element with several material layers is computed by averaging the contributions across the layers from the component center to the component surface.

2.4.2. Qualitative Perspective

The modeling of the thermophysical properties follows well-established principles and contains appropriate detail. The treatment is for the most part state of the art, and there appear to be no serious errors.

2.4.3. Technical Adequacy

On the whole, the model is judged to be adequate. The principal needs are for some additional discussion and clarification of the parallel-resistance concept. In the long term, the use of either the series or parallel thermal resistance concepts for layers specified by geometry-dependent models of effective thermal conductivity will need to be assessed. On strictly theoretical grounds, a correct effective thermal conductivity cannot be computed on the volume-averaging approach, particularly for the parallel-resistance concept, when there is a large difference in thermal conductivities between adjacent material layers. The model does not calculate the properties of debris beds and therefore is not applicable to later intervals of an accident. The average thermal conductivity is not calculated correctly; rather, an inverse of the thermal conductivity that relates to the thermal resistance is calculated.

2.5. Fuel-State Models

2.5.1. Phenomenological Package Description

This model defines the material structure in the fuel- and simulator-rod models, adjusts the axial power profile to allow for fuel relocation, and specifies the temperatures of grid spacers. A weighted-average method is used to calculate the axial peaking factors at axial locations where relocated crusts are present, using the peaking factors in the nodes from where the relocated material came. This applies to delayed sources of heat generation; peaking factors for prompt sources are user defined. The pedigree of the physical modeling was not established in the documentation.

2.5.2. Qualitative Perspective

The model for axial peaking factors is inappropriate at high burnup, where radial power-peaking factors may be large, and in recriticality situations. Although a model is available for the slumping of fuel fragments into ballooned regions, this is not invoked in the reviewed version of the code. This could lead to underpredictions of temperature in strongly ballooned regions. The part of the model dealing with definition of the material layers and compositions is judged to be adequate.

2.5.3. Technical Adequacy

The model for axial peaking is essentially a zeroeth order treatment, which would be appropriate for a PRA-level code. Consideration should be given to reintroducing the fuel fragment slumping model, taking due account of interactions with other models such as that for fission-product release. If calculations for high-burnup fuel are to be performed, the changes in radial power-peaking factors will need to be considered.

2.6. Heat-Conduction Model

2.6.1. Phenomenological Package Description

A finite-element approach is used to solve one-dimensional transient conduction equations in fuel rods and structures. Both plate-type and cylindrical geometries are analyzed. Any of the three types of boundary conditions can be applied. In obtaining the solution for the temperature field, the Galerkin method of weighted residuals is employed. Temperature-dependent effective thermal properties and volumetric heat generation are used. The model is only applicable to intact structures. Heat-transfer coefficients at the structure surface are obtained from the RELAP5 code and were not reviewed. Recently, the model has been upgraded to two dimensions. An alternating-direction-implicit scheme is used to solve for the temperature in the axial and radial directions.

2.6.2. Qualitative Perspective

The model as described in the documents has one-dimensional and two-dimensional options, which should be appropriate during the heatup interval of a severe core-damage accident. A two-dimensional conduction model is needed for boiloff and quenching conditions. The numerical results are dependent on the type of boundary conditions used. At present, significant concern exists with respect to appropriateness of the RELAP5-imposed boundary conditions in a particular situation. The model does not appear to include failed and partially molten structures in regions where a debris bed is formed.

2.6.3. Technical Adequacy

The model appears to be adequate for steady-state and transient simulations as long as appropriate boundary conditions are imposed during the core heatup and degradation processes. The model is limited to intact structures. No meaningful tests for model validations appear to have been performed.

2.7. Cladding-Deformation Models

2.7.1. Phenomenological Package Description

The model for clad ballooning is based on standard calculations of stresses in a thick-walled tube using the Lame-Clapeyron equations and on plasticity using the von Mises/Prandtl-Reuss approach. An axisymmetric model is used at low strains, changing to a localized nonaxisymmetric model when an instability strain (calculated by the code or input by the user) is exceeded. The localized ballooning model calculates azimuthal variation of strain resulting from a prescribed asymmetry in fuel temperature. The rupture criterion is either a temperature-dependent stress limit (from MATPRO subroutines) or a user-defined strain. Flow-area reduction resulting from ballooning is calculated using an experimentally derived probabilistic model.

2.7.2. Qualitative Perspective

The physical processes involved in clad ballooning are now well understood. The treatment in the code is in many respects first order and adequate for situations where oxidation is unimportant. However, at higher temperatures (over ~950°C), the effects of oxidation in strengthening the cladding and reducing the rupture strain through stress concentration at oxide cracks can be very significant but are not modeled in the code.

The principal effects of these oxidation-related phenomena can in many cases be simulated by suitable user choice of a limit strain and/or rupture strain in a parametric way. In some cases, particularly for cladding at low stresses in the high alpha-phase region (600–800°C), the code can calculate unreasonably large strains; this may be because the creep/plasticity database is derived from early measurements, which may not be representative of modern materials.

2.7.3. Technical Adequacy

While the overall level of detail in the deformation model is judged appropriate, there are important deficiencies concerning the modeling of the effects of oxidation on ballooning, and these should be remedied by the introduction of suitable first-order models. The user always needs to check the output from the ballooning model to see if the results are reasonable and to use parametric input, e.g., to limit clad strain, if this is not the case. The current model is therefore assigned to a zeroeth-order category.

2.8. Fuel-Rod, Internal-Gas Pressure

2.8.1. Phenomenological Package Description

The fuel-rod, internal-gas pressure is calculated as a function of temperature and gas volume using the ideal gas law. In calculating the fuel-void volume, the following volumes are included:

- fuel-rod plenum volume
- fuel-void volumes (cracks, voids, etc.)
- gap volume
- additional gap volume due to cladding ballooning

The gap and void gas temperatures are assumed to be at the average fuel-rod temperature in the i-th axial node, while the gas temperature in the fuel-rod plenum is assumed to be the coolant temperature of +6 K at the top of the rod, which is consistent with the FRAP-6T model. This last assumption is not justified.

The hot-void volumes are calculated either by interpolation of user-specified tables providing relative void volumes as functions of average fuel temperatures or are based on PWR and BWR-specific "correlations" to the FRAPCON-2 code calculations.

These correlations have been developed to account for the impact of fuel burnup and temperature on fuel-void volumes. The FRAPCON-2 calculations were performed for constant system pressures of 15.51×10^6 N/m^2 and 7.14×10^6 N/m^2 for PWRs and BWRs, respectively, over the following temperature and burn-up range:

PWR:	$500 \leq$ Temp. ≤ 1100 K
	$0.1 \leq$ Burnup ≤ 30 MWd/kg-U
BWR:	$500 \leq$ Temp. ≤ 1200 K
	$0.1 \leq$ Burnup ≤ 30 MWd/kg-U

Corrections to the calculated void volumes due to variations from standard BWR and PWR designs are made by assuming that variations in area are independent of inner cladding radius, and that it only depends on the temperature difference between hot and cold fuel rods. Furthermore, it is assumed that the relative fuel-void volume calculated from empirical correlations holds for varying as-fabricated geometries.

2.8.2. Qualitative Perspective

The documentation does not describe the procedure for calculating the number of moles of fission gas. Model parameters are greater for the initial coolant pressure than for the internal rod pressure. The impact of depressurization on the calculated correction factor is not described. Feedback between expansion of fuel and fuel porosity is not considered, and the correlations are only valid up to 1100–1200 K. It is not clear if vapor pressure curves for control-rod materials have also been developed. A justification for assuming a uniform rod temperature above 750 K is not provided.

2.8.3. Technical Adequacy

The present model as incorporated in the SCDAP/RELAP5 is at most zeroeth order. The assumption of setting the gas temperature in the fuel-rod plenum to that of the coolant temperature at the top of the rod (+6 K) is arbitrary and can be easily remedied. The approach of correlating void volumes to FRAPCON-2 calculations is not based on first principals; nevertheless, it is acceptable for the intended applications of SCDAP/RELAP5, especially when considering the larger phenomenological uncertainties inherent in the simulation of other more important issues relevant to severe accidents. The corrections to nonstandard designs is not unreasonable; however, no benchmarks are presented to confirm technical acceptability.

2.9. Fuel-Rod Liquefaction, Flow, and Solidification

2.9.1. Phenomenological Package Description

The fuel liquefaction, relocation, and solidification processes are calculated using models (Refs. 2-1, 2-2, and 2-3) for (1) fuel- and zirconium-cladding liquefaction, (2) cladding-oxide-shell failure, (3) relocation of liquefied Zr-U-O from the breached fuel element on the fuel-rod surface and subsequent solidification, and (4) reliquefaction of previously solidified material.

2.9.2. Qualitative Perspective

It is assumed that liquefied zirconium will instantly dissolve the UO_2 fuel that it contacts. The dissolution rate is calculated from an empirical relation due to Hofmann et al. (Ref. 2-4), which is based on steady-state uniform heating of ZrO_2-Zr-UO_2; however, the calculated amount of fuel dissolution is limited to that based on an equilibrium ternary-phase diagram. The effect of the heat of formation is not included. The limit on the maximum amount of liquefaction is set to the solidus temperature due to a better agreement with experimental data. The breach of the cladding-surface ZrO_2 layer will lead to a relocation of liquefied fuel and cladding over the fuel-rod outer surface.

The liquefied fuel-cladding mixture is assumed to spill out of the breached cladding and flow downward on the same (failed) fuel pin. The flowing mixture takes the form of a slug ring, with an initial thickness assumed to equal the average thickness of the in situ liquefied mixture. The spilling effects of the liquefied fuel-cladding mixture on the surrounding rods is not modeled.

The slug-ring velocity is calculated by the numerical integration of an equation of motion for the liquefied mixture. A steady-state, pipe-friction factor is used to calculate the liquid slug-ring drag as a function of flow-regime (turbulent and laminar) Reynolds numbers.

A slug-ring drop distance is determined from the calculated slug-ring velocity; however, in the event a grid spacer is encountered, the slug-ring velocity is arbitrarily set to zero. This arbitrary assumption has been found to dominate the predicted melt relocation behavior.

Heat transfer from the flowing slug ring to the cladding surface is modeled by convection on the basis of steady-state, heat-transfer coefficients for both laminar and turbulent flow. The heat-transfer coefficient is calculated based on the Reynolds analogy.

The rate of formation of a solid crust on the outside surface of cladding is calculated through a transient-energy equation that balances the heat loss to the cladding via convection and heat gained from the liquid mixture via a set of parametric equations to simulate molecular conduction and turbulence effects.

In the event that the slug-ring contact is much shorter than the temperature front propagation time into the cladding, the heat transfer from the crust of solidified Zr-U-O into the fuel-rod cladding is calculated assuming the cladding is a <u>semi-infinite</u> medium subject to a <u>uniform</u> temperature boundary. Otherwise, the cladding or cladding-oxide midplane temperature is calculated through a lumped-parameter, transient-energy equation, with conduction across the cladding and/or oxide layer. The lumped mass consists of the entire zirconium cladding mass in a given axial mesh plus one-half of the adjacent oxide layer.

The previously solidified crust is allowed to remelt upon reaching the melting temperature of the Zr-U-O mixture. The reliquefied crust is assumed to flow downward, due to gravity, until it reaches the closest axial mesh with a temperature below the melting temperature of the Zr-U-O mixture, where the film is assumed to refreeze.

2.9.3. Technical Adequacy

The present fuel liquefaction, relocation, and solidification models are not totally mechanistic and require a number of parametric inputs. They suffer from several modeling and phenomenological inadequacies, which are listed below.

1. The assumption of intimate contact between fuel and cladding material dictates eutectic dissolution as the only mode of fuel and clad failure.

2. The oxide-shell failure criteria are based on achieving either a user-specified failure temperature and cladding-oxide fraction or an oxide-shell melting temperature. Mechanistic models are not included to account for possible effects of internal pressure (especially for low-pressure accident sequences) and structural weakening of the fuel rods. The amount of relocation calculated can be very sensitive to the choice of oxide-shell breach temperature. No single value gives the best agreement in all experiments. These parametric fuel-rod failure criteria are expected to dominate the SCDAP/RELAP5 predictions of fuel failure conditions during severe accidents. This is probably the most important uncertainty associated with the early-interval melt progression and fission-product release, where the SCDAP/RELAP5 models are inadequate.

3. In a ballooned geometry, some of the melt may run down inside the balloon rather than form a crust on the outer surface. Because of capillary action, molten zirconium may also be sucked into cracks in the fuel pellets. Thus, the code will overpredict a blockage in such circumstances because it assumes all melt will freeze on the outside. Furthermore, the UO_2/zircaloy reaction will be inhibited, at least early on, by the enlarged pellet/clad gap and the internal oxidation of the cladding by steam. These effects are not treated. However, the effects of ballooning in CORA-15 (limiting the damage to the top half of the bundle and advancing the oxidation excursion) were well predicted.

4. There is no proper coupling between the models for the reactions between zircaloy/steam on the outside of the cladding and UO_2/zircaloy on the inside. A coupled model should be able to take into account dissolution of the oxide shell from the inside, oxygen availability at the outer cladding surface, etc.

5. The Zr-U-O mixture relocation is based on a gravity-driven mixture, slug-ring flow over the outside surface of the same failed fuel rod.

6. Because Zr-U-O slumping is treated on a node-by-node basis, strong nodalization dependencies are expected.

7. The heat-transfer models for slug-ring relocation are very deficient. Only conduction (steady state) from the flowing Zr-U-O mixture to the solid crust has been considered. Heat transfer by convection to the coolant (steam, and in the case of reflood, to water) and radiation to the coolant and the surroundings have been neglected. Large convection and radiation fluxes could potentially lead to rapid freezing of the moving Zr-U-O mixture.

8. Rivulet, rather than slug, flow (referred to as film flow in SCDAP/RELAP5 documents) has been established experimentally as the dominant relocation process (seen especially clearly in the CORA tests). Again, this process is not modeled.

9. Heat generation within the relocating material is neglected. Furthermore, additional zirconium oxidation during Zr-U-O relocation is not modeled. Surface renewal processes could potentially enhance metal oxidation and, thereby, the relocation process due to additional heat generation.

10. The effects of grid spacers on the slug-ring relocation is based on an arbitrary assumption. This has sometimes been found to dominate the SCDAP/RELAP5 predicted relocation.

11. Many of the stated assumptions and selected models are without adequate physical basis and are sometimes arbitrary. The choice of these models has sometimes forced more complications than are actually necessary.

12. The documentation of the models is inadequate because of inconsistencies, typographical errors, and a lack of clarity among various documents (Refs. 2-1, 2-2, and 2-3).

These models are more appropriate for parametric studies useful to PRAs than for application to mechanistic predictions and accident management studies.

2.10. Fuel-Fission-Product Release*

2.10.1. Phenomenological Package Description

Release Model for Intact Fuel. The fuel-fission-product release model is based on the PARAGRASS/FASTGRASS and CORSOR-M computer codes.

The release of volatile noble gases (xenon and krypton), cesium, iodine, and tellurium are based on the PARAGRASS model, while the release of semirefractory and refractory species is based on the CORSOR-M model.

The PARAGRASS model was developed for noble-gas release from fuel. In the version integrated into SCDAP/RELAP5, the release of volatiles like iodine, cesium, and tellurium is accomplished by a combination of diffusion through the fuel matrix and noble-gas bubble capture of individual atoms. The gas diffusion (to grain boundaries) is treated via the solution of time-dependent diffusion equations. The modeling of other processes (e.g., gas atom resolution, coalescence, trapping by gas bubbles) impacting fission-product gas behavior is accomplished through the solution of a second-order, time-dependent balance equation.

Following fission transport into the gap (as predicted by PARAGRASS), the release of fission products to the coolant is based on the following approach:

Noble gases are released instantaneously upon cladding failure. Additional releases of noble gases to the coolant are defined as the PARAGRASS-calculated, noble-gas releases into the gap region.

Cesium and iodine are released on the basis of a model that includes (1) a burst component that accompanies the initial cladding breach and depressurization, and (2) a diffusion component that describes the subsequent time-dependent releases of the remaining iodine and cesium species. These two components are assumed to be independent.

* It was brought to the Committee's attention recently that the NRC does not intend to use the SCDAP/RELAP5 code for best-estimate, source-term prediction. The VICTORIA code being developed under NRC sponsorship is intended to fulfill this role.

The model assumes that all of the iodine will react with available cesium to form CsI, with any leftover iodine being released as I_2 or any leftover cesium reacting with water to form CsOH and hydrogen.

The release of the fission-product tellurium appears to be identical to that of iodine and cesium; however, when zirconium cladding is not more than 90% oxidized, tellurium release is reduced to 1/40 of the calculated release (on the basis of ORNL experimental observations). Otherwise, the tellurium release is unimpeded.

The release of nonvolatile species is based on the first-order-rate equations of CORSOR-M with Arrhenius-type rate constants fitted to the fission-product-release data.

A simple mass-transfer-driven, first-order release model is used to calculate the release of tin. In addition, a simple empirical approach (a function of temperature only) is used to calculate the control-rod release of silver and cadmium.

Release Model During UO_2 Liquefaction and Fragmentation. The release of volatile fission products of xenon, krypton, cesium, and iodine during UO_2 liquefaction and fragmentation is calculated by assuming (1) an instantaneous release into the gap from all calculated liquefied fuel, and (2) an instantaneous release of calculated accumulated grain boundary material following a calculated fuel-fragmentation process.

Subsequent release within the rubble bed is stated to be controlled by the intragrain processes; however, it does not appear from the documents provided that any releases from the rubble bed are actually calculated.

The release of other less-volatile, fission-product species is assumed to be unaffected by UO_2 liquefaction and fragmentation processes.

The fuel-fission-product release model also calculates the enthalpy of released gases.

2.10.2. Qualitative Perspective

Intact Fuel. The present model for release of fission products before fuel liquefaction, fragmentation, and slump is based on an adaptation of a first-order model developed as part of the GRASS series of computer codes at Argonne National Laboratory

(ANL). The PARAGRASS models are only applicable to the prediction of volatile fission-product releases, whereas the CORSOR-M model is used to calculate the release of less-volatile species. The model version that has been integrated does not reflect the state of the art.

The burst release model, which is based on a relatively outdated empirical approach developed at ORNL for analysis of LOCAs for typical PWRs, is used as part of the present model. In addition, a longer-term diffusion release is included using an empirical first-order release model. The experiments on which these models are based used short sections of fuel rods with low-gap inventories. These models are used for all temperatures in SCDAP, while the adjustable parameters are based on experimental data in the temperature range of 970–1170 K. In general, these models have been found to represent the data on which they were based within a factor of three.

The CORSOR-M code parameters (pre-exponential and activation energy terms) are based on experimental data, except for refractory species, where the heat vaporization has been substituted for the so-called activation energy. In addition, the influence of changes in surface-to-volume ratios during meltdown are not included. The control-rod release is based on purely empirical relations.

In the absence of eutectic interactions, release of volatile fission products will be enhanced as result of continued fuel heatup. Furthermore, the present model does not appear to treat melt releases (both during melting and relocation). Therefore, these inadequacies become more consequential.

Release Model During UO_2 Liquefaction and Fragmentation. The liquefaction and fragmentation release model suffers from similar shortcomings. There are no models currently present in the code to calculate releases within a rubble bed configuration.

2.10.3. Technical Adequacy

The fuel-fission-product release models are for the most part untested, and they suffer from several inadequacies, including:

1. PARAGRASS models are only applicable for the prediction of volatile (noble gas) fission-product releases.

2. The version of the PARAGRASS code that has been integrated into SCDAP/RELAP5 does not reflect the state of the art.

3. The present burst release model is an outdated empirical approach. The longer-term diffusion release is also an empirical first-order release model. The experiments on which these models are based used short fuel-rod sections with low gap inventories and were limited to a narrow temperature range of 970–1170 K. In SCDAP/RELAP5, these correlations are assumed to remain valid for all temperatures.

4. Fission-product release during relocation is not modeled. The impact of changes in fuel-surface-to-volume ratios during meltdown and relocation is also an important consideration for fission-product release.

5. The current release model from control rods is purely empirical and nonmechanistic.

6. In the absence of eutectic interactions, release of volatiles will be enhanced (heatup and time at temperature). The present SCDAP/RELAP5 fission-product release model does not include any models for melt release (both during melting and relocation). In addition, models for fission-product release from rubble beds and core debris inside the lower plenum are not included.

Overall, the fission-product modeling approach is excessively unbalanced. The model to predict release from intact fuel is fairly detailed. It accounts for several important mechanisms of lattice migration and transport. On the other hand, the remaining models are for the most part highly empirical and outdated.

The documentation also leaves a lot to be desired. Many typographical errors exist in the equations, and most of the support information is not readily accessible.

2.11. Control-Rod And Core-Structure Models

2.11.1. Phenomenological Package Description

The code includes models for PWR control rods, BWR control blades, spacer grids, and flow shrouds. There is no model for burnable poison rods.

The PWR control-rod model considers heat conduction (one or two dimensional), oxidation of the zircaloy guide tube, melting and relocation of the Ag-In-Cd absorber, unoxidized zircaloy, and a stainless-steel liner. Relocation of all materials is deemed to occur at the melt temperature of the steel liner. The molten absorber material is assumed to freeze when it reaches an elevation with a temperature 200°C below its solidus temperature; there is no solution of energy/momentum equations.

The BWR control-blade structure is approximated by a cylindrical representation. Only radial heat conduction is modeled. The oxidation of stainless steel and boron carbide (B_4C) is considered. Relocation is treated using a model for incompressible viscous flow over a cylindrical geometry. This model is being replaced by newer models under development at ORNL (see Section 2.26.1).

The published version of the code contains only a simple model of spacer grids. These spacer grids are considered only as obstructions for melt flow. The grids are considered to melt instantaneously when their melt temperatures are reached. Inconel and zircaloy grids are treated by the code. A more-detailed model that considers the material interaction between Inconel and zircaloy has been developed at INEL (see Section 2.26.2).

The shroud model considers heat conduction in user-defined structures where dimensions and material composition are set by input data. User-specified material data may be employed, e.g., for thermal insulators used in experimental facilities. Oxidation of zircaloy is calculated, and relocation of unoxidized zircaloy above its melting point is considered.

2.11.2. Qualitative Perspective

The models outlined above, which are present in the published version of the code, treat the thermophysical behavior of the structures considered in a reasonable manner, but take no account of the various eutectic reactions, which can take place under severe-accident conditions and have been well quantified in separate effects tests. Examples of these phenomena are the reactions between zircaloy and the Ag-In-Cd absorber; zircaloy and the Inconel spacer-grid material; zircaloy and boron carbide; and stainless steel and boron carbide. These can all promote early melt relocation because the eutectics formed have low melting points, and first-order models should take these reactions into account. The boron-carbide reactions can change the chemistry of the system and thus alter the fission-product transport.

Model development, which addresses many of these deficiencies, is under way or should occur soon. The spacer-grid model has received recent attention; the more detailed treatment that was prepared accounts for the zircaloy/Inconel reaction and includes a calculation of the local temperature distribution. This model appears to be at an appropriate, first-order level but cannot be fully assessed at present in the absence of detailed comparisons with experimental data. Section 2.26 considers this new modeling, as well as additional developments in BWR-specific areas.

2.11.3. Technical Adequacy

The current models for control rods and spacer grids are considered zeroeth order; the principal deficiency is that well-known and quantified material interactions that affect the course of melt progression are not taken into account. However, model developments in progress now and/or in the near future promise to remedy the problem. The flow shroud model is considered to be fully adequate for its purpose.

2.12. Radiation Heat-Transfer Model

2.12.1. Phenomenological Package Description

The model analyzes the thermal radiation heat exchange between various components in the core, as well as between components and coolant. Coolant is considered to be capable of absorbing and emitting radiation, but scattering due to the presence of aerosols is neglected. The model calculates the net (leaving minus incident) radiation heat-transfer rate at a component surface and the net (emitted minus absorbed) radiation heat-energy rate by the coolant zone (finite volume). These rates are then used in thermal boundary conditions of any vessel component such as the fuel and control rods or flow shrouds.

Zonal Radiation Model. The analysis is based on the zonal approximation of radiation heat transfer. The major idealizations are that all surfaces are diffuse emitters of radiation, and reflection of radiation is partially diffuse and specular. The temperature, radiation characteristics, radiosity, irradiation, and net radiation heat flux are assumed to be uniform over each surface or a zone of a surface. Coolant absorbs and emits thermal radiation, but scattering by droplets and particles is neglected in comparison to absorption. Both the surfaces and coolant are assumed to be gray, nonselective emitters and absorbers of radiation. The basic model is well established in the literature and has been developed and used for nuclear reactor applications. The coolant separating the surface in the enclosure is assumed to be at a uniform temperature as far as the emission of radiation is concerned.

View Factors. An integral part of the radiation heat-exchange calculations involves evaluation of numerous view (angle or configuration) factors between various surfaces forming real or fictitious enclosures. The view factors of the intact geometry are calculated from analytical expressions for cylindrical rods of infinite length. These factors are defined by the geometry of the rods in a rod bundle and are independent of either the radiation characteristics or the surface temperature. Dynamic adjustment of the view factors due to the change in geometry following fuel-rod melting and material relocation is not attempted, but the disappearance of components for a highly degraded core is taken into account. An assumption is made in the calculations that if a shroud is present around a rod bundle, only the outermost rows and columns of rods are considered to exchange radiant energy with the shroud.

Absorptivities and Emissivities. The absorptivity of a gray surface is equal to the emissivity. The emissivities of surfaces are obtained from the MATPRO package (Ref. 2-5) and are computed as a function of temperature. Variation of emissivity of zircaloy with zirconium-oxide layer thickness is taken into account, but the dependence of the steel emissivity on the degree of oxidation or the difference in emissivity between wet and dry surfaces is not considered.

The emittance and absorptance of the gaseous coolant is obtained in a manner similar to that used by the Transient Reactor Analysis Code (TRAC)-BD1. In calculating the emittance of the coolant, only steam is considered, and the presence of a nonradiating gas such as hydrogen is ignored. The absorption spectrum of water vapor is considered to consist of six major bands, and the dependence of the spectral absorption coefficient on temperature is accounted for. However, the dependence on the spectral absorption coefficient on temperature is ignored. The integration over the spectrum to obtain a mean-wavelength-independent emittance or absorptance is carried out by ignoring the detailed band shape and using a "top-hat" (rectangular) band model to approximate the true shape of the band.

Mean Beam Length. Evaluation of the spectral emittance of a gas requires knowledge of the mean beam length or the mean path length through the gaseous medium separating any two surfaces confining the gas. This path length is calculated approximately. First, the path length between any two fuel rods is obtained, and the average path length between two component groups then is calculated by weighting with respect to the corresponding view factor. This is equivalent to assuming that coolant is not capable of absorbing or emitting thermal radiation. There is no discussion or reference to the published literature to justify use of such an approach to calculate the mean beam length.

2.12.2. Qualitative Perspective

The radiation heat-transfer model in SCDAP/RELAP5 represents a reasonable compromise between reality and computational effort in predicting a dominant phenomena. There is no discussion in the documentation of how the linkage between RELAP5 and SCDAP heat structures is accomplished in the code. Instead of coupling, it appears that RELAP structures were replaced by SCDAP components. In developing and implementing the model in the code, several assumptions have been made. Of these, the assumption that the coolant (gas) is a gray radiator and that the anisotropic reflection fraction of the surface does not depend on the level of oxidation of either zircaloy or steel is the most serious.

The transmittance of the gas is assumed to equal one minus the coolant emittance for water vapor that has some strong absorption bands. This and the gray gas assumption causes the net rate of radiative heat transfer to steam to be underestimated and the rate of heat transfer to the shroud wall to be overestimated. To avoid using a gray gas assumption, a much more realistic, yet computationally efficient, model could have been adopted.

2.12.3. Technical Adequacy

In general, the model represents a reasonable compromise between reality, the level of detail, and the computational effort. The implementation of the model suffers from a number of questionable assumptions and specifications of model parameters. For example, there is no justification provided in the manual as to why half of the radiation incident on a component is reflected backwards, half is reflected diffusely, and why this fraction does not depend on the degree of the material oxidation.

There is no evidence given in the documents that the diffuse-plus-specular reflection model for radiation exchange is superior over the diffuse reflection model. If the diffuse-plus-specular model is to be implemented correctly, an exchange factor needs to be computed from the knowledge of the radiation characteristics of surfaces and the fuel-rod geometry in the bundle, not specified arbitrarily. No model parameter sensitivity or benchmarking studies have been identified that would permit one to assess the adequacy of the model.

The view factors for the radiation heat-transfer model are computed for the intact core geometry, but after the core degrades or when the rods go to the debris, and the geometry changes, the view factors are not recomputed. This is a deficiency that becomes serious at elevated temperatures when the core becomes more open, and at the same time, the convective heat-transfer rate becomes relatively less important than radiation. Ring-to-ring radiation heat exchange and axial radiation heat transfer are missing from the model. The calculation of gas emittance is incorrect after zircaloy begins to oxidize, and hydrogen becomes a significant constituent of the coolant mixture.

In summary, thermal radiation is a dominant mode of heat transfer at elevated temperatures, particularly during the later stages of some postulated severe accidents, and, therefore, its correct modeling is very important.

2.13. Core-Region Debris Modeling

2.13.1. Phenomenological Package Description

The core-region debris model is one of the most important packages in the SCDAP/RELAP5 code. The model is based on a number of severe fuel-damage experiments and the TMI-2 accident. These experiments have shown that reactor-core damage proceeds in several stages before the core slumps to the lower head. The model describes the thermal response of three postulated core configurations during a severe accident, and the model is essentially parametric in nature. The three configurations postulated are described below.

1. Formation and Heatup of Nonporous Debris. The first stage of the core-damage progression is the degeneration of the reactor core into nonporous debris. This phase is caused by the melting of control rods and the metallic parts of fuel rods. The meltdown stage may begin as soon as a region of the core exceeds the temperature for eutectic melting of stainless-steel-clad control rods with an Ag-In-Cd absorber material and occurs at a temperature of ~1500 K. The meltdown becomes widespread when a reactor-core region exceeds the melting temperature (~2200 K) of the fuel-rod cladding. The submodel describes the heatup of the core and the subsequent slumping and solidification of the relocated metallic material.

The process is assumed to continue until the space between the fuel rods is completely filled and a nonporous debris bed is formed, which consists of a metallic cohesive debris with embedded, intact fuel rods. This nonporous debris layer supported by intact fuel rods may extend radially across the core, and its axial location, thickness, and temperature may vary with the radial position and time. The temperature distribution in the nonporous debris for each component group of fuel rods is calculated using a heat-balance integral method. Heat generation in the nonporous debris layer and convection and radiation heat transfer from the top and bottom surfaces are considered.

2. Formation and Heatup of Porous Debris. Two mechanisms are considered for porous debris formation. In the first, thermal shocking by reflood water of a reactor core embrittled by oxidation is assumed to form porous debris. The embrittled fuel rods are considered to break into particles during cooldown when the cladding temperature is decreased below the coolant saturation temperature plus a temperature increment that is a function of the rate of cooldown. This value ranges from 50–1273 K and is defined by the code user. The particle size and porosity of the debris resulting from fragmentation is assumed to be the same as that formed by thermal shock during the TMI-2 accident. The second mechanism of porous debris formation is assumed to be instigated by the melting of cladding with a very thin oxide layer and a small enough amount of dissolved oxygen such that the fuel will not melt. This mechanism of debris formation is assumed to occur when the metallic cladding temperature exceeds the melting temperature (~2000 K) and simultaneously when the cladding-oxide layer thickness is less than 0.01 mm.

Each time, a map is constructed of the debris regions resulting from degeneration of segments of fuel rods into the porous debris. In modeling, an arbitrary distribution of porous debris throughout the core is considered. The heatup of the debris region is calculated from a lumped mass energy balance. Volumetric heat generation is considered, but oxidation of particles is neglected. Convective heat transfer between the particles and fluid (water) in the debris voids is assumed to be the only mechanism for heat removal from the debris, and conduction heat transfer between the particles in the debris bed is neglected. The debris heat-transfer model considers three states: dryout, quenched, and quenching. In the dryout regime, steam and debris are considered to be in equilibrium, and in the quenched state, water and debris are taken to be in equilibrium. In the quenching regime, the quench-front speed is calculated from RELAP5.

3. Molten Pool. As the accident progresses, liquefied material is assumed to permeate into a colder region of the core and refreeze, or its movement is blocked by the nonporous debris formed during the metallic meltdown of the core. The uniform mixture of liquefied metallic melt and previously porous solid debris is assumed to be supported and contained by the nonporous debris layer. Heat in the melt pool is transferred from the interior to the exterior by natural convection. The pool is mapped into an idealized configuration (i.e., a hemisphere) and is used to calculate natural convection heat-transfer coefficients. The coefficients calculated from a published empirical correlation in this manner are then mapped back into the actual configuration of the molten pool.

The change in temperature of the molten pool during a timestep is calculated from an instantaneous overall energy balance. This balance accounts for volumetric heat generation, heat transfer to the nonporous crust that supports the pool, and heat transfer to the crust above the pool. In calculating the temperature distribution in the nonporous debris supporting the molten pool, the boundary condition reflects the melting of the top surface of the nonporous debris due to contact with the molten pool. The thickening and thinning of the crust above the pool is modeled using a heat-balance integral method. The approach is similar to modeling the bottom crust. The melting of the crust is coupled thermally to the molten pool and the thermal-hydraulic conditions in the core above the molten pool.

2.13.2. Qualitative Perspective

The phenomena of core-damage progression under severe-accident conditions are poorly understood; therefore, many submodels of the core-region debris model are not mechanistic, and the phenomena are treated parametrically. Some phenomena considered may only represent the code developer's view of an accident progression. A number of core-region debris modeling deficiencies have been identified by the code developers and need not be repeated, but others have not. For example, no consideration is given to heat transfer between molten metal drops and fuel rods in the cooler core region or of time taken by the drops to move and solidify into a nonporous debris layer.

In some cases, no justification is provided for neglecting a transport process. For example, conduction and radiation heat transfer are neglected in a porous debris bed in comparison to convection, but no supporting evidence is provided that this is justifiable at high temperatures when the bed is dry. Also, the assumption is made that the liquid in the melt pool is perfectly mixed. The segregation of components and thermal stratification could greatly impact natural convection in the pool, as well as heat transfer to and failure of the crust.

The documentation also leaves something to be desired. There is no indication (Ref. 2-3) of how the heat-transfer coefficients at the top and bottom surfaces of the nonporous debris [Eqs. (3-228) and (3-229)] and the heat-transfer coefficient at the surface of the debris particles [Eq. (3-246)] are being calculated. There is also no discussion on how the emissivity of the nonporous debris, which may be an alloy of several relocated metals, is determined.

2.13.3. Technical Adequacy

The main technical findings related to core-region debris modeling can be summarized as:

1. Model assumptions are not fully articulated in the documentation, and many important idealizations are not justified;

2. The lack of models for debris fragmentation represents significant limitations of modeling severe-accident phenomena;

3. The lack of a transient natural circulation model in the melt pool limits the capability to predict sudden crust failure and spillage of a large mass of the molten debris to the lower head;

4. Many submodels are incomplete and as such are not capable of describing the physical processes in the core;

5. Benchmarking, validation, and sensitivity studies are very limited or do not exist; and

6. Parametric models are deficient in time resolution and dynamics.

In many respects, the core-debris model is not adequate. If exercised by a knowledgeable user, the model may be suitable for parametric studies, but the model is judged to be inadequate for analysis of recovery scenarios as part of accident management work.

2.14. Core-Slumping Models

2.14.1. Phenomenological Package Description

The slumping of the material from the molten pool in the core to the lower head of the reactor vessel is described by mechanisms that are considered to trigger the slumping. These mechanisms are: (1) the meltthrough and failure of the nonporous debris that supports the molten pool, and (2) the meltthrough and failure of the upper crust and the resulting displacement of liquid from the molten pool by the solid material that falls into the molten pool. The criteria for slumping from the molten pool are parametric in nature, and the thickness of the crust is used to determine the time of failure of the crusts that surround the molten pool.

The crust supporting the molten pool is considered to fail when its thickness is less than the user-specified value of 25 mm. After such a failure, all liquid in the molten pool above the point of failure is assumed to drain to the lower head. The crust on top of the molten pool that supports solid debris is considered to fail when its thickness becomes less than 5 mm. If some liquid has drained from the molten pool due to failure of the lower crust, then the upper crust is considered to fail when its thickness becomes less than 25 mm.

The crust failure criteria are based on calculations of stresses under conditions estimated for the TMI-2 accident, and there are large uncertainties associated with these calculations. Large uncertainties in configuration, composition, load, and temperature during the evolution of a severe accident require that the failure criteria be treated by the user as a sensitivity parameter.

2.14.2. Qualitative Perspective

Because the slumping models are parametric, it is only appropriate that the failure criteria, rate of liquid release or spillage from the molten pool, and degree of interaction of the slumping material with the water through which it falls be defined by the code user. Parametric calculations employing different user-specified input parameters do not appear to have been performed for severe nuclear reactor accident scenarios because no documents reporting results of such sensitivity calculations have been identified.

2.14.3. Technical Adequacy

The slumping models are not mechanistic but parametric. The model is deficient in that it does not account for the interaction of the slumping material with water (i.e., breakup and steam formation), fuel and control rods, lower support plate, and core baffle plate (i.e., refreezing of the melt on structures or possibly melting structures). There is also a possibility that the slumping melt from the molten pool may form flow passages in the core and support structures as it moves downward by the action of gravity. Creep-rupture failure of the lower crust supporting the molten pool is also not modeled.

No details are provided in the documentation of how the crust failure criteria were established. It is unclear if the criteria are based on some simple zero-order models or if they are educated guesses consistent with the TMI-2 accident. The models are not validated and, in the absence of confirmation, may only represent code-developers' views of core slumping during a severe LWR accident progression.

2.15. Lower-Plenum Debris Heatup

2.15.1. Phenomenological Package Description

The model calculates the heatup of the debris that slumps into the lower plenum and the heatup of the vessel wall in the axial and radial directions. The most important output of this model is the calculation of the vessel wall temperature from which the time for creep-rupture failure, or melting of the lower head, can be estimated. The model accounts for the time-dependent debris bed and considers spatially varying initial internal energy, decay heat, porosity, particle size, and effective (thermal and radiative) conductivity of the porous material. Debris dryout and quench are based on the Lipinski model.

A claim is made in the documentation (Ref. 2-3, p. 3-189) that melting and freezing of the debris has been accounted for, but the description is so terse that it is not clear how this was done. The heatup of the lower head is coupled with core-slumping and system thermal-hydraulic models. The fallen debris segregates in some unspecified way into a nonporous metallic layer in contact with the vessel wall. Above this denser debris, there is a porous ceramic debris layer, and this debris is overlaid with water. There are several submodels of the lower-head heatup model, which are highlighted below.

Conduction-Advection (COUPLE) Model. COUPLE is a two-dimensional, finite-element, steady-state, and transient heat-conduction and advection code that is available (Ref. 2-6) for both planar and axisymmetric geometries. The general code was developed to handle anisotropic heat conduction and advection; however, according to the SCDAP/RELAP5 code manual (Ref. 2-3, pp. 3-186 to 3-187), only the pure heat-conduction and isotropic effective conductivity version has been adopted.

It is fortunate that the code developers did not use the more general version of the COUPLE model in SCDAP/RELAP5 because the velocity is computed from the conservation of mass equation only for an inviscid, incompressible fluid by ignoring the presence of particles in the bed. For correct prediction of the flow field in the particulate the Darcy (or a modified form of the Darcy) law would also have to be used. The energy equation of the porous bed is based on the assumption that the solid is at the same temperature as the coolant (e.g., the thermal equilibrium model) and that the continuum Fourier-Biot heat-conduction law is appropriate for an inhomogeneous and anisotropic particle bed.

Porosity Model. The local porosity of the debris bed is not calculated but is specified by the input. In this manner, the porosity can be specified by the user as functions of position and time. The local thermal characteristics of the porous bed are calculated by a simple volume averaging based on the local volume-fraction-weighted solid and fluid properties. This type of averaging is appropriate for the density and specific heat but not for the effective conductivity.

Thermal Conductivity Model. The total effective thermal conductivity of a dry porous bed is represented as a sum of conductive and radiative contributions. The two components are calculated using Imura-Takegoshi and Vortmeyer models, respectively. The severe limitation of the model is that they have been developed for homogeneous, uniform-diameter, spherical particle beds and not for inhomogeneous, irregularly shaped, nonuniform-diameter, particle beds.

Phase Change. Phase transformation of a material is computed using the enthalpy method. This is an accepted procedure and requires the assumption that instead of a discrete temperature (i.e., for a pure substance), the phase change occurs over a small temperature difference. The results are found to be insensitive to the difference chosen for the calculations.

Dryout of Debris. The quenching or heatup of a dry debris bed is handled by the COUPLE model. Debris quench is determined using the Lipinski correlation, which calculates dryout as a function of relevant debris parameters and coolant (water) properties. If debris is in a state of dryout, the COUPLE model calculates debris heatup considering that the voids in the debris bed are filled with steam. Heat transfer from the debris to coolant and the vapor generation rate in the control volumes of the lower plenum are handled by RELAP5.

2.15.2. Qualitative Perspective

There are several inherent limitations of the COUPLE model used to calculate flow and heat transfer in debris beds, which include the following: (1) an inadequate model of heat transfer between the solid and fluid in the porous bed, (2) the lack of a model to predict multidimensional and heat-transfer flow in the porous bed, (3) the lack of a model to describe the rise of vapor or migration of liquid in the debris bed, and (4) an inadequate description of total effective conductivity of a porous bed comprised of irregularly shaped, nonuniform-diameter particles. These limitations and other issues have a potential of impacting the predictions of debris heatup in the lower plenum.

Documentation is not adequate for the empirical correlations used to calculate the convective heat-transfer coefficient of the liquefied-pool bottom surface and other structural material surfaces in contact with the liquefied debris pool. Documentation of these empirical correlations, as well as the model's emissivity factor, could not be found in either the

SCDAP/RELAP5 code manual (Ref. 2-3), the COUPLE documentation (Ref. 2-6), or the citation made in Ref. 2-6 to the published literature. It is not possible to determine if these correlations are adequate or have been implemented correctly in the code if they are not available for assessment.

The change with time of the debris bed in each control element impacts the heatup of the debris through the composition variation with time, and it affects the thermophysical properties, as well as the fusion temperature of the bed. However, this dependence has not been addressed in the model. If the melting of debris is accounted for in the model, as claimed in the code manual (Ref. 2-3), then it appears that the molten material is stagnant and cannot migrate due to gravity.

2.15.3. Technical Adequacy

Our current state of knowledge with respect to key phenomenological issues related to core-debris quench and heatup in the lower plenum is not complete. The main technical findings related to the model can be summarized as:

1. Benchmarking and validation of the submodels and model against the test data under severe LWR accident conditions are lacking;

2. Sensitivity studies with the intent to assess the performance of the model for "reasonableness" have not been performed;

3. The parametric-variable, element-porosity model is deficient in time resolution and dynamics. The model is not described, and its physical basis is uncertain. For example, how does the porosity vary with time as the debris bed heats up and melts if migration of the liquefied debris within the solid is not modeled?; and

4. No guidance is provided to the user in the available documentation of how to select model parameters for default values. For example, how is the density of specific heat of the debris mixtures (metals and oxides) updated with time in each bed control volume? The relative amounts of metals and oxides with each control volume change during calculation, however, as the lower-melting-temperature metals tend to relocate downward within the bed. This

movement increases the metal-to-oxide ratio in the lower part of the bed. These changes should be reflected as adjustments to the local mixture densities, specific heats, and solidification (fusion) temperatures. This composition dependence of the nonporous metallic debris in contact with the vessel would impact the heatup of the structure.

The treatment is inadequate because quenching, dryout, material convection in the porous debris bed, and natural convection in the liquefied debris has not been treated in the model.

2.16. Structural Creep-Rupture Model

2.16.1. Phenomenological Package Description

A two-dimensional temperature field in the structures or vessel wall is obtained by using the COUPLE code. The principal stresses for a given system pressure are calculated by using a thin-shell approach (uniform tangential stress and no shear stress). The equivalent stress is calculated using the Distortion Energy Theory. The equivalent stress reduces the two- or three-dimensional principal stress state into a one-dimensional stress state. The equivalent stress is related to the temperature and creep-rupture time through the Larson-Miller parameter or through the Manson-Hafred theory. The Larson-Miller parameter is used for 316 Stainless Steel and Inconel 600. However, the Manson-Hafred theory is used in the lower range of stresses in A-508 Class 2 Carbon Steel, whereas the Larson-Miller theory is used at the higher range of stresses.

To allow for variations in temperature with time, a creep-damage parameter is introduced. The damage that occurs during the time a structure remains at a given temperature is obtained by dividing this time by the rupture time obtained from the Larson-Miller parameter. These damage fractions are added, and when the sum reaches a value of unity, the structure is assumed to fail by creep rupture.

2.16.2. Qualitative Perspective

The model is based on a thin shell approach and utilizes a creep-rupture database obtained under isothermal conditions. In several severe-accident scenarios, the temperature in the vessel wall will be nonuniform. The effect of the presence of openings in the structures on creep-rupture times is also not considered.

2.16.3. Technical Adequacy

The approach is correct. However, concern exists with respect to the application of the creep-rupture criterion when structure wall temperature is nonuniform and openings exist in the structures. Thin-shell theory may not be valid in all structure types and situations. For cases in which the theory is not valid, it is recommended that a detailed finite-element analysis be performed instead using codes such as ABAQUS and PATRAN.

2.17. Aerosol Agglomeration Models

The aerosol and fission-product behavior models were implemented to resolve the source-term effects during an accident.

2.17.1. Phenomenological Package Description

There are two sections to the agglomeration model: (1) a description of the agglomeration mechanisms, and (2) the calculation of the mass distribution due to agglomeration, deposition, and other sources. The mass-distribution calculation requires input from the deposition, evaporation/condensation, and the heterogeneous chemistry models. The results from these models and the agglomeration model determine the mass distribution in a given cell at a given time. Given the new mass distribution, a mapping of the fission products within the vessel and reactor coolant system can be obtained. The mapping provides the input for both the next mass-distribution calculation and for the source-term results. The agglomeration calculation requires input from the other SCDAP/RELAP5 sections, including the materials properties, thermal hydraulic, and release models.

2.17.2. Qualitative Perspective

The aerosol agglomeration and mass-distribution calculation were originally based on the TRAP-MELT code. The models were linked to the SCDAP/RELAP5 MATPRO properties routines, and a time-temperature-dependent agglomeration kernel was introduced. The mass-distribution calculation was reprogrammed to improve computational speed and include the time-dependent agglomeration kernel. The agglomeration kernel combines the three processes expected to occur in the reactor coolant system during a reactor accident scenario because of (1) Brownian motion, (2) gravitational motion, and (3) turbulence.

2.17.3. Technical Adequacy

The agglomeration kernel provides models for the expected rate terms. However, the models have not been upgraded to use the more widely based correlations. Furthermore, the models have not taken advantage of the RELAP5 calculation for flow field parameters to vary the coefficients from laminar through turbulent regimes. The aerosol mass-distribution calculation uses a brute-force numerical technique when a more sophisticated technique should have been attempted. The replacement technique was not compared with the original TRAP-MELT technique nor did it reduce computation time sufficiently. Because this technique was not pursued further, tests of the radiological source-term models have not been published.

2.18. Aerosol Particle Deposition

2.18.1. Phenomenological Package Description

The aerosol particle deposition package provides models for the reduction in aerosol mass due to deposition in both the SCDAP core regions and on RELAP5 structures. The reduction in mass for each aerosol size bin in each cell is then passed to the mass-distribution calculation. The package requires input from other SCDAP/RELAP5 sections, including materials properties, thermal hydraulics, and agglomeration.

2.18.2. Qualitative Perspective

The aerosol deposition calculation was originally based on the TRAP-MELT code. The models were linked to the SCDAP/RELAP5 MATPRO properties routines. The deposition model includes five types of deposition processes: (1) gravitational settling, (2) thermophoresis, (3) turbulent deposition, (4) laminar flow deposition, and (5) deposition in bends. The processes are combined additively, as is typical of many aerosol deposition codes.

2.18.3. Technical Adequacy

The deposition package provides models for many of the expected phenomena. However, the models have not been upgraded to more widely based correlations, have not been compared with available data, and do not use the RELAP5 flow calculation to determine flow-dependent coefficients. In addition, models for resuspension, intervolume settling, and diffusiophoresis are not included. The models are also not adequately described in the manual and may have been implemented incorrectly. Because the mass-distribution calculation was not pursued, these models have not been adequately tested.

2.19. Vapor Condensation/Evaporation

2.19.1. Phenomenological Package Description

This package combines both vapor condensation on aerosols and walls and the equilibrium vapor mass concentration calculation. The former phenomena accounts for the distribution of the condensed phase between aerosols and walls. The latter phenomena is the chemistry calculation. It allows seven species to exist (I_2, CsI, CsOH, Te, Ag, Cd, and Sn).

2.19.2. Qualitative Perspective

Both the condensation/evaporation and chemistry calculation are based upon the TRAP-MELT code. Both models were linked to the SCDAP/RELAP5 MATPRO properties routine but were kept close to the original TRAP-MELT coding.

2.19.3. Technical Adequacy

The condensation/evaporation model has a reasonable formulation, although the availability of data for the mass-transfer coefficients is a concern. The output of this model is an input to the mass-distribution calculation, which was found unacceptable. This model has not been used within the SCDAP/RELAP5 framework.

The chemical speciation model is limited to those situations where the seven species are dominant in the system. Scenarios where other fission products would be available (e.g., barium or ruthenium) or other species could be formed (e.g., HI or AgI) could not be calculated with this model; thus, the model is inadequate.

2.20. Heterogeneous Chemical Reactions

2.20.1. Phenomenological Package Description

This package models the reactions between fission-product vapors and the surfaces they flow across. This model removes vapor from the bulk gas directly, as opposed to either condensation or aerosol deposition, and is therefore not related to the mass-distribution calculation directly.

2.20.2. Qualitative Perspective

The heterogeneous reaction model is based upon the TRAP-MELT code, which in turn is based upon a limited set of vapor surface reaction experiments. The model uses an assumption of a constant vapor deposition velocity (e.g., Te_2 is deposited at a rate of 1 cm/s) that is unaffected by the other thermal-hydraulic phenomena occurring within the cell.

2.20.3. Technical Adequacy

The literature indicates that the physics in question is more complex than this model assumes. An exact model may be out of the scope of the SCDAP/RELAP5 code; however, some account should be taken of the other thermal-hydraulic parameters in the cell. In particular, the temperature in the cell has been shown to have a major effect. A more elaborate model is necessary; this includes deabsorption when the thermal-hydraulic conditions change to account for variation velocities. A review of the experimental data is also warranted.

2.21. Materials Properties

2.21.1. Phenomenological Package Description

The SCDAP/RELAP5 properties packages supply materials properties data as a function of temperature, pressure, and composition for the SCDAP and TRAP-MELT parts of the code. The MATPRO library contains the values for solids, liquids, and noncondensable gases found in the vessel and reactor coolant system. The TRAP-MELT part of SCDAP/RELAP5 contains values for the condensable species.

2.21.2. Qualitative Perspective

The MATPRO library has been available for several years. Its latest issue is from 1989, but the property data used in the correlations are from several years before that date. The correlations used are well documented and are well compared with the data. The manuals contain error estimates for most properties. The major functions are: melt temperature, specific heat, enthalpy, thermal conductivity, thermal strain, and density. Some materials have additional functions for Young's modulus, Poisson's ratio, viscosity, and vapor pressure. The major solid/liquid materials detailed are: uranium dioxide, uranium metal, zircaloy, zircaloy oxides, stainless steel and its oxides, control-rod materials and poisons, and Zr-U-O mixtures. The noncondensable gases are: helium, argon, krypton, xenon, hydrogen, nitrogen, oxygen, carbon monoxide, and carbon dioxide. The gas functions are: density, specific heat, thermal conductivity, emissivity, and viscosity (I_2, CsI, CsOH, Te, Ag, Cd, and Sn).

The TRAP-MELT property functions are the diffusivity of a vapor in steam and the mass concentration of the seven species used in the chemistry calculation.

2.21.3. Technical Adequacy

The MATPRO library is extensive, and the correlations are detailed. Only three concerns were expressed: (1) the lack of data for eutectics other than the Zr-U-O mixture, (2) the lack of data for grid spacers, and (3) the level of detail of some of the correlations, which may cause excessive computer usage. The TRAP-MELT/MATPRO gas property functions reflect the limited chemistry allowed in the code system. An expanded list will be required to calculate source-term effects.

2.22. Fission-Product Decay Heat

2.22.1. Phenomenological Package Description

The model accounts for the reduction in fission-product decay heat resulting from the loss of volatile elements after a major disruptive event. If it is determined that fuel failure has occurred, FDECAY is called by the nuclear-heat-generation routine NHEAT to return a correction factor by which the decay-heat rate, predicted by the methods and data described in the ANS-5.1.-1979 standard, is to be multiplied. The correction factor represents the fraction of the total fission-product, decay-heat source term remaining with the fuel following a loss of volatile elements.

The decay-power fraction curves used in the FDECAY model were developed from the results of isotopic summation calculations using the ORIGEN2 code. Fission-product, decay-heat calculations were performed for four representative fission-product inventories, which included: (1) a generic PWR core (33,800 MWd/MTU), (2) TMI-2 just before the March 1979 accident, (3) a nominal severe fuel-damage (SFD) test conducted in the Power Burst Facility (PBF) at INEL, and (4) nominal power burst conducted at the PBF. Calculations were then performed in which the inventories were allowed to decay with and without the volatile element removal modeled.

The decay-heat power fraction is determined by taking the ratio of decay power with volatile release to the decay power without volatile release using the method suggested by Schnitzler. The model distributes the decay heat into cesium, iodine, tellurium, and beta-gamma contributions. The decay-heat contributions resulting from the loss of the remaining lower volatility species (i.e., tellurium, barium, strontium, etc.) are lumped into a single composite group and are neglected, even though they may be appreciable at some periods following release. The decay heat is then converted into heat source contributions by relative population and attenuation models.

2.22.2. Qualitative Perspective

The four fission-product inventories considered are representative of a wide range of possible fuel exposures, and the three volatile element release scenarios considered represent expert-opinion estimates of volatile release in quench scenarios only. The cases cover a wide range of conditions likely to be encountered by a SCDAP/RELAP5 user examining INEL experiments on TMI-2. But if the considered scenarios are felt to be inadequate for the problem being analyzed (i.e., station blackouts), the user may provide problem-specific data. The SCDAP user selects by an input flag from among the available data sets.

The accuracy of the NHEAT code predictions may largely depend on the applicability of this input to the particular problem. However, unless reactor parameters such as power density, burnup, core composition, and previous power history are close to one of the four representative inventories analyzed by INEL, it would be prudent to repeat the fission-product, decay-power fraction calculations for the particular reactor of interest.

To minimize the number of species being tracked, only tellurium, iodine, and cesium are considered. The decay-heat contribution from these three volatile elements is ~24% at 130 s but is ~70% at 1080 s. Energy conservation in the model is assured, but there may be disparity of where the volatile species are deposited. Because other volatile species (e.g., krypton, xenon, and others) are not being tracked, the normalized fractional decay-heat contributions from tellurium, iodine, cesium, and the decay-heat model may be inaccurate.

2.22.3. Technical Adequacy

The model is physically reasonable, and model implementation is correct. The model captures major trends. Unfortunately, the documentation does not provide a great deal of information in this area. The model does not appear to have been validated by comparing model predictions against more detailed computations, but code acceptance testing was conducted to check the code. The decay-heat contributions from elements other than tellurium, iodine, and cesium are appreciable at some periods following release. The fractional contributions from these elements can be isolated for tracking with minimum effort.

It appears that FDECAY can predict acceptable decay-power fractions for each of the inventories considered. However, the available data cannot be applied with confidence to accident scenarios where the fuel failure mode, and as a consequence, the resulting volatile element releases, differ substantially from the fuel heatup and quench failure mode assumed in the analysis.

2.23. Decay-Heat, Energy-Deposition Model

2.23.1. Phenomenological Package Description

The energy-deposition model is used in the SCDAP code to determine the location in which the released energy is deposited in a given cell. The decay energy from the fission products carried in the vapor space as an aerosol or deposited on a structural surface is distributed according to the local group population in a given cell; however, there is no reference in the document of how and where in the code the population is being computed. When a decay occurs, the released energy can be assigned to either the vapor space, the solid that is first impacted by the decay particle, or the solid upon which the product is deposited; however, there is no discussion in the document of how these probabilities are being computed in the code. The submodels included in this model are:

Gamma-Energy Release from Airborne Fission Products. The energy released is treated as a volumetric contribution in the solid heat structures connected to the control volume. The distribution of the gamma energy deposited within the solid structure is calculated on the basis of an average, predetermined attenuation factor corresponding to the particular heat structure, but no information is provided nor are relevant literature sources cited as to how the attenuation factor is being specified.

Gamma-Energy Release from Deposited Fission Products. The energy released is treated in a manner similar to that released from airborne fission products. Half of the energy released from the fission products on the surface of the heat structure is assumed to enter the structure, and half would be released toward the adjacent vapor space. By assumption, this half is not attenuated by the vapor and is distributed among heat structures bounding the control volume.

If a control volume has only one associated heat structure, all gamma energy emitted by the deposited fission products is assigned to the emitting structure and is attenuated within the structure. If the control volume has more than one heat structure, reabsorption of energy by the emitting surface is considered to be as likely as reabsorption by the emitting surface because the view factors are based on the relative surface area, which includes the area of the emitting surface.

Beta-Energy Deposition from Airborne Fission Products. The beta-energy release is similar to gamma energy and is distributed to the various heat structures according to the relative area of each structure. Attenuation within vapor space is according to the Katz and Pendolf model, which considers the presence of steam only. The calculated beta range is compared to a characteristic distance to the solid surface, determined by calculating an equivalent radius for airborne release. The energy absorbed by the steam in the vapor space is taken as a volumetric energy source for the control volume, and the remaining energy is apportioned among the various heat structures according to the view factor as an incident energy flux at the surface.

Beta-Energy Release from Deposited Fission Products. Beta-energy release from deposited fission products is treated similarly to that for gamma release. Half of the energy is assumed to be incident on the heat structure and the remaining half on the vapor space. The only difference is that some fraction of the beta energy is absorbed in traversing the vapor space. An attenuation calculation is performed to determine the fraction of the beta energy reaching the other heat structures bounding the control volume. This calculation is again based on the equivalent distance to the bounding heat structures and on a characteristic endpoint energy for each group. The attenuated energy calculated in this manner is assigned to the control volume as a volumetric heat source.

2.23.2. Qualitative Perspective

The treatment of the model in the code document (Ref. 2-3) is very terse, and little quantitative information or relevant literature sources are provided. The focus is on the model description and statement of the assumptions, and little or no justification of the assumptions is provided. Perhaps the code developers did not consider the energy deposition contributing significantly to the overall decay-heat deposition in a given cell. No references are cited where relevant information can be found. A few specific items are mentioned here.

No discussion is included in the document as to how the "predetermined attenuation factor corresponding to the particular structure" is to be evaluated for calculating the gamma-energy deposition from airborne fission products. What criteria are used to select each fission-product group, and how many such groups are used for beta-energy-deposition calculations from the airborne fission products? No theoretical justification is provided as to how to determine the equivalent radius or equivalent diameter needed to calculate the absorption of beta rays by steam to calculate beta-energy deposition from airborne and deposited fission products, respectively.

It is stated in the code manual (Ref. 2-3, p. 6-3) that the characteristic distance to the solid surfaces is determined by calculating an equivalent radius for a right circular cylinder with the same volume and surface area as the control volume. However, from the defining equation (Ref. 2-3, Eq. 6-2), it is not clear if the area includes the ends of the cylinder or only the lateral surface. The absorption coefficient of steam for beta rays is different than that in a steam-hydrogen mixture, but there is no justification provided as to why an absorption coefficient of steam is used for both cases.

2.23.3. Technical Adequacy

Our current state of knowledge with respect to energy deposition in the coolant and the structures connected to the volume due to fission-product transport and decay is much better than the simplified treatment used in the model would suggest. The approach may be adequate for the purposes intended in SCDAP/RELAP5 because the rate of energy deposition may be relatively small compared to the heat-transfer rates to or from a given cell. However, the relative importance of the different contributions does not appear to have been assessed and is not reported in the documentation.

The general approach taken in the energy-deposition model seems to be adequate, and improvements would be warranted only if more detailed computations indicated otherwise. Such calculations do not appear to have been performed as the model has not been assessed, and validation is lacking. The documentation in the code manual (Ref. 2-3) is inadequate and needs attention. Another volume of the code manual similar to Ref. 2-3 may contain additional information necessary for code application and input data preparation when it is published.

2.24. Decay-Heat, Gamma-Attenuation, Complete-Absorption Model

2.24.1. Phenomenological Package Description

The local absorption of gamma energy for radioactive decays is evaluated in the model by assuming both a characteristic gamma energy and materials properties of the heat structures. The local volumetric heat generation is simplified by assuming exponential decay with distance from the face of the structure. This is inconsistent with the assumption of isotropic gamma-energy release from fission products on the surface of a heat structure (Ref. 2-3, p. 6-2). The energy-deposition rate in a slab of material (between x_1 and x_2) is obtained by integration of the local rate. Gamma-energy deposition in cylindrical heat structures is treated in the same manner as slab geometry structures because the difference in the deposition rates is not considered to be sufficiently significant to warrant detailed calculations for the two geometries. However, no quantitative information to support this conclusion for materials having different attenuation coefficients is provided in the documentation.

If gamma energy is not fully attenuated in the cell containing the initial radioactive decay, the partial absorption model is not implemented. Such a model is feasible and could be developed, but the model was judged by the code developers to be unwarranted at this time. However, arguments against inclusion of a partial absorption model were not discussed, but there appears to be difficulty in developing logic and in how to track partial absorption, escape, and reabsorption of the gamma radiation by the structure(s) in a cell.

2.24.2. Qualitative Perspective

The volumetric energy generation rate due to gamma attenuation is computed on the basis that the linear or mass attenuation coefficient is independent of the gamma ray energy. This is accomplished in the model by assigning some "characteristic gamma energy" and "materials properties of the structure." However, there is no discussion of how these model input parameters are to be determined to insure that the local generation rate computed on a detailed energy basis and employing mean characteristics are the same. Because the gamma-attenuation coefficient of a structural material varies with the energy, without appropriate weighting over the energy spectrum there is no assurance that Eq. (6-4) (Ref. 2-3, p. 6-5) would predict correctly the volumetric energy generation rate due to gamma attenuation. This is an important issue and appears to have been overlooked by the code developers. An undefined "... appropriate attenuation constant for the structure" (Ref. 2-3) may not yield a correct local volumetric energy generation rate.

It is well established in the literature that the local volumetric energy generation rate due to an isotropic plane gamma source obeys an integroexponential and not an exponential law, and no justification is provided in the documentation for using the approximate exponential law. The expression will underpredict local gamma attenuation. At least some mean direction should be estimated for the gamma beam and assigned to improve the predictions.

2.24.3. Technical Adequacy

The state of knowledge of the volumetric energy-deposition rate due to gamma attenuation in heat structures is much better than that suggested by the simplified treatment in the model. However, the modeling approach may be completely adequate for SCDAP/RELAP5 purposes because of the large uncertainties in predicting fission-product deposition rates on the heat structures after a severe, core-disruptive accident. There is a lack of justification for the model chosen and the prescription of the important model input parameters. Also, it appears that the model has not been validated against benchmark predictions or test data.

Documentation needs to be improved. Specific guidance to the user is essential. For example, how are the characteristic volumetric generation rate at the surface of a structure and the gamma-attenuation coefficient to be prescribed? The ultimate judgment for the determination of the adequacy of the gamma-attenuation, complete-absorption model requires comparison with either detailed predictive models and/or separate effects tests.

2.25. Severe-Accident Thermal Hydraulics

2.25.1. Phenomenological Package Description

Most of the heat-transfer and friction models are from the standard RELAP5 package. RELAP5 has undergone significant assessment and review, and the strengths and weaknesses of the modeling are generally known. The specific models in RELAP5 were not reviewed by the Peer Review Committee. This represents a significant limitation of the present review because the friction and heat-transfer models determine the flow and heat-transfer boundary condition on structures.

When the RELAP5 models are used, SCDAP passes hydraulic diameter information to RELAP5, and the standard RELAP5 correlations are used. A key assumption is that by using a properly defined hydraulic diameter, the models will produce reasonable results. The

RELAP5 correlations have not been modified specifically to address severe-accident conditions. The correlations and the interphase exchange modeling are used in the solution of the mass, energy, and momentum equations. The RELAP5 models are used until a cell is blocked. From that point on, thermal-hydraulic calculations are not performed for that cell, even if the blockage melts and relocates.

In some cases, SCDAP has its own models for heat transfer. The wall friction model is modified (Refs. 2-7 and 2-8) when a debris bed is present. Currently, the only modification to the wall friction model is changing the hydraulic diameter from a value corresponding to a rod-like diameter to a value corresponding to porous debris. The RELAP5 wall friction model is then used to determine the liquid and vapor wall friction coefficients needed in the momentum equations. The hydraulic diameter (d_h) corresponding to porous debris is calculated from

$$d_h = 4 \text{ (bed fluid volume) / (surface area of particles).}$$

The porosity and particle diameter of the debris resulting from fragmentation is assumed to be the same as that formed by thermal shock during the TMI-2 accident. Analysis (Refs. 2-9 and 2-10) of this debris determined that it had an average porosity of 0.54 and an average particle diameter of 0.87 mm. There is no rationale for determining average particle diameter and porosity in the presence of shards and other nonspherical particles. The characteristics of the TMI-2 debris varied spatially, but these variations are not taken into account in the modeling.

The wall heat-transfer model is modified (Refs. 2-7 and 2-8) when a porous debris bed is present. Three heat-transfer regimes are identified, and a different equation for the rate of heat removal is used for each regime. The regimes are dryout ($\alpha_g > 0.9999$), quenched ($\alpha_g < 0.9999$ and $T_{debris} \leq T_{sat}$), and transition between dryout and quenched ($\alpha_g < 0.9999$ and $T_{debris} > T_{sat}$). These regimes and equations are used instead of the RELAP5 wall heat-transfer model when a porous debris bed is present. The dryout regime assumes all the heat is transferred to the gas and that the heat-transfer coefficient between debris and gas is infinite; the gas is instantly heated to the temperature of the debris.

The quenched regime assumes all the heat is transferred to the liquid. The equation has two parts: the first part assumes all the heat generated per unit volume (P) in the debris is immediately transferred to the liquid. The second part allows for the case that the rate of heat transfer is increased by a decreasing coolant pressure and thus, a decreasing saturation temperature. The variable h_s is assumed to have a constant value of 1000 W/m2K. The transition regime assumes all the heat is transferred to the liquid, based on the assumption of a quench front passing through the debris. The model assumes debris at the location of the quench front immediately transfers all of its stored energy to the liquid, and thus, the rate of heat removal from the debris is proportional to the velocity of the quench front.

Special treatment of melt quenching is also used; as it relocates into the lower plenum, the quenching is treated in a parametric manner using one of two options. In the first option, the user specifies a quenching time, i.e., the time period during which the melt will quench provided there is sufficient water. By using a thermal equilibrium assumption, the state at the end of the quench time can be determined. The integral heat transfer to the coolant is then passed to the RELAP5 thermal-hydraulic model. The second option uses the assumption of no interaction of the melt with the water, which results in a stratified configuration in the lower plenum.

2.25.2. Qualitative Perspective

Intact and Nearly Intact Core Modeling. The use of RELAP5 models and correlations during the first two time intervals is generally viewed to be acceptable. Most of the severe-accident flow regimes are treated by RELAP5. Thus, calculations performed during these first two intervals would be expected to produce best-estimate results.

One potential area of weakness is the heat-transfer modeling of laminar forced convection flow for vapor and turbulent-free convection flow for vapor. These models are viewed to be deficient based on the assessments in the RELAP5 Models and Correlations document (Ref. 2-11). The authors urge further study of this area because proper treatment of vapor flow is important for calculating natural circulation. The discrepancies noted in the Westinghouse 1/7-scale experiments (Ref. 2-12) may be partly due to inaccuracies in the heat-transfer correlations for vapor flow. Significant concern also exists in using correlations that are not representative of a geometry of interest and employing these correlations beyond their original database.

Core Debris Region Modeling. For dried-out porous debris beds, heat transfer is not modeled in a mechanistic manner; the modeling assumes instantaneous equilibrium. This means that potentially important rate effects, which may influence the flow behavior, will not be calculated. Large debris regions that extend over multiple RELAP5 volumes may yield erroneous results. The modeling is deficient and should include rate effects in a mechanistic manner. Several studies in the open literature (e.g., Ref. 2-13), provide an approach to accomplish this recommendation.

For quenching cases, instantaneous equilibrium is assumed between the water and debris. After quenching, all heat is immediately transferred to the coolant. A constant heat-transfer coefficient accounts for depressurization effects. Again, important rate effects are neglected. The modeling is deficient and should include rate effects in a mechanistic manner. Reference 2-13 provides an approach to accomplish this recommendation.

The correlations used for calculating degraded geometry and core debris region friction losses are deficient. Pipe flow correlations that provide a hydraulic diameter modified for the debris conditions are used with SCDAP. Correlations applicable to flow through porous media should be implemented. This is evident from the fact that poor agreement is observed between code-predicted results of bottom quenching and results from a laboratory-scale experiment.

The treatment of thermal hydraulics in nodes that were blocked and then had the blockage melt away is very uncertain. Proper treatment of such nodes will be important for reflood calculations. The current modeling is deficient in treating this case.

Lower-Plenum Region Modeling. The modeling of the interactions of molten material with water in the lower plenum is treated in a parametric manner. Only approximate or bounding calculations can be performed. Best-estimate calculations of the expected behavior cannot be performed within the scope of the current modeling because important rate effects are neglected. The modeling is deficient and should include rate effects in a mechanistic manner.

2.25.3. Technical Adequacy

The use of RELAP5 models and correlations during the first two time intervals should be technically adequate. Calculations performed during these first two intervals are expected to produce best-estimate results.

As core degradation progresses, the applicability of these models diminishes. During the later intervals, SCDAP models, which are relatively simple, are used to predict heat transfer to the coolant. The models are judged to be largely parametric without sufficient validation to demonstrate applicability for all intended applications of the code. The models will not necessarily yield best-estimate predictions for the range of severe-accident conditions that might be anticipated. As such, the thermal-hydraulic modeling in Intervals 3 and 4 is judged not to be technically adequate according to the Committee's evaluation criteria.

2.26. Options for Additional Models Currently Being Developed or Upgraded

The Committee examined several additional models not fully implemented at the time of the review. A review of these models is presented in this section in summary fashion with details provided in Appendix E.

2.26.1. BWR Control-Blade and Channel-Box-Component Models

2.26.1.1. Phenomenological Package Description. The CORA experiment-specific BWR bundle heating and melting experiments model has been modified and adopted as a new control-blade and channel-box component package in SCDAP/RELAP5. The model is based on slab geometry and accounts for the fact that the stainless-steel control blade must melt and relocate before it can interact with the zircaloy channel box. The model also includes the effect of B_4C/stainless-steel interactions to more accurately predict the control-blade material relocation and also account for the effect of stainless-steel/zircaloy interactions.

2.26.1.2. Qualitative Perspective. The approach appears to be reasonable. The model contains sufficient detail consistent with the objectives of the SCDAP/RELAP5 code. The model accounts for important physico/chemical phenomena and B_4C/stainless steel, as well as stainless-steel/zircaloy interactions.

2.26.1.3. Technical Adequacy. No special deficiencies of the model have been identified because the SCDAP/RELAP5 code with the new BWR control-blade and channel-box component model has been exercised only to a very limited extent, and some inconsistencies have been observed. No benchmarking and validation studies have been reported. Model developments in progress now and/or in the near future and future validation studies should remedy the problem. Validations and benchmarkings are essential because the earlier model for CORA-18 BWR bundle heating and melting experiments significantly overpredicts the measured temperatures in the lower half of the bundle.

2.26.2. Zirconium/Inconel Eutectic Model for Grid Spacers

2.26.2.1. Phenomenological Package Description. A new model for spacer grids that takes into account the zircaloy/Inconel eutectic reaction has recently been developed. The rate of growth of the reaction zone is described by parabolic rate equations developed at Kernforschungszentrum Karlsruhe (KfK), which take into account the presence of existing zircaloy oxidation. Various simplifying assumptions are made to map the growth of the reaction zones in the spacer grid and cladding onto the one-dimensional system appropriate to the experimental correlations.

2.26.2.2. Qualitative Perspective. This is the first known attempt to model zirconium/Inconel grid spacer interactions in a mechanistic code. The treatment adopted appears physically reasonable, and initial comparisons with data from the CORA-2 experiment are encouraging. Comparisons of model predictions with data from other facilities and under different test conditions (e.g., low heatup rates) would be helpful.

2.26.2.3. Technical Adequacy. The modeling is first order and represents a significant advance over the previous treatment. It is too early to give a formal categorization because of a lack of sufficient assessment against independent data.

2.26.3. Lower-Head Failure

2.26.3.1. Phenomenological Package Description. Heat transfer from the core debris (solidified or liquefied) to the vessel wall and to the penetrations determine the modes in which the vessel lower head can fail. The mode of vessel failure is also a function of thermal loading of the structures. Examples of vessel failure modes are: tube ejection, tube rupture, localized failure of the vessel, global failure of the vessel, etc. At present, only heat transfer from solidified core debris to the vessel wall is considered, and this is also considered in a parametric way. More recently, as part of the activity supporting the Savannah River reactors, a correlation for heat transfer from liquefied debris to the vessel wall has been included. However, this correlation is flawed because it is applicable to a circular trough and not to a spherical cavity.

For heat transfer from solidified or liquefied debris to a structure, gap conductance is defined. Gap conductance between solidified debris and structure is specified by the user. For liquefied debris, a natural convection heat-transfer type of correlation is used. Recently, under another research program sponsored by the NRC, models for investigation of modes and timing of vessel failure have been developed. If these models are implemented in SCDAP/RELAP5, they should enhance the capability of the code in analyzing the modes and timing of vessel failure.

2.26.3.2. Qualitative Perspective. The present model is parametric and of very limited value. At present, no models are available in the code with respect to localized failure of the vessel at penetrations. The inclusion of the models developed for Savannah River reactors should mitigate the situation somewhat for heat transfer to the lower head.

2.26.3.3. Technical Adequacy. At present, the model for vessel lower-head failure is severely deficient in predicting the mode and timing of vessel failure.

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3. TOP-DOWN REVIEW: FINDINGS RELATIVE TO CODE-DESIGN OBJECTIVES

The specific code-design objectives were provided by the NRC (Refs. 3-1 and 3-2) and are listed in Section 1. Along with each design objective, the NRC provided success criteria that allowed the Committee members to evaluate the technical adequacy of the SCDAP/RELAP5 code. In this section, Committee findings are documented for each of the five specified code objectives.

3.1. Design Objective 1

3.1.1. Statement of Objective and Success Criteria

Modeling detail shall be capable of representing key and important phenomena of severe-accident experiments, the TMI-2 accident, and anticipated plant accidents and transients.

General Success Criterion: The code can model the PWR and BWR reactor coolant systems and operator actions plus experimental facilities used for code assessment.

a. Expected modeling uncertainties should be comparable to uncertainties in integral severe-accident experiments and TMI-2 accident conditions and results.

Success Criterion: Uncertainties in important parameters calculated by the code should be less than or equal to measured values. For example, if the uncertainties in the measured bundle temperatures and associated boundary conditions are +/-20%, then the success criteria for the code would be +/-20% for calculated temperatures.

b. User-defined parameters, other than those needed to define experiment or plant-unique features, should be eliminated where experimental or other credible bases exist to define those parameters.

Success Criterion: The code will not contain any user-defined parameters other than those noted.

3.1.2. Specific Findings Relative to General Success Criterion

The simulation of a severe-accident sequence involves calculating a large number of phenomena over long periods of time, even for the in-vessel interval. The importance of any particular phenomenon will change as the accident progresses. To account for the changing priority of phenomena in the review process, the findings for this objective have been divided into four time intervals:

INTERVAL	DEFINITION
Thermal-Hydraulic and Neutronic Transient	Time from accident initiation until superheat in core
Core Uncovery	Time from superheat in core until maximum core temperature is 1500 K
Core Heatup/Oxidation/ Relocation and Slumping in Core	Time from when core is at 1500 K, through the oxidation transient, and up to formation of molten pool
Lower Plenum and Vessel Failure	Time from when molten material relocates into lower plenum until vessel failure

While these intervals are somewhat arbitrary, the Committee decided that this partitioning was a reasonable approach for the review process. For each interval, a table has been developed that ranks the phenomena that are most important (see Appendix D). The findings in this section address the integration of these key phenomena and whether or not SCDAP/RELAP5 is capable of modeling PWR and BWR reactor coolant systems, operator actions, and experimental facilities used for code assessment.

The Committee agreed that the review would focus on the 7X version of the code. This was interpreted to mean that the Committee would evaluate the results of applications of this particular version of the code against the NRC-specified criteria. Analyses performed with earlier versions of the code were considered. The Committee acknowledges that earlier versions of SCDAP/RELAP5 have been used to address accident management strategies and the lower-head failure, as well as a large number of experiments. The Committee understood that these previous analyses directed toward severe-accident issues were being independently reviewed and the Peer Review Committee did not review these analyses performed with earlier versions of the code in significant detail.

The Committee was formally provided with the results of three plant calculations: a pump seal LOCA simulation for the Surry plant (PWR), an LBLOCA for the Browns Ferry plant (BWR), and a station blackout simulation for the Forsmark plant (BWR). The Committee was informed of a set of five additional Surry calculations, but only summary results were provided. A much larger number of experiment analyses were presented to the Committee.

None of the plant calculations had been documented in formal reports at the time of the review [some documentation is now in progress, including a NUREG/CR report related to direct containment heating (DCH)], and thus, the results should be treated as preliminary. This fact severely limited the ability of the Committee to perform the top-down review. If the Committee had narrowly interpreted the evaluation criteria, then they would conclude that Version 7X has no demonstrated and documented capability to predict PWR and BWR severe accidents. However, the Committee took a more liberal interpretation of its charter and considered the preliminary plant calculations in assessing the code against the evaluation criteria.

Only the PWR calculation was run to vessel failure. The BWR calculations were run to the point of initial core melting. While there was evidence of many PWR plant calculations performed with previous versions for the code, the Committee was not provided with any BWR calculations other than the two partially completed calculations discussed above. It should be noted that the Committee had requested that a complete BWR calculation be performed for the peer review, but such a calculation was not completed during the peer review process.

Thermal-Hydraulic and Neutronic Transient. This interval represents the time period from accident initiation until there is superheated vapor in the core. During this interval, the important capabilities that should be in SCDAP/RELAP5 include the modeling of two-phase flow, fission heat, decay heat, and plant-specific system features. In principle, the code has modeling to deal with the PWR- and BWR-system, thermal-hydraulic response through the coupling to RELAP5. However, the RELAP5 models were not reviewed by the

Peer Review Committee. Nevertheless, the coupled code is expected to have the same capabilities related to system thermal hydraulics as the standalone RELAP5 code.

The simulations of the TMI-2 accident [A40, A61 (App. B)], PWR calculations [A4, A56 (App. B)], and LOFT FP-2 [B5 (App. B)] experiment indicate that the overall system thermal-hydraulic response is adequately modeled for PWRs and experimental facilities for this interval. The BWR calculations presented to the Committee did not provide sufficient evidence that the code would adequately predict BWR thermal-hydraulic behavior.

The core heat-transfer modeling in SCDAP is adequate for predicting the thermal response during the early interval of a severe accident. This finding is based on the comparisons to experimental data [B8 (App. B)], as well the detailed model review in Section 2. The electrical heating modeling also is adequate for modeling the CORA experimental facility.

Severe-accident neutronic transients (i.e., those that depart radically from normal operating conditions) will not be calculated correctly with SCDAP/RELAP5 because the neutronic modeling is limited to point kinetics.

Core Uncovery. This interval covers the period from core uncovery until the time the peak temperature in the core reaches 1500 K. The 1500 K value was chosen as a point where only limited core damage would have occurred and rapid oxidation would not have started.

The code has sufficient modeling to treat core uncovery and boiloff in PWR plants. Again, this assessment is based on the coupling to RELAP5. This capability was demonstrated in the TMI-2 simulations [A40, A61 (App. B)] and in the PWR plant calculations [A41, A56 (App. B)].

While the overall capabilities for this interval are adequate, it is important to note that the coupled code will have the same limitations as the standalone RELAP5 code. In particular, the multidimensional hydraulic cross-flow modeling can introduce limitations in modeling important phenomena. If the flow is predominantly one dimensional, then the code should yield reasonable results. However, the flow may not always be one dimensional. The results of assessment calculations suggest that poor agreement with experimental data may be due to the simplified flow modeling. Certain phenomena, such as counter-current flow in the hot legs and natural convection in the reactor pressure vessel, will require special modeling to capture the dominant effects [A41 (App. B)].

The code has not been sufficiently exercised to demonstrate that the modeling of this interval is adequate for BWR plants for a wide range of accident conditions. The Committee was provided only with the results of two preliminary calculations, and no BWR calculations have been formally documented. The one calculation presented to the Committee was for a LOCA simulation. In this case, the prediction of core uncovery is dominated by the reactor coolant system blowdown and is only a modest test of the core uncovery modeling.

The code models ballooning, which would be expected to occur during this interval. This capability is probably adequate in its current form in many circumstances, although there are cases (e.g., for low cladding stresses in the high alpha-phase region or where there is significant cladding oxidation) where ballooning is substantially overpredicted. In these cases, the excess ballooning must be limited by input data. As demonstrated in recent calculations, the code predictions are very sensitive to this model (Ref. 3-3). This may indicate a sensitivity to nodalization and the fact that all rods in a given cell balloon at the same time.

The capability of the code to predict fission-product behavior during this interval could not be adequately assessed due to a lack of calculations.

Core Heatup/Oxidation/Relocation and Slumping in Core. This interval covers the interval from the point where the peak core temperature is 1500 K until a significant amount of core material relocates into the lower plenum. During this interval, the capabilities that are needed include modeling of heat-transfer, oxidation, melting, relocation, and fission-product behavior.

In general, the conduction and convection models used to predict core heatup are adequate for PWRs. This finding is based on the code assessment work [B8 (App. B)], as well as the review of the models (see Section 2).

The PWR calculation presented to the Committee (Ref. 3-3) exhibited a surprising result. The core was noded into 10 axial levels and 3 radial rings. After significant core damage had occurred, the calculation showed that the center and outer rings had blocked and were melting, while the second ring remained relatively cool. Apparently, the timing of blockage formation occurred in such a way that all flow was directed through the second ring. This kept the second ring cool and relatively intact. This result shows a core melt-progression picture that had never been considered in the past.

The Committee judged this melt-progression picture as being unreasonable for three reasons. First, this picture is not supported by the evidence at TMI-2. While there is some evidence of nonsymmetric behavior in TMI-2, the physical evidence does not support an annular meltdown of the core. Second, the code prediction resulted in a physical state in which material with temperatures greater than 3000 K was within 30 cm of material that had a temperature of 1500 K and yet no radiation heat transfer was calculated by the code. Third, the Committee was told that the logic that shuts off flow was limited to the axial direction only. Blockages were expected to be large, and radial flow should also have been shut off. Failure to do this resulted in the flow diversion through the second ring. The Committee believes that detailed nodalization studies would not necessarily support the results seen in this calculation.

The calculation of the in-vessel natural circulation model uses a reasonable engineering approach but requires experimental data for validation because of the simplification in the momentum equation needed to decrease the complexity of the numerical method. By using this simplification, the calculation of transverse momentum is not conservative, and the predictions of the flow distribution will be nodalization dependent. With proper experimental data, the validity of this approach can be established. Otherwise, extensive analytical studies will be needed. Such studies have been proposed, and the Committee understands they will follow the assessment of the code against the Westinghouse 1/7-scale experiments.

The modeling of counter-current flow in the hot legs uses a reasonable engineering approach. The merits of this approach have been reviewed separately and are not evaluated in this peer review.

The oxidation modeling for PWRs may be deficient during the later part of this interval. The modeling probably is adequate for intact geometry, based on the results of assessment calculations [B8 (App. B)]. However, the modeling does not allow molten, relocating material to oxidize [A4 (App. B)], and total hydrogen generation will be significantly underpredicted. Total hydrogen generation during the in-vessel interval is calculated to be nearly 50% lower than the estimates developed in the NUREG-1150 study for the same accident scenario (Ref. 3-4). The results of assessment calculations indicate the code is within +/-40% for experiments without reflood and are biased by a factor of two in reflood cases. Furthermore, the debris characteristics are predetermined by the TMI-2 particle size.

The thermal-hydraulic modeling for debris beds is deficient in the RELAP5 part of the code because of the use of pipe flow correlations instead of porous media correlations. As discussed in the bottom-up review, this is not a reasonable modeling approach.

The treatment of eutectics is very limited. For example, no interactions of zirconium with ZrO_2 are allowed. PHEBUS test results indicate that these interactions may be important [E13 (App. B)].

The treatment of molten pool formation and growth is very simplistic. Important processes such as melt migration are not treated in a mechanistic manner. Furthermore, the failure criteria are overly parametric and are unlikely to be valid for a range of postulated accident conditions. As illustrated in the calculations presented by Knudson (Ref. 3-3), the parametric modeling leads to molten pools that have "gaps" but are treated as continuous. This picture is not physically reasonable.

The coupling to RELAP5 is not integrated after blockage formation. The code shuts off RELAP5 volumes that have blocked, which precludes any possibility of predicting reflood of these regions even if the blockage melts and relocates. This modeling deficiency will bias any studies of the effect of water addition to a degraded core. Although this problem existed with the code version reviewed by the Committee, this apparently has been corrected and may not represent a problem with the very latest code version.

The modeling of core heatup and oxidation is deficient for BWRs. For example, control-blade and channel-box models do not represent experimentally observed interactions between the blade and channel box. The code has not demonstrated the ability to predict massive core melting and vessel failure for BWR plants.

Lower Plenum and Vessel Failure. SCDAP/RELAP5 uses very simple modeling in this time interval, and important rate effects are not predicted in a mechanistic manner. The only exception here is the use of the COUPLE modeling, which is a mechanistic heat conduction model. The current modeling would not be expected to predict the later intervals of TMI-2 in a credible manner. Virtually all models in this time frame are deficient and could be improved. As discussed in the presentation by Knudson (Ref. 3-3), only bounding estimates of the melt progression behavior can be calculated by SCDAP/RELAP5 at this time. Long-range model development plans provided to the Committee indicate that any remaining parametric models, such as the melt slumping model, will be replaced as more detailed models are developed.

3.1.3. Specific Findings Relative to Success Criterion "a"

It is very difficult to develop findings for this success criterion because systematic evaluations of uncertainties in experiments or in the TMI-2 accident, except for the early intervals, are very limited. A systematic evaluation of uncertainties for the early intervals of an accident has been initiated, but the results are incomplete. Another problem is that there are virtually no counterpart or repeat tests that would allow the technical community to develop a realistic uncertainty estimation in the experiments. Therefore, it is difficult to conclude that the calculations lie within experimental uncertainty.

The situation is further complicated by the very nature of the experiments that are conducted. The experiments must operate at high temperatures, which will naturally make heat losses important. The computer code must be able to model these heat losses or else it cannot model the test.

3.1.4. Specific Findings Relative to Success Criterion "b"

There seem to be relatively few user-defined parameters during the early intervals of an accident. Later, the melt relocation temperature, ballooning model parameter (maximum strain), and crust failure criterion are key parameters. Variations in each of these have significant effects on the calculations. The following is a list of key user-defined parameters and their perceived impact.

Parameter	Impact
Fragmentation Temperature	Low
Minimum Debris Flow Area	Medium
ZrO_2 Failure Temperature	High
ZrO_2 Failure Thickness	Medium
Crust Failure Thickness	High

Melt Relocation Time	Medium
Maximum Strain	High
Double-Sided Oxidation	Medium
Lower-Plenum Debris Heat-Transfer Model	High
Melt Contact Fraction with Lower Head	Medium

This assessment indicates that there are a number of input parameters that could significantly affect the predicted course of an accident, especially during the later intervals.

3.2. Design Objective 2

3.2.1. Statement of Objective and Success Criteria

The code should provide reasonable predictions of the in-vessel, melt-progression phenomena during the course of a severe accident. It should also permit estimates of the uncertainties of severe core-damage predictions without requiring modifications to the code.

Success Criterion: The code predicts major trends for dominant phenomena based on assessment against integral facility data. The code also predicts values of important parameters associated with dominant phenomena within measurement uncertainty when assessed against integral facility data. The code employed for these assessments would be the frozen-released code without any code modifications made during the period of application.

3.2.2. Specific Findings Relative to Success Criterion

It has not been demonstrated that the code can predict fission-product behavior. Such calculations have not been attempted because of excessive runtimes.

Multiple code versions have been used in assessment calculations, and use of multiple versions prevents a systematic evaluation of the modeling. Even for the calculation supporting the peer review, the code used was still under development.

A difficulty in assessing the capabilities of the codes is that the amount of experimental data is very limited for the later intervals of the accident. In general, there are sufficient data through the middle of the third interval for PWRs. Data are very limited for

BWR conditions. In the later intervals, data are virtually nonexistent except for information from the TMI-2 examinations.

The findings for this objective are also divided into the four time intervals used in the previous section.

Thermal-Hydraulic and Neutronic Transient. The results from the comparisons with available experimental data during the first interval indicate that the integral code predictions during the initial interval are adequate if the test conditions are properly considered. Any experimental analysis will require some degree of model calibration to agree with the transient boundary conditions. SCDAP/RELAP5 may be better than other codes in this area (PHEBUS B9+).

The MOD3 version of the code appears to be less accurate than earlier versions for selected tests (Ref. 3-1). Currently, poorer agreement is attributed to the changes in the noncondensable gas models between MOD2.5 and MOD3 thermal-hydraulic models. Work is currently under way on RELAP5/MOD3 model development activities to improve the noncondensable gas models.

The code apparently was able to adequately predict the thermal-hydraulic part of the TMI-2 accident [A61 (App. B)]. This calculation was performed with an earlier version of the code. Furthermore, accurate calculations could only be produced when detailed representations of the steam generators (both primary and secondary side) were used. The time to uncovery could be reasonably predicted given a detailed model and self-consistent assumptions. The TMI-2 model is fairly detailed with respect to operator actions. No effort has been made to investigate how accurate the predictions would be if a less-detailed model were used.

Core Uncovery. The severe-accident database to support assessment during this interval is very limited. Some of the available tests are difficult to interpret due to neutronic coupling effects (e.g., PBF and FLHT).

The analysis of TMI-2 provides some indication that the code results for this interval in PWR systems may be reasonable [A61 (App. B)]. Although there is considerable disagreement concerning what the operators actually did, the code predictions provide reasonable estimates of the boiloff and final water level in the core. This indicates that mass and energy are conserved with acceptable accuracy, and the transport of material is predicted to within an acceptable degree of accuracy.

In the DCH issue calculations, the prediction of the boil-off seems reasonable (Ref. 3-3). However, the sensitivity of the predicted results to ballooning behavior is questionable.

BWR calculations available to judge the adequacy of the modeling during this interval are more limited than for PWRs. More calculations, covering a wider range of accident conditions, would be needed to judge the adequacy of the modeling.

Core Heatup/Oxidation/Relocation and Slumping in Core. There is a lack of data in the later part of this time interval. The heatup of the fuel rods seems reasonable in experiments [F12 (App. B)]. In plant calculations, the heatup of the core is not always reasonable. For example, relatively minor changes to the ballooning model's maximum strain parameter resulted in a significantly different heatup of the core and melt progression behavior [F14 (App. B)]. The reasons for this behavior have not been fully explored.

The code underpredicts the amount of hydrogen by ~50% during reflood (Ref. 3-9). The major characteristics of the TMI-2 accident are unlikely to be calculated accurately with the code during this interval, and no BWR calculations have been run through this entire time interval.

Lower Plenum and Vessel Failure. The lack of mechanistic modeling severely limits the ability of the code to provide reasonable predictions during this interval. Limitations identified in the detailed review of the COUPLE model reduce the ability of the code to accurately predict flow and heat transfer in lower-plenum debris beds. Even though COUPLE is integrated into SCDAP/RELAP5 calculations and is initiated with core debris in the lower-plenum COUPLE mesh, these calculations are not fully integrated. The COUPLE model does not treat melt relocation in the debris, and the standalone calculations cannot adequately address uncertainties in the initial and boundary conditions.

3.3. Design Objective 3

3.3.1. Statement of Objective and Success Criteria

The code should be applicable for severe core-damage studies under various accident sequences for both PWRs and BWRs.

Success Criterion: The code can predict the core damage resulting from risk-dominant accident sequences identified by PRA studies for both PWRs and BWRs. Physical models, as well as component models, exist sufficiently to accurately predict dominant phenomena.

3.3.2. Specific Findings Relative to Success Criterion

The code is deficient in its demonstrated ability to analyze risk-dominant sequences. In the BWR case, no risk-dominant accident calculations have been completed. Key components are not adequately modeled in the core (e.g., control blades, core blades, and control-rod drives), especially with interactions among components. The staff at ORNL is currently attempting to improve the modeling.

The situation is slightly better for PWRs. However, the calculations have been very limited, and no comparison to other calculations have been made. In fact, the only complete calculations have all been slight variations of either a TMLB' or a TMLB' with a pump seal LOCA. Other users have tried to run other calculations but have encountered code problems [E13–E21 (App. B)]. More calculations would be needed to judge the applicability of the code for its intended applications. Also, no calculations have been published evaluating the source term to the containment, which may be considered a very dominant phenomena that requires accurate prediction.

Appendix C lists the dominant phenomena associated with PWRs and BWRs and identifies specific code models associated with each phenomenon. Appendix D provides a ranking of the dominant phenomena by severe-accident intervals and gives associated code models and relative importance, as well.

3.4. Design Objective 4

3.4.1. Statement of Objective and Success Criteria

The code should be robust, portable, and fast running.

General Success Criterion 1: While runtime is machine dependent, the following general expectation can be used: runtime should be reasonable so as to not handicap the user's ability to perform sensitivity/uncertainty analyses for the phenomena/conditions the code is designed to model. Runtime should be a small fraction of the time required to perform the entire analysis.

General Success Criterion 2: Based on user's guidelines and lessons-learned information in the code manual, code users shall be able to set up a plant model (e.g., an input deck) to truly represent a full-scale LWR plant and successfully perform plant calculations for various severe-accident scenarios, which are in the domain of targeted applications of the code.

a. The code should not abort prematurely because of user-input errors or numerical nonconvergence but should exit with sufficient diagnostic messages for users.

Success Criterion: The code performs as noted.

b. Numerical precision should be compatible with modeling precision. Spatial convergence should be compatible with the modeling scale. Timestep control should be automatic.

Success Criterion: The code performs as noted.

c. The code should be transportable for mainframe and workstation computing machines.

Success Criterion: The code is transportable.

3.4.2. Specific Findings Relative to General Success Criterion 1

The code runtime is large but not unreasonable. A larger fraction of the analyst's time is devoted to problem setup and not to running the code [E13 (App. B)]. For experimental analysis, runtimes seem acceptable [E13 (App. B)]. Runtime can be very long if fission-product models are used.

3.4.3. Specific Findings Relative to General Success Criterion 2

The problem setup time for a plant calculation is significant. This conclusion is supported by the fact that less than 10 plant calculations have been run [E14 and E15 (App. B)].

3.4.4. Specific Findings Relative to Success Criterion "a"

A major deficiency in the code is its robustness. Except for INEL, no user group has been able to complete a plant calculation with MOD3 or earlier versions [E13, E20, E14, and E16 (App. B)]. The users complain about water property errors, but this is likely a symptom of a deeper problem. One user found numerical stability problems particularly burdensome during the CORA-13 [International Standard Problems (ISP)-31] simulation after quenching. Nonconvergence or divergence of results has been offered as possible root causes.

3.4.5. Specific Findings Relative to Success Criterion "b"

Results sometimes seem to be timestep dependent. CORA-9 is the quoted example [E-13 (App. B)]. Users have experienced severe problems until the code reduced the timestep to a very small value, and problems became progressively worse as core damage and oxidation progressed.

3.4.6. Specific Findings Relative to Success Criterion "c"

Portability seems to be acceptable. However, one of the code users reported that antique FORTRAN still exists in RELAP5 (use of shift and mask operations for bit manipulations, for example) and probably should be removed. Users report that answers differ on different machines [E13–E21 (App. B)].

3.5. Design Objective 5

3.5.1. Statement of Objective and Success Criteria

The maintenance of the code should follow accepted QA standards for configuration control, testing, and documentation.

General Success Criterion: Code-design assurance procedures and associated documentation should be sufficient to allow the certification of the code for ANSI/ASME NQA-1 or the equivalent where required for NRC applications.

a. All code changes should be controlled and verified by redundant means.

Success Criterion: Code changes should be made as noted.

b. Testing standards and benchmarks should be defined for all versions released for production applications.

Success Criterion: These perform as noted.

c. Documentation should define the theoretical basis, limits of applicability, and testing or assessment results of the code.

Success Criterion: Documentation defines these as noted.

3.5.2. Specific Findings Relative to General Success Criterion

A QA procedure exists to control, test, and document changes to the code. However, the procedure is not adequate. Even though a procedure exists to provide updated information to code manuals at the same time as the code models are changed, release of new documentation occurs infrequently (approximately once per year). This results in inconsistencies between what is in the code documentation and what is in code. Presently, undocumented features, as well as documented, but unimplemented, models, exist in the framework of the QA program.

Conformance to NQA-1 has not been demonstrated. An NQA-1 QA program should be developed for the code.

3.5.3. Specific Findings Relative to Success Criterion "a"

Configuration control is maintained at a level such that every code version is uniquely identified; however, the release of beta versions leads to considerable confusion [E-13 (App. B)]. More care should be exercised in releasing versions of the code.

3.5.4. Specific Findings Relative to Success Criterion "b"

The code QA program does not conform to NQA-1 standards. For QA, the code should have an NQA-1 quality-assurance program. Within such a system there is some flexibility, but the structure is important.

3.5.5. Specific Findings Relative to Success Criterion "c"

Documentation of code modeling changes is not timely and is often confusing. Undocumented features, as well as documented, but unimplemented, models, exist in the framework of the QA program. The documentation is weak and largely dated; some is over 10 years old. If no updates to modeling have occurred over the last 10 years, even though the database has grown considerably, then the presumption is that the modeling is very poor. The Committee learned of enhancements that are not adequately documented [F1 (App. B)].

The documentation was not designed to allow easy review. Information may be available, but it is so scattered as to be virtually useless. This finding is indicative of a project that is not adequately reviewed. As part of the code documentation activity and to support future peer reviews, code developers should perform the ranking of dominant phenomena. They should prepare a document that lists dominant phenomena and presents a current understanding of them. In the same document, the code developers should provide a peer review body with their view (i.e., their models) and justify why their models are an adequate representation of each phenomenon.

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4. TOP-DOWN REVIEW: FINDINGS RELATIVE TO CODE-TARGETED APPLICATIONS

The specific code-targeted applications were defined by the NRC (Refs. 4-1 and 4-2) and are listed in Section 1. Along with each targeted application, the NRC provided success criteria that allowed the Committee members to evaluate the technical adequacy of the SCDAP/RELAP5 code. In this section, Committee findings are documented for each of the five specified targeted applications.

4.1. Targeted Application 1

4.1.1. Statement of Application and Success Criterion

Experimental analysis and support for in-vessel, severe-accident experimental programs such as CORA, PBF, LOFT, and NRU.

> Success Criterion: The code has been or can be used to analyze these facilities and can provide reasonable predictions of associated dominant phenomena. (Compared with experimental data, the calculated results will be within the experimental uncertainty bands.)

4.1.2. Specific Findings Relative to Success Criterion

The Committee felt that an appropriate way to evaluate the SCDAP/RELAP5 code-specific capabilities in modeling integral experiments was to consider the following particular time intervals during the severe accident:

1. Thermal-hydraulic and neutronic transient,
2. Core uncovery,
3. Core heatup/oxidation/relocation and slumping in core, and
4. Lower plenum and vessel failure.

The Committee recognized that these time frames are somewhat artificial in their division but are useful in delineating when SCDAP/RELAP5 models are applicable, as well as delineating the limitations of the simulations.

Our general finding is that the SCDAP/RELAP5 code can adequately represent certain experimental facilities and their associated integral tests with sufficient modeling of the thermal-hydraulic initial and boundary conditions, as well as the test geometry. A list of these facilities and the particular tests that were reviewed are given in Table 4-I. Conversely, there are particular experiments in which the SCDAP/RELAP5 code is probably not appropriate because of the small scale of the tests and the multidimensional aspects it introduces that SCDAP/RELAP5 cannot represent; e.g., the debris coolability and melt progression tests at the Annular Core Research Reactor (ACRR).

Thermal-Hydraulic and Neutronic Transient. The thermal-hydraulic transient can be adequately modeled by SCDAP/RELAP5 within the limitations of the RELAP thermal-hydraulic models. These limitations have been documented by more complete thermal-hydraulic analyses in the past (Ref. 4-3).

Core Uncovery. Fluid uncovery of a simulated core during experiments is again largely controlled by RELAP models, and the previous finding applies.

Table 4-I
Experiments Being Used as Benchmarks for SCDAP/RELAP5

• ACRR DF1, DF2, and DF4 (Only DF4 with SCDAP/RELAP5 MOD3)
• PBF SFD ST, 1-1, 1-3, 1-4 (ST, 1-3, 1-4 with SCDAP/RELAP5 MOD3)
• NRU FLHT 2,3,4
• LOFT FP-2 with SCDAP/RELAP5 MOD2.5 without reflood
• CORA 2, 7, 9, 12, 13, 17, 18 (7, 13, 18 with SCDAP/RELAP5 MOD3)

It should be noted that the Committee is concerned that the RELAP thermal-hydraulic model is used after significant core uncovery and heating begins because the correlations used in RELAP may not be applicable when mixed convection effects become important (natural and forced) during the transient. The Committee is aware of a recent report (Ref. 4-4) that compared RELAP5/MOD3 predictions of natural circulation for a single steady-state SF6 test in the Westinghouse facility. Therefore, the Committee recommends that the SCDAP/RELAP5 code be used to analyze the Westinghouse 1/7-scale transient natural circulation experiments.

Heatup/Oxidation/Relocation and Slumping in Core. Fuel-rod temperature response and hydrogen generation during heatup and oxidation depends on the ZrO_2 failure-melting temperature criterion chosen by the user. This parameter is an empirical value, where the experiments suggest a range of values that is appropriate to match the data (Ref. 4-5) The Committee felt that although the model is more sophisticated, this uncertainty causes the SCDAP/RELAP5 simulation to be parametric in character, given that all other values are adequately known.

The analysis of the SFD, CORA, and DF experiments (see Presentations 2-23 and 2-27 of Ref. 4-6) indicated that the fuel-melt relocation model in SCDAP/RELAP5, which assumed that film flow is not physically correct, and a rivulet flow pattern is expected. The importance of rivulet flow relative to the effect of grid spacers is still unclear. The INEL staff acknowledged this deficiency and is now correcting the model.

In the LOFT-FP2 and CORA tests (see Presentations 2-32 and 2-33 of Ref. 4-6) involving reflood, the SCDAP/RELAP5 model does not adequately treat clad fragmentation and oxidation due to water reflood into the core and must be improved.

Lower Plenum and Vessel Failure. All experiments analyzed by SCDAP/RELAP5 were terminated before significant fuel/clad slumping and melt-pool development occurred; thus, no direct comparison could be made to data. The availability of data for this phenomenon is quite sparse; thus, there is still a significant lack of understanding of the physical process. The Committee is aware of the recent initiation of the FARO-LWR experiments on melt quenching being partially sponsored by the NRC. These experiments, with appropriate planning and guidance, may be helpful in modeling the late-interval phenomena.

The core-slumping-into-the-lower-plenum and vessel-failure model is known to be deficient and as yet is not compared to even this sparse database. This deficiency should be corrected to provide a consistent calculation up to vessel failure. We would propose that the SCDAP/RELAP5 code developers provide a consistent model of melt transport and energy transfer to the lower-plenum water as the melt accumulates on the lower head. This will provide the initial and boundary conditions for vessel-wall heatup and possible melting. In addition, a model would have to be included for the structural response of the vessel wall. The details of such a thermal-structural model must still be specified. The model output should also provide the code user with information regarding probable modes of vessel failure.

4.2. Targeted Application 2

4.2.1. Statement of Application and Success Criterion

LWR plant analysis with and without water addition.

Success Criterion: The code has been or can be used to analyze LWRs and can provide reasonable predictions of associated dominant phenomena. The code must have been shown to predict the dominant phenomena associated with these two actions (with or without water addition) by assessment against sufficient experimental results.

4.2.2. Specific Findings Relative to Success Criterion

The Committee felt that the particular time frames mentioned previously are also useful in assessing the acceptability of SCDAP/RELAP5 for full-plant simulations. The Committee, though, emphasizes that the amount of complete plant calculations available for review is extremely limited, and none were formally documented in a written report that could be reviewed. Only an oral presentation was provided to the Committee for one complete PWR calculation. Table 4-II gives the extent of complete plant simulations performed with SCDAP/RELAP5. There are only six full-plant calculations for a station-blackout sequence with a seal-induced LOCA for a PWR and only two partial calculations up to the core heatup interval for a BWR for a station blackout and LBLOCA. Cooperative Severe-Accident Research Program (CSARP) members from other countries have attempted calculations with

Table 4-II

Full-Plant Simulations with SCDAP/RELAP5

• PWR—Station-Blackout Sequence for Surry—six cases related to DCH reactor pressure vessel/reactor coolant system failure (Case 4 presented to Committee out of six cases).
• BWR—Station Blackout for Forsmark and LBLOCA for Browns Ferry (not completed; performed up to core uncovery with limited heatup).

mixed success, and their comments have been considered (Refs. 4-7 to 4-15). Because of this situation, our findings are predicated on a primary recommendation that more efforts be employed to perform more full-plant simulations. Without such efforts, the adequacy of SCDAP/RELAP5 for use in plant analysis is in doubt.

<u>Thermal-Hydraulic and Neutronic Transient.</u> The thermal-hydraulic transient before core uncovery can be adequately modeled by SCDAP/RELAP5 with RELAP limitations.

<u>Core Uncovery.</u> The core uncovery behavior (timing and response) for the limited plant calculations the Committee has reviewed seems reasonable.

<u>Core Heatup/Oxidation/ Relocation and Slumping in Core.</u> Radiation heat transfer between components at elevated temperatures during core heatup was a noted problem for the British in their Sizewell plant calculations (see Presentation 4 in Ref. 4-16). This result is consistent with Committee comments about radiation heat-transfer deficiencies in Section 3.1.2.

Zircaloy oxidation and hydrogen generation during heatup depends, as noted previously, on the assumed ZrO_2 failure-melting temperature chosen. Based on CORA (see

Presentations 2-32 and 2-33 of Ref. 4-6) and SFD test analysis (see Presentations 2-23 and 2-27 of Ref. 4-6), the model can produce reasonable early-interval results with an appropriate choice of this parameter, leaving SCDAP/RELAP5 as a tool for plant calculations with a range of empirically specified failure temperatures.

The clad ballooning model is not considered adequate (see Presentation 2 in Ref. 4-16) because even though the model is quite sophisticated, it is missing some key effects of oxidation strengthening of the zircaloy clad. Nevertheless, the level of sophistication is not consistent with other parts of the code.

Reflood of the core by water addition cannot be handled by SCDAP/RELAP5 as LOFT-FP-2 analysis indicates (Ref. 4-17 and Presentation 2-30 of Ref. 4-6). This is a physical process that is not clearly understood from the limited data available; the code developers have agreed that this should be corrected.

Lower Plenum and Vessel Failure. The treatment of core material relocation and slumping was shown to be crude. The limited review of the PWR station-blackout plant calculations indicate that certain parametric settings were needed to successfully perform the complete simulation. A number of particular concerns were raised by the review.

1. Core slumping and melt-pool formation was assumed to be asymmetric, and this seems unphysical based on the TMI-2 experience;
2. The hydrogen generation history shows a rapid rise and then levels out during this interval; this is unphysical. This illustrates that no clad oxidation occurs during relocation, and this was an acknowledged deficiency;
3. The molten core temperatures predicted by SCDAP/RELAP5 at the time of core slumping seems unphysically large (3400–3700 K). This was a concern for the Committee because it may indicate an underlying heat-transfer difficulty. The TMI-2, post-accident examination does not support this;
4. The lower-plenum, core-slumping behavior is bounding and not currently consistent; therefore, a more consistent model needs to be developed, as previously discussed; and
5. The lower-head heatup analysis with COUPLE seems somewhat arbitrary because, although it is an adequate conduction heat-transfer model, the initial and boundary conditions are the primary determinants for the results obtained. It is these initial and boundary conditions that are questionable.

4.3. Targeted Application 3

4.3.1. Statement of Application and Success Criterion

Selected detailed analyses are needed for specific technical issues—lower-head-failure analysis, influence of water addition, natural circulation, hydrogen generation upon reflood, and accident management evaluations.

Success Criterion: The code has been or can be used to provide a detailed analysis of these specific technical issues, predicting the associated dominant phenomena with reasonable agreement. The code must have been shown to predict the dominant phenomena associated with these specific technical issues by assessment against sufficient experimental results.

4.3.2. Specific Findings Relative to Success Criterion

The SCDAP/RELAP5 code may contribute to a detailed analysis for specific technical issues, but because of inadequacies in the modeling, the code cannot be the sole tool to resolve such issues. In the Committee's opinion, at the code's current level of maturity, the SCDAP/RELAP5 code cannot be used at this time to provide such a detailed analysis. To illustrate this opinion, consider a few specific issues.

Lower-head-failure analysis is beyond the capabilities of SCDAP/RELAP5 because core-slumping and lower-head heatup cannot be treated realistically with the present set of code models. It is the opinion of the Committee that the influence of water addition currently cannot be treated with the SCDAP/RELAP5 code because there is no evidence that a validated or tested model exists to treat water reentry if the core degrades. This was demonstrated in the attempt to model the LOFT FP-2 Test. Because of this limitation, hydrogen generation upon reflood is also an issue that cannot be treated. Natural circulation issues during core uncovery and heatup should not be treated with SCDAP/RELAP5 until the code has been compared and assessed to the Westinghouse 1/7-scale natural circulation tests.

Accident management evaluations are possible before core degradation and heatup have occurred, as demonstrated by a RELAP5 depressurization study (Ref. 4-17). However, if core degradation occurs, then the amount of experience with the SCDAP/RELAP5 code is limited (Presentation 2-26 of Ref. 4-6 and Presentation 3 in Ref. 4-16) and has not been compared to integral test data to gain confidence in its use in accident management decisions.

4.4. Targeted Application 4

4.4.1. Statement of Application and Success Criterion

MELCOR benchmarking and assessment.

Success Criterion: The code has been used to benchmark and assess the MELCOR code in-vessel behavior, at least for integral experiments.

4.4.2. Specific Findings Relative to Success Criterion

The SCDAP/RELAP5 code has not been used to directly assess and benchmark the MELCOR code. Nevertheless, we are aware of standard problem exercises in which both code models have been used to compare the results to data (Ref. 4-18). The ISP-28 exercise is one such problem in which SCDAP/RELAP5 and MELCOR gave reasonable predictions (within the experimental error) of hydrogen generation for the PHEBUS SFD Test B9+. However, both models showed poor agreement with the measured amount of UO_2 dissolution.

It is the Committee's view that such a benchmark assessment activity is quite difficult to perform unless both codes are directly compared to experimental data. This leads us to the question: Should code-to-code benchmarking for assessment be supported as a reasonable activity? The Committee could not obtain a consensus on this point, although it appears there is consensus on the utility of comparisons where experimental data are directly available or code-to-code comparisons for plant calculations are available.

4.5. Targeted Application 5

4.5.1. Statement of Application and Success Criterion

TMI-2 accident evaluation.

Success Criterion: The code has been or can be used to evaluate the TMI-2 accident with reasonable prediction of the dominant phenomena. (Compared with the TMI-2 data, the calculated results will be within the measured uncertainty bands.)

4.5.2. Specific Findings Relative to Success Criterion

An earlier version (MOD2.5) of the SCDAP/RELAP5 code (Ref. 4-19) has been applied to the TMI-2 accident and was able to adequately predict the thermal-hydraulic portion of the accident. The analysis also seemed to provide reasonable results for the core uncovery early in this portion of the accident. These results could only be produced when a detailed representation of the primary circuit and steam generators was used.

The Committee is aware that SCDAP/RELAP5 is being used as part of the OECD-CSNI TMI-2 analysis. It is the opinion of the Committee that the SCDAP/RELAP5 code could not adequately produce simulation of the late intervals of the severe accident, particularly when a large-mass molten pool formed in the core, relocated to the lower plenum, and eventually quenched on the lower-plenum wall. The models simply do not exist to treat this important interval of the accident.

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5. OTHER REVIEW FINDINGS

5.1. Inconsistencies Between What is in Code Documentation and What is in the Code

5.1.1. Background

Development of a large-scale computer code requires the production and maintenance of a documentation database covering, in particular, descriptions of the models and the input data requirements, including recommendations as to the best use of the code. This forms an essential part of the QA of the code; it is important that users fully understand the models and their application if reliable conclusions are to be drawn from the analyses carried out. Furthermore, effective review of a code is helped greatly by the provision of accurate, up-to-date documentation.

5.1.2. Committee Findings

The Committee found that the documentation supplied was not always accurate and up to date, so discovering what precisely was in the models was often difficult, requiring questions to the development team to elucidate the current position (the answers to these were usually helpful and informative). It was not possible to review the code using the formal documentation alone; reference to informal papers and answers given in the review sessions was necessary. Part of the problem is that the code is changing rapidly; the difficulties that this brings are discussed in Section 5.2.

Some obvious errors found, for example, in the fuel and fission-product sections in the SCDAP code description (Ref. 5-1) suggest the need for improved quality checking of the documentation before issue. No input description specific to SCDAP/RELAP/MOD3 was initially provided for review; those given (e.g., Ref. 5-2) referred to an earlier code version (MOD1) and were obsolete. However, the draft manual (Ref. 5-3) provided by INEL to CSARP members as part of the transmittal package for the beta-test MOD3 version 7o was similarly organized to the MOD1 documentation and reflected the changes in the code input requirements going from MOD1 to MOD3 (7o). A final, related point is that the amount of detail provided in the written material (e.g., Ref. 5-1) varied considerably from one section of the code to another.

Specific recommendations regarding improvement of the documentation are made in Section 5.3.2 below.

5.2. Status of Code Assessment

5.2.1. Background

During the history of SCDAP and SCDAP/RELAP5 code development, the models have been assessed against a range of experimental data, both in INEL and the outside user community. This has resulted in many significant model improvements when code deficiencies have been identified and errors have been corrected. Assessment of SCDAP/RELAP5/MOD3 is reported most comprehensively in Ref. 5-4. This also draws on assessments of earlier code versions (Ref. 5-1). Responses to the User Survey (Refs. 5-5 to 5-13) show evidence of other assessment activities, including International Standard Problems 28 (PHEBUS B9+ test) and 31 (CORA-13 test), and the analysis of other tests in the CORA series.

The standard problems form a particularly good basis for code assessment in detailed areas. As well as providing an opportunity to benchmark the code against generally well-qualified data, the results of simulation of the standard problems also demonstrate the range of answers obtained by users of varying degrees of skill and experience. Results of ISP-28 are currently available (Refs. 5-14 to 5-17), but the code has not yet been updated in response to the deficiencies identified. Conclusions from ISP-31, which cover the important area of reflooding degraded fuel, are not yet available.

Assessment of the code also incorporates the use of idealized transients for consistency checks, e.g., numerical schemes, model options such as decay heat, and results obtained on different computers (Ref. 5-4). Useful feedback is also obtained from users in the field of plant applications (Ref. 5-18), which is also reported in newsletters (Ref. 5-19). The range of plant studies is becoming quite wide, including the TMLB' sequence, SBLOCA, LBLOCA, and V-sequence.

5.2.2. Committee Findings

The overall approach of combining the experimental assessment with the use of idealized problems for consistency checks is sound. The main concern here is whether the assessment covers a sufficiently wide range of data and transient conditions.

The validation program against experiments necessarily reflects the availability of data in the areas concerned. Thus, the models in intervals 1 and 2 have received much more attention than those in the later intervals, where relevant data are very sparse. Comparisons of the code against experimental data have tended to concentrate on temperature histories, times of ballooning rupture, and total hydrogen production.

Less attention has been paid to quantitative blockage data (i.e., location and composition); only in later works (Refs. 5-14 and 5-20) have axial blockage profiles received detailed study. No specific validation of the COUPLE models as implemented in MOD3 has been identified in the supplied documentation, apart from testing in artificial, idealized problems. The test matrix should ensure that all important parts of the code have been exercised, e.g., through the use of runtime analyzers. A more complete picture of sensitivities would be provided by comparing code results on different computers for plant calculations, rather than only experimental studies.

The changing nature of the code models raises difficulties for code assessment; this is one aspect of reviewing a code that is still under development, as discussed below. Comparisons of code calculations before and after a series of model changes have taken place are required to demonstrate clearly the effect of the modifications over a range of conditions. These comparisons should indicate quantitatively the errors in the data, errors due to model inadequacies, and uncertainties due to a reasonable choice in parametric input. While there has been some work of this nature [e.g., B-4, B-19 (App. B) and Presentation 2-27 in Ref. 5-21), more is required.

While it is noted that there has been considerable progress recently in assessing SCDAP/RELAP5/MOD3, the process cannot be regarded as being complete. The developmental assessment matrix identified by INEL (Refs. 5-21 and 5-22), as shown in Table 5-I, forms a good basis for this work, and further efforts in this direction are recommended. Good features of the quoted matrix are the use of further experiments from the CORA series (with their orderly variation of test conditions) and the use of design basis

Table 5-I
Overall Developmental Assessment Matrix for Damage-Progression and Fission-Product Models

Problem Type	Experiment
Phenomenological Problems[a]	
1. Fuel-rod heatup	NA
2. BWR channel-box/control-blade heatup	NA
3. Simulator heatup	NA
4. Nuclear-heat generation RELAP5 and SCDAP tables, point kinetics, decay heat	NA
5. Debris-bed formation in-core lumped model, 2D model	NA
6. Core relocation into lower plenum	NA
7. RELAP5-heat-structure–SCDAP-fuel-rod comparisons	NA
8. SCDAP/RELAP5/MOD3–SCDAP/RELAP5/MOD2.5 comparisons	CORA-12, SFD 1-4
9. SCDAP/RELAP5/MOD3–ICARE comparisons	SFD 1-4
10. Computer dependency (DEC-5000, CRAY X-MP IBM-6000)	CORA-9
Separate-Effects Problems [a,b]	
1. Rod bundle and rupture during LOCA	PBF LOC-3
2. Bundle ballooning and rupture during reflood	REBEKA Tests
3. Bundle ballooning and rupture during steam injection	MRBT-5
4. Debris-bed coolability	DCC and UCLA Tests
5. Debris-bed quench	BNL and UCLA Tests
6. Molten-pool formation	To be determined

Table 5-I (cont.)

Overall Developmental Assessment Matrix for Damage-Progression and Fission-Product Models

Problem Type	Experiment
7. Fission-product release	ORNL VI Test,[b] ACRR ST Tests
8. Melt-coolant interactions	To be determined
Integral-Effects Problems[a]	
1. Fission-driven bundle boildown, heatup, ceramic melting	PBF SFD-ST, SFD 1-1, SFD 1-3, SFD 1-4
2. Fission-driven bundle boildown, heatup,* metallic melting	NRU FLHT-2, FLHT-4, FLHT-5, FLHT-6, PHEBUS B9+(ISP-28), PHEBUS B9R
3. Fission-driven bundle heatup in steam	ACRR DF-1, DF-2, DF-3, DF-4
4. Decay-heat-driven core heatup ceramic melting (FP-2 only)	LOFT FP-1, FP-2
5. Decay-heat-driven core heatup ceramic melting, molten-pool formation and relocation	TMI-2
6. Electrical heat-driven bundle heatup and metallic melting with slow cooling	CORA 2, 3, 5, 7, 9, 10, 15, 16, 18, 27, 28 29, 30, 31
7. Electrical heat-driven bundle heatup and metallic melting with quenching	CORA 12, 13, 17
8. Fission-driven bundle boildown, heatup, ceramic melting with fission-product release and transport	PHEBUS FP Tests PBF SFD 1-3, 1-4
9. Decay-heat-driven core heatup ceramic melting with fission-product release and transport	LOFT FP-2

[a]Selected problems will be repeated to assess the influence of modeling changes under FIN A6889 and FIN L22302.

[b]Alternative problems may be substituted depending upon the availability of necessary test conditions.

data for assessing the ballooning model. (Here, a prediction of the blockage, as well as of the ballooning temperatures, should be considered.)

The planned Developmental Assessment Report should be completed. The report should clearly delineate how the results of the experiments will be used to strengthen, revise, or confirm the existing models. The assembly of a reference set of test cases under configuration control, covering all major areas modeled by the code, is encouraged. Participation in future relevant ISPs is recommended. It is important to establish a consistent philosophy for the code assessment work; figures of merit should be established ideally before the assessment calculations are run to judge the adequacy of the modeling. This is especially true for the treatment of melt relocation.

5.3. Difficulties Encountered by Assessing a Code That is Still under Development

5.3.1. Background

The SCDAP/RELAP5/MOD3 code is currently in a state of rapid flux. The merger of SCDAP/RELAP5/MOD2.5 ("3 series") and RELAP/MOD3 ("6 series") resulted in the 7X series of the code, with numerous sub-versions created as deficiencies have been identified and corrected. More recently, the 8X series has been created to encompass core-damage-progression model enhancements. During the review period, results from calculations with several 7X sub-versions of the code have been presented, both as experimental analyses and plant studies. The 7o version has been distributed to selected CSARP countries for beta-test assessment and forms the basis of the responses concerning MOD3 in the user survey. More recently, the 7af version has been distributed for beta testing on a more limited scale.

5.3.2. Committee Findings

Two main areas may be identified:

a. Documentation

With the code changing rapidly, the documentation lags significantly (sometimes years) behind the latest code version available. This applies to both the model (Ref. 5-1) and input data descriptions. Inconsistencies are therefore found between the code and documentation, which hinder both review and use of the code, the latter through possible misinterpretation of code results. A recourse must often be made to inspect the source code and/or query the code authors to determine exactly what model is in place in the given version, and in the case of new versions, what changes to the input data are required.

It would be helpful if changes to the specifications, particularly for input data, were placed on line under configuration control so that the code version and description were closely linked. INEL (Ref. 5-23) has suggested proposals of a similar nature in response to Advisory Committee on Reactor Safeguards (ACRS) comments on the code documentation. The code version and date of release should be clearly stated in detail on all documentation released to the users, including model descriptions and user guides. It would also help if changes between versions were clearly marked.

b. Validation

It is often difficult to identify which exact version of the code has been used for a given assessment run or application. The validation reports do not always specify the specific sub-versions of MOD3 used for the benchmarking against data nor (a related point) what values of critical input data, such as the oxide-shell breach temperature, have been used. It is not obvious that the same detailed version has been used for assessment against all the major experiments; thus, it is hard to be sure that improvement in one area has not resulted in a deterioration in others. Any official code release should be accompanied by an assessment report that compares the performance of that precise version against the agreed validation matrix; such a report should include a definition of the critical input data.

5.4. User Image

5.4.1. Background

The term "user image" is here used to cover such aspects of code use as consistency and comprehensibility of input/output, training, and general user friendliness. These matters are significant because they impact on the time spent in analysis and on the reliability of the conclusions drawn in code applications.

5.4.2. Committee Findings

a. Input requirements

The parts of the code differ markedly in their input requirements. The RELAP part is well structured with extensive input checking. The processing does not necessarily stop when the first error is encountered, thus enabling any further problems to be identified in the same attempt. Comment lines may be included to aid understanding of the input deck.

Conversely, the SCDAP part is less well structured, the format is different from that in the RELAP part, and cards are not identified by number; input errors do not always result in easily comprehensible diagnostic messages, and each error normally results in the code failing immediately or shortly thereafter in the processing, usually with a FORTRAN error. Redundant input introduces the possibility of error. There is no formal method for introducing comments in this section of the input, although short remarks can be included after the end of the numerical data demanded from each card. The COUPLE input follows yet another format.

Input to the SCDAP and COUPLE parts cannot be changed on restart. This hinders the performance of sensitivity studies, particularly in long plant calculations.

The differences in input specification reflect the composite nature of the combined code; the component input routines have not yet been fully integrated. Incomplete integration has been noted in earlier sections regarding the physical models, e.g., radiation between SCDAP components and RELAP heat structures is not modeled. The input problems would be avoided by rewriting the SCDAP part of the input using a RELAP numbered format, including the same level of error checking and diagnosis with the same facilities for commenting and data change on restart.

The Committee understands that substantial changes have been made to the SCDAP/RELAP5 input and output based on initial feedback from the code users and the Committee, and a draft document describing these changes will be produced.

b. Output requirements

The structure of the output again reflects the composite nature of the code. Detailed output (major edits) are produced at user-specified intervals and before an abnormal termination. Specific variables may be printed more frequently in the form of tables (minor edits). In addition, variables are written to the restart/plot file for subsequent graphics postprocessing.

Early versions of SCDAP/RELAP5 gave misleading and/or inconsistent printed output, particularly from the SCDAP routines. This has now been largely eliminated in MOD3. However, the selection of variables for plotting is inconsistent between SCDAP and RELAP; in the former, all variables for plotting must be specified explicitly in all their dimensions; in the latter, a standard set is dumped, while others need to be specified explicitly. Many of the RELAP variables are not important in much severe-accident analysis, and their presence increases the size of what may already be a large restart/plot file. Conversely, the lack of a standard set of SCDAP variables (or even of an "implied DO loop" facility) increases the size of the input deck.

It is therefore recommended that (1) the standard list of plotting variables be reassessed for severe-accident needs, (2) the ability to deselect variables from the list be introduced, and (3) an abbreviated format for multiple plot requests, e.g., for over the complete length of a SCDAP component, be provided.

It is not easy to identify key events in the printed output, e.g., ballooning rupture, melt relocation, and debris-bed formation. It is recommended that a common highlighted format be adopted and that the key event information be duplicated in the on-line ("tty") output stream.

c. Training

Training in the use of the code is generally obtained on the job and through attendance at user workshops. The experience of the latter has generally been good, particularly if extensive hands-on participation is featured. Exchange of experience among users and code developers is another valuable aspect of the workshops and particularly of user seminars. While user guidelines are available for RELAP5 (Ref. 5-24), more written guidance is needed for use of the SCDAP and COUPLE parts of the code; a similar volume to Ref. 5-24 would be welcome here. The use of template inputs provided by experienced users of the code (and/or the code developers) to illustrate good modeling practice would assist in efficient application of the code.

d. General user friendliness

Overall, the user friendliness can be characterized as moderate. The code still has the look and feel of a research tool; in the hands of a skilled and knowledgeable user, it is capable of providing good results in the areas of experimental support and analysis, such as for the CORA series; experience has been more mixed for detailed plant studies. The user must always check the results of calculations carefully to ensure that the results are physically reasonable and consistent. The general appearance is that full integration of the subprograms making up the whole, while well advanced, has still not been completely achieved.

The Committee recognizes that efforts are under way to alleviate some of these concerns. Furthermore, even though SCDAP/RELAP5 was thought by code users to have some problems, it is nonetheless being used extensively in the international community because it reflects the state of the art.

5.5. Design Philosophy of Coupled Code

5.5.1. Background

SCDAP/RELAP5, while one integrated code now, represents the combination of two previously existing codes. The importance of thermal hydraulics in severe accidents makes it essential that a code such as SCDAP/RELAP5 have adequate thermal-hydraulic models. It was not the charter of the Committee to review all of the thermal-hydraulic models in RELAP5. However, the Committee was interested in whether or not the design philosophy of the coupled code provided sufficient emphasis on the thermal-hydraulic models needed for severe accident simulations. Thus, the Committee reviewed the conceptual design reports for the coupled code.

5.5.2. Committee Findings

The technical evaluation of the thermal-hydraulic models needed for the coupled code focused on the early intervals of the accident (Ref. 5-25). Proper treatment of natural circulation and flow around blockages were the main thermal-hydraulic modeling needs identified for the coupled code. It should be noted that the mission of SCDAP as stated in Ref. 5-25 is more limited than that which the NRC provided to the Committee.

Technical requirements related to the flow through porous media, quenching of debris, and lower-plenum, melt-water interactions were not considered in the documentation of requirements as supplied to the Peer Review Committee. As stated in Ref. 5-25, the view at that time was that SCDAP would be run only for the early intervals of the accident.

The Committee did not receive sufficient evidence that technical requirements were established to extend the mission of the coupled code to simulating the later intervals of a severe accident. The available documentation did not specify the dominant phenomena to be calculated in the later intervals or how these phenomena should be modeled.

5.6. Summary of Responses to the SCDAP/RELAP5 User Survey

5.6.1. Background

A questionnaire was sent to users of the code requesting detailed information related to code-design objectives and targeted applications (see Appendix F for a sample questionnaire). Comments were returned from nine user groups, including:

	Country or Organization	Code Version
•	The United Kingdom	MOD2.5/v 361sc and v 3f, MOD3/v 7o
•	Spain	MOD2.5/v 3f
•	Finland (2)	MOD2.5/v 3f
•	Korea	MOD2.5/v 3f
•	Switzerland	MOD2.5/v 3f
•	The Netherlands	MOD2.5/v 3f, MOD3/v 7o
•	Germany	MOD2.5/v 3f
•	Oak Ridge National Laboratory	MOD3/v 7q

These user groups have experience that covers both experimental analysis and plant accident simulations. Table 5-II shows code applications indicated by user group responses. Table 5-III shows the range of computer hardware employed by users and general code runtime performance.

5.6.2. Survey Findings

Most users were positive in their replies, but all have experienced some degree of difficulty in using the code. Five areas will be presented that generally encompass user feedback: (1) Experience: Numerical Behavior, (2) Experience: Physical Models, (3) Comments on Input, (4) Comments on Transportability, and (5) General Comments.

Table 5-II
Applications of SCDAP/RELAP5 by User Groups

Type	Number of Cases
CORA-13 (ISP-31)	4
PHEBUS B9+ (ISP-28)[a]	2
Other CORA Tests	5
PWR	
TMLB' Sequence	6
SBLOCA	4
S_2D Sequence	1
LBLOCA	1
VVER (Russian reactor)	2
Small BWR (not complete)	1
Asea Brown Bavarian BWR	3

[a]ISP-28 attracted a total of five submissions (semiblind phase) and four submissions (open phase) using SCDAP/RELAP5; these figures include participation from users not covered in the survey response. Five submissions with the code were made for the blind phase of ISP-31.

Table 5-III
Computer Hardware Employed by Users/Runtimes

Machine	Average CPU-s/Real Time-s
CRAY Y-MP/264	55
SUN	2200 2.3–2.5 2
MicroVAX	125
CRAY X-MP	10–20

<u>Experience: Numerical Behavior.</u> Code failures due to water property errors was a common theme expressed by code users. This was thought to be due to the linkage between SCDAP and RELAP5. Because of excessive code failures, analysis of late intervals of severe accidents was not obtained by any users. It was also reported that results and numerical stability are quite sensitive to timestep size. One user found numerical stability problems particularly burdensome during the CORA-13 (ISP-31) simulation after quenching. It was found that the code could not handle noncondensable gases properly.

Furthermore, it was believed that excessive numerical diffusion in the calculations leads to nonconvergent thermal hydraulics, and when nonconvergence occurred, the code did not give adequate error messages. There was also a difference noted in running the CORA-9 test case between different computers. Users also thought that some information should be provided on which subroutines in the code are CPU intensive.

<u>Experience: Physical Models.</u> The results reported by code users for the early interval were deemed to be reasonable up to the point of core melting. However, it was generally thought that results were unreasonable after core degradation for:

1. Cladding temperature oxidation fraction,
2. Control-rod relocation,
3. ZrO_2 dissolution, and
4. Metallurgical interaction of core material.

Results were deemed unsatisfactory following reflood, as well. It was noted that reflooding of the hot core does not result in additional hydrogen generation. The radiation heat transfer between SCDAP and RELAP components was thought to be a problem, and ballooning is not adequately predicted in older versions employed by some users. Users reported a poor prediction of material interactions, particularly in PHEBUS B9+. Users also found that limits for applicability of the code are not explicitly stated in any of the code documentation and that the theoretical basis for phenomenological models provided in the code documentation does not always agree with what is actually in the code.

Comments on Input. Users expressed the opinion that redundant input that was sometimes required for the SCDAP/RELAP5 input deck is a source of potential errors and should be eliminated. They felt that the SCDAP input manual was not very helpful, while the RELAP manual was quite good. In fact, users often had to resort to investigating the source code to clarify the input requirements. In general, users found that they could not set up a complete input deck without consulting with the code developers (INEL). They also found that input error checking was nonexistent for the SCDAP portions, while it was quite good for the RELAP components. Users also thought that there should be more information provided on steady-state calculations. They would like to see SCDAP input restructured and allow option-to-change SCDAP input on restart.

Comments on Transportability. As shown in Table 5-III, the SCDAP/RELAP5 code has been used on a variety of main frames and workstations, including the CRAY Y-MP, CRAY-2, CRAY X-MP, Convex-220, Sun, IBM-RISC, Apollo-DN, and MicroVAX. In general, users found it relatively easy to install, although there appeared to be machine dependencies with some differences noted on sample problems when run on different computers. It was thought that some nonstandard coding in RELAP5 sometimes presented a problem on porting the code to different machines.

General Comments. The users thought that code-design objectives expressed in the survey (as provided by the NRC) were good but not always followed. They cited several undocumented features and unimplemented features that caused confusion and misapplication of the code. It was thought that test cases should be reviewed more carefully and results examined closely before releasing a new version of the code. Users agreed that SCDAP documentation should be enhanced, QA was insufficient, too many undefined versions of the code were sent out, and the documentation needs QA, as well.

Users expressed concern that because of the excessive computing time required to perform fission-product calculations, there was almost no experience simulating the fission-product behavior. Users said that the integration of SCDAP with RELAP is far from complete and that the code is not very user friendly. It was deemed typical of a research tool with medium difficulty encountered to apply it successfully. Users also said that they would like to see BWR models provided in the code.

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6. RECOMMENDATIONS

The Committee recognizes that a significant effort has gone into the development of the SCDAP/RELAP5 code, and some of the code-design objectives and targeted applications have been met. In addition, the NRC has directed that work be performed to improve the code even further. At the same time, the Committee also recognizes that additional work is needed if all the phenomenological models are to become technically adequate and if all of the design objectives and targeted applications are to be met.

This section provides major recommendations the Committee deemed important in the following areas:

1. Improving phenomenological models,
2. Meeting code-design objectives,
3. Meeting code-targeted applications,
4. Addressing other findings, and
5. Making additional recommendations.

The Committee sought to include recommendations they believed would have an important impact on code improvement. To that end, the Committee worked to identify the dominant PWR and BWR severe-accident phenomena; this effort was based on the work documented in Appendix C of the MELCOR Peer Review report (Ref. 6-1). This effort is documented in Appendix C. The Committee also developed a method for ranking the dominant severe-accident phenomena (Hierarchy-by-Interval approach), along with the associated code models. This ranking process allowed the Committee to arrive at a consensus on technical adequacy of phenomenological models. The ranking also served to identify the important phenomenological models (based on their potential effect on source term and core damage) that needed additional improvement. The Hierarchy-by-Interval approach to ranking phenomena and code models is described in Appendix D.

6.1. Improving Phenomenological Models

6.1.1. Fuel-Rod Liquefaction, Flow, and Solidification Models Should Be Improved (Sections 2.9 and 2.13)

This is one of the most important models impacting the behavior of core material during severe-accident conditions, particularly during the melt relocation and slump time interval, and it suffers from several modeling and phenomenological inadequacies listed in Appendix E, Section E.9. In general, many of the assumptions stated and the models chosen are without adequate physical basis and are sometimes arbitrary. However, the present knowledge for many of the phenomena in this area is questionable, and a better understanding of the basic phenomena may be necessary before a truly technically adequate code model can be developed. The Committee acknowledges that there are ongoing activities directed by the NRC to address this deficiency. For example, in DCH analyses, radial spreading has not been accounted for previously, so code developers have been directed to add an option forcing radial spreading of liquefied material into adjacent channels. Also, problems with poor prediction of rivulets and free-falling droplets for experiment analyses are being addressed as part of ongoing model improvement (under FIN A6889).

6.1.2. Fission-Product Release, Transport, and Deposition Models Should Be Improved (Sections 2.10, 2.17, and 2.18)

The prediction of fission-product release is one of the most important issues for accident management studies. Overall, the fission-product modeling approach is excessively unbalanced. While modeling to predict the release from intact fuel is fairly detailed, the remaining models are for the most part highly empirical and somewhat outdated. This may be partially due to the fact that most of the phenomena are understood, while the state of knowledge for other fission product-related phenomena is questionable. The associated documentation has many typographical errors in the equations, and most of the support information is not readily accessible. Additional improvements in this area may not be warranted because the NRC is sponsoring the VICTORIA code for mechanistic source-term calculations.

6.1.3. Improvements Should Be Made to Control-Rod and Core-Structure Models (Sections 2.11 and 2.26)

The presence of absorber material can significantly affect the nature and timing of both PWR and BWR core-melt progression. Grid spacers can have an important effect on blockage formation. The control-rod models do not treat some important chemical/eutectic reactions involving their constituents. The Committee notes that work is currently under way to incorporate the influence of these interactions.

The grid-spacer model the Committee reviewed is parametric in nature. A new model (described in Section 2.26.2) that takes into account the zircaloy/Inconel eutectic reaction has recently been developed that appears to be first order, but cannot yet be formally categorized because of a lack of assessment against experiments. The state of knowledge in this area is good, so an adequate model can be developed.

6.1.4. Improvements Should Be Made to Enhance Simulation of Debris Heat Transfer, Fragmentation, and Quenching in the Core and Lower Plenum (Sections 2.13, 2.14, 2.15, and 2.25)

The major limitations of the lower-plenum debris heatup model include: (1) an inadequate description of the effective thermal conductivity model of the porous bed, (2) the lack of a model to predict buoyancy-induced, multidimensional flow in the bed and buoyancy-assisted melting of nonporous metallic and porous ceramic debris, (3) the lack of a model to describe the rise of vapor or migration of liquid in the debris bed, (4) the neglect of oxidation of the debris bed and no release of fission products in the debris bed, and (5) an inadequate description of gap conductance between the nonporous metallic debris layer and the vessel wall.

The Committee found that the state of knowledge of the phenomena associated with lower-plenum debris modeling was poor for the most part, and thus, the basic phenomena will need to be better understood before technically adequate code models can be developed. Therefore, the Committee recommends that, as a minimum, the modeling be made self-consistent. It should be noted that the NRC, under its ongoing activities to resolve code problems, plans to improve the debris-bed, heat-transfer models and add models to allow stratification of debris-bed layers and eutectic formation within a single debris volume.

In addition to the problems described for debris bed behavior, reflooding of the core after core degradation should be treated. In particular, a model should be introduced for the excess hydrogen production that may occur in such circumstances; this has accident management implications. The Committee recognizes, however, that the state of knowledge about the physical phenomena is questionable in this area, and additional knowledge may be required before a technically adequate code model can be developed. As a minimum, self-consistent models should be used in SCDAP/RELAP5.

6.1.5. Improvements Should Be Made to Enhance Simulation of Molten Pool Formation, Crust Behavior, and Convection in Molten Pools (Sections 2.13, 2.15, and 2.25)

See Sections 2.13, 2.15, and 2.25 for specific findings and identification of model deficiency. See Appendix E, Sections E.13, E.15, and E.25 for additional background and detail. Because of poor understanding of the physical processes, SCDAP/RELAP5 models, as a minimum, should be made self-consistent.

6.1.6. Improvements Should Be Made to Enhance Simulation of Heat Transfer to Lower Head and Vessel-Head Response (Sections 2.15, 2.16, 2.25, and 2.26)

See Sections 2.15, 2.16, 2.25 and 2.26 for specific findings and identification of model deficiency. See Appendix E, Sections E.15, E.16, E.25, and E.26 for additional background and detail.

6.2. Meeting Code-Design Objectives

6.2.1. Additional Assessments Should Be Performed

Additional assessments need to be performed for tests where expected uncertainties of key measured parameters are clearly identified. The expected uncertainties have not been clearly identified to date in either the TMI-2 data or data from other experiments. However, the Committee understands that work is currently under way to perform a systematic assessment of modeling uncertainties for the early intervals of an accident. This work will include an evaluation of the relative magnitudes of experimental and modeling uncertainties once it is completed.

6.2.2. The Code Should Be Made More Robust

Code users are generally unable to complete a calculation from start to finish for an entire accident sequence without code failure. Code robustness should be improved to allow complete calculations to be performed. Although code robustness is one of the code-design objectives and represents a major concern to code users, the Committee acknowledges that improving the code in this area does represent a significant challenge, especially considering the necessary complexity and interaction of the code models. The NRC has ongoing activities to improve code robustness as part of the CSARP, DCH, and other programs. These activities include, for example, expanding input error checking and diagnostic printout, as well as incorporating system level error trapping for SCDAP routines.

6.2.3. Noding Sensitivities Should Be Identified

Time and spatial nodalization studies should be performed. The Committee did not have documentation at the time of the review of any studies performed to investigate noding sensitivities either relating to timestep size or number of spatial volumes. It should be noted that the NRC has several ongoing activities to address the problems that cause time and spatial noding sensitivities. These activities include work in support of direct containment heating, water addition, and CSARP programs (e.g., time smoothing of the interface conditions between RELAP5 and SCDAP).

6.2.4. Code Documentation Should Continue to Be Improved

Any official code version released should be accompanied by an assessment report. A list should be provided of consistent models and existing assessment cases that code developers use for checking code before each version is released. More time should be spent documenting what is in the code and bringing the manual up to date. An NQA-1 configuration management system should be developed.

Code documentation should provide an up-to-date compendium on models and correlations that are used in SCDAP/RELAP5, identifying the basis for each model (pedigree), its implementation, applicability, and degree of assessment. The material provided to the Committee has not always been consistent with what is in the code (particularly the need for an accurate description of models and correlations), and some of the models described were obsolete.

Changes to the documentation, particularly for input data, should be placed on line under configuration control so that the code version and description are closely linked. Also, the Committee suggests using a "living document" approach where "change pages" to existing code documentation are provided that reflect the state of a newly released code version. It is recognized that this effort will not directly improve the code, per se, but will facilitate code assessment and the identification of needs for future improvements.

6.3. Meeting Code-Targeted Applications

6.3.1. More Full-Plant Simulations Should Be Performed

The Committee did not have enough of the necessary information to assess the code's success in meeting its targeted applications. Both PWR and BWR full-plant simulations should be performed with the simulations proceeding through all four intervals of the severe accident (see Section 3 for the Committee's definition of the four intervals) and including parametric studies to determine spatial and timestep noding sensitivities (see Section 6.2.3).

6.3.2. SCDAP/RELAP5 Should Be Used to Analyze Westinghouse Natural Circulation Tests

The Committee is aware of a recent report (Ref. 6-2) that compared RELAP5/MOD3 predictions of natural circulation for a steady-state SF_6 test. However, there has been no direct simulation providing acceptable assessment of the code capability to predict transient natural circulation phenomena after the accident has progressed into a degraded core thermal-hydraulic situation in a mixed-convection regime. Targeted Application #3 (see Section 4) specifically states that this should be a code application for specific technical issues.

6.3.3. SCDAP/RELAP5 Modelers Should Provide a Consistent Model of Melt Transfer and Energy Transfer to the Lower-Plenum Water and Lower-Head Structure

This capability is important for the code to be employed for Targeted Application #3, where one of the specific technical issues identified for study with SCDAP/RELAP5 is lower-head-failure analysis.

6.4. Addressing Other Findings

6.4.1. The Code Output Should Be Streamlined

A list of output variables should be reassessed for severe-accident needs. A plotting package should also be provided with the SCDAP/RELAP5 code (e.g., a package similar to the present Nuclear Plant Analyzer, but one that provides more detailed information).

6.4.2. The Code Input Should Be Streamlined

The parts of the code differ markedly in their input requirements. The RELAP part is well structured, with extensive error checking, while the SCDAP part is less well structured, and the format is different from that in the RELAP part. Also, input to the SCDAP and COUPLE parts cannot be changed on restart. The input for SCDAP and COUPLE parts of the code should therefore be made consistent with that for the RELAP part, and the same facilities for error checking and input change on restart should be introduced.

6.4.3. The Code Assessment Goals Need Clarification

The code assessment goals should be clearly identified, and success criteria (similar to those specified with design objectives and targeted applications) should be determined. Currently, the goals do not appear to be well defined, and it is not clear when an assessment has successfully accomplished its objective.

6.5. Additional Recommendations

6.5.1. Future Code Development as It Relates to SCDAP/RELAP5 Should Be Peer Reviewed

The SCDAP/RELAP5 Peer Review has provided a measure of the degree of completion of the code relative to prespecified design objectives and targeted applications. It is the Committee's opinion that future code development decisions would benefit from a periodic independent peer review of the code after major code modifications have been completed. This periodic review, in the opinion of the Committee, would enhance any additional code development effort. By measuring the degree of completion and the overall technical adequacy of the code at the time of a review, an independent perspective would be available that would assist the NRC in deciding what further code development, if any, is needed.

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Acknowledgments

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APPENDIX A

Members of the SCDAP/RELAP5 Peer Review Committee

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Michael L. Corradini is a professor of nuclear engineering and engineering physics, College of Engineering, University of Wisconsin–Madison. During the last 12 years, he has been engaged in research related to nuclear and industrial safety, with specific emphasis on subjects involving multiphase flow and heat/mass transfer. His current research focuses on vapor-explosion phenomena, jet-spray breakup, and mixing dynamics, as well as heat/mass transfer and chemical reactions involved in molten-core-concrete interactions. He is a Fellow of the American Nuclear Society and was a recipient of the 1990 Young Members Engineering Achievement Award. He serves as a consultant for the NRC Advisory Committee on Reactor Safeguards, as well as for the Department of Energy (DOE) National Laboratories (Los Alamos National Laboratory, Oak Ridge National Laboratory, Idaho National Engineering Laboratory, and Brookhaven National Laboratory), and participates in research with the national and international sponsors. Professor Corradini obtained his B.S. degree in mechanical engineering from Marquette University in 1975 and his M.S. and Ph.D. from the Massachusetts Institute of Technology in 1978. He was a member of the technical staff at Sandia National Laboratories for three years before joining the faculty of the University of Wisconsin–Madison in nuclear engineering.

V. K. Dhir

Vijay K. Dhir is a professor of engineering and applied science, Mechanical Aerospace and Nuclear Engineering Department, School of Engineering and Applied Science, University of California at Los Angeles (UCLA). During the past 17 years, he has done both basic and applied research in the thermal sciences and in energy conversion systems. His basic research is on the phenomenological studies of phase-change heat and mass transfer. This research includes experimental and analytical investigations of pool and forced flow boiling under saturated and subcooled conditions, two-phase flow in porous media, film condensation, simultaneous melting and condensation under steady-state and transient conditions, and evaporation. In the applied areas, he has worked on safety and thermal hydraulics of fission and fusion nuclear power reactors. The studies have included reflood heat transfer, degraded core heat transfer and fluid flow phenomenology, core-concrete interactions, natural convection and stratification in liquid-metal-cooled reactors, and melting, freezing, and plugging of coolant channels in transient overpower accidents. He has served on various DOE review panels; has been a consultant to Atomics International, Canoga Park, in support of the design efforts for a pool-type, fast-breeder reactor with inherent safety characteristics; and has served as a member of the MELCOR Peer Review Committee. He has also been a consultant to the National Bureau of Standards; Science Applications International Corporation; Battelle Northwest Laboratories; EG&G, Idaho, Inc.; General Electric; Electric Power Research Institute; and Pickard, Lowe and Garrick. Dr. Dhir obtained his Ph.D. in mechanical engineering from the University of Kentucky, Lexington, and has published more than 100 papers in various national and international journals and conference proceedings. At UCLA, he teaches courses in heat transfer, thermodynamics, and nuclear reactor thermal-hydraulic design.

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Tim Haste is a senior scientist in the Reactor Safety Studies Department, Safety and Performance Division, AEA Reactor Services, AEA Technology, Winfrith, United Kingdom. He is currently engaged in analysis of early-phase melt progression in PWR systems, is project manager for UK activities involving MELCOR and SCDAP/RELAP5, is collaborating actively with national laboratories in the US, France, and Germany, and has served as a member of the MELCOR Peer Review Committee. He is presently chairman of the UK SCDAP/RELAP5 User's Group. Before his appointment at Winfrith, he worked for 10 years at the Springfields Laboratories of AEA Technology, specializing in theoretical analysis of fuel performance in advanced gas-cooled-reactor and PWR systems under normal and design basis LOCA conditions and in thermophysical properties of reactor materials. One year of this period was spent as a visiting scientist at the OECD Halden Reactor Project, Norway. He was a coauthor of the UK Zircalloy Data Manual. Before working at Springfields, he researched into Doppler broadening in fast reactors at Harwell, UK. Dr. Haste maintains a general interest in the water reactor fuels area, acting as a referee for papers and reviews in international journals and conferences and contributing regularly in these areas. After graduating in theoretical physics from Cambridge University, UK, he obtained a Ph.D. in Nuclear Science from Oxford University, UK. He is a Fellow of the UK Institute of Mathematics, a member of the UK Institute of Physics (Chartered Physicist), and a member of the British Nuclear Energy Society.

T. J. Heames

Terry Heames is a senior engineer at Science Applications International Corporation and has over 20 years of experience in the reactor safety area. He is part of a long-term contract with Sandia National Laboratories to provide expert assistance in the nuclear safety research area. Mr. Heames is currently coordinating the development of the VICTORIA fission-product behavior code. He was a developer of the MELPROG water reactor melt progression code and of the SAS liquid metal reactor melt progression code. Mr. Heames maintains a general interest in melt progression and accident sequence phenomena by reviewing and contributing papers in that field. He holds an M.S. degree in mechanical engineering from Northwestern University.

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Rick Jenks is Committee Chair for the SCDAP/RELAP5 Peer Review Committee. He is a staff member at Los Alamos National Laboratory, Los Alamos, New Mexico, in the Terrestrial Reactor Technology Section within the Reactor Design and Analysis Group. Mr. Jenks has performed LWR thermal-hydraulic code assessments, conducted nuclear plant systems analyses, provided code-user support, and developed computer systems models. Previous experience includes BWR fuel engineering work for Westinghouse Electric Corporation. He holds an M.S. degree in nuclear engineering from Oregon State University.

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John E. Kelly is manager of the Accelerator Production of Tritium Project, Department 6414, Sandia National Laboratories, Albuquerque, New Mexico. His current interests are directed toward assessing the safety of neutron spallation technology. During the past 12 years, his principal activities have been in four areas related to reactor safety: thermal hydraulics, severe accidents, probabilistic risk assessment (PRA), and new production reactor safety. He has performed basic research in developing, assessing, and applying numerical methods to complex nuclear reactor thermal-hydraulic problems. In addition, he developed computer models for in-vessel melt progression analysis and was the principal investigator for a program that developed an integrated, best-estimate computer code for analyzing in-vessel core melt progression (MELPROG). He managed and directed the development of the MELCOR severe-accident computer code and participated in the Committee for Safety of Nuclear Installations (CSNI) working group that studied the TMI-2 accident. Dr. Kelly received his Ph.D. in nuclear engineering from the Massachusetts Institute of Technology in 1980. He has authored or coauthored over 30 articles and reports in the nuclear reactor safety area.

M. Khatib-Rahbar

Mohsen Khatib-Rahbar is president of Energy Research, Inc., in Rockville, Maryland. His research focuses on nuclear reactor safety and PRA. He has published extensively on severe accidents, source terms, methods for uncertainty analysis, consequence assessment, thermal hydraulics, and numerical methods. He has also developed computer models for simulation of thermal-hydraulic and neutronic transients in LWRs and liquid-metal, fast-breeder reactors. Before starting Energy Research, Inc., he was a staff scientist at Brookhaven National Laboratory, where he managed programs dealing with level 2/3 PRA reviews, verification, and benchmarking of the source-term code package (STCP) and MELCOR, source-term uncertainties (QUASAR), Zion/Draft NUREG-1150, and regulatory implications of new source terms. Dr. Khatib-Rahbar was a visiting scientist at Gesellschaft fuer Reaktorsicherheit in Germany (1982) and at the NRC Office of Nuclear Regulatory Research (1988–1989). He is currently a consultant to the US DOE, the NRC, the Swiss Federal Nuclear Safety Inspectorate, the International Atomic Energy Agency, the European Space Agency, several national laboratories and private organizations, and has served as a member of the MELCOR Peer Review Committee. He holds a Ph.D. in nuclear science and engineering from Cornell University.

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Raymond Viskanta is W.F.M. Goss Distinguished Professor of Engineering at Purdue University. His research focuses on heat transfer in buoyancy-driven flows, solid-liquid phase change, flow and heat transfer in porous media, radiative transfer in participating media, and combined conduction radiation, as well as convection-radiation heat transfer. Dr. Viskanta has been a Springer and Visiting Professor at the University of California, Berkeley, and a Guest Professor at the Technical University of Munich and at the Tokyo Institute of Technology. Before accepting employment with Purdue University, he was a mechanical engineer at Argonne National Laboratory in the Reactor Engineering Division. He has served as a consultant on heat transfer and thermal hydraulics to a number of national laboratories and industry, was a member of the Peer Review Panel on the Draft Reactor Risk Reference Document (NUREG-1150), and was a member of the MELCOR Peer Review Committee. He was a consultant to the PRA Subcommittee of the US Department of Energy Advisory Committee on Nuclear Facilities Safety. Dr. Viskanta is the technical editor of the *ASME Journal of Heat Transfer*, serves on advisory editorial boards of several journals, and is an author of over 300 journal publications on heat transfer, thermal sciences, and radiative transfer. He holds a Ph.D. in mechanical engineering from Purdue University and is a member of the National Academy of Engineering.

APPENDIX B

SCDAP/RELAP5 Document Database

This database is a reference to the documents that were used to perform the SCDAP/RELAP5 Peer Review. <u>The documents referred to in this database are not necessarily accessible to readers.</u>

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D9. Dr. C. M. Allison, "ACRS Comments on SCDAP/RELAP5 Documentation and Modeling - CMA -78-91," letter to Dr. F. Eltawila, US Nuclear Regulatory Commission (August 30, 1991).

D10. C. M. Allison and L. J. Siefken, "Application of SNL DEBRIS and SCDAP/RELAP5 to SNL Melt Progression Experiments - CMA-5-91 and LJS-1-91," letter to Dr. Y-S. Chen, US Nuclear Regulatory Commission (January 16, 1991).

D11. Dr. C. M. Allison and S. M. Modro, "Application of RELAP5 for SBWR Analysis - CMA-15-91, SMM-11-91," letter to Dr. L. M. Shotkin, US Nuclear Regulatory Commission (March 13, 1991).

D12. Dr. C. M. Allison, "Limitations for SCDAP/RELAP5 for BWR Accident Management Calculations - CMA-36-90," letter to Dr. Y-S. Chen, US Nuclear Regulatory Commission (June 26, 1990).

D13. Dr. C. M. Allison, "SCDAP/RELAP5 Analysis of CORA-16 - CMA-28-89," letter to Dr. T. J. Walker, Sandia National Laboratories (October 30, 1989).

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D15. Dr. C. M. Allison, "Reply to NRC Comments on 'Benefits of Proposed Validation and Application of Accident Code Tasks - CMA-1-89," letter to Dr. T. J. Walker, Sandia National Laboratories (January 4, 1989).

D16. Dr. C. M. Allison, "Submittal of a Paper to the Fourth International Topical Meeting on Nuclear Reactor Thermal -Hydraulics - CMA-2-89," letter to Dr. Klaus Wolfert, Gesellschaft fur Reaktorsicherheit (January 10, 1989).

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D18. INEL Staff, Handout 1, "Idaho National Engineering Laboratory Responses to Committee Questions," provided at SCDAP/RELAP5 Peer Review Committee Meeting #2, Idaho Falls, Idaho (April 7, 1992).

D19. D. L. Hagrman, Handout 2, "Response to Questions Asked Before April 1992," provided at SCDAP/RELAP5 Peer Review Committee Meeting #2, Idaho Falls, Idaho, draft internal memorandum to C. M. Allison, EG&G Idaho, Inc., DLH-03-92 (April 9, 1992).

D20. L. J. Siefken, "Transmittal of Draft Copy of Responses to Written Questions - LJS-06-92," letter to Committee members (April 21, 1992).

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E1. INEL Staff, SCDAP/RELAP5 Workshop notebook, workshop, UNIDAD Electrica S.A. (UNESA), Madrid, Spain (March 11–13, 1991).

E2. "Developmental Assessment Plan for a Common Version of SCDAP/RELAP5/MOD3 and RELAP5/MOD3," Rev. 1 (March 5, 1991) presentation handout contained in R. P. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #2 Handouts," internal memorandum to B. E. Boyack, N-12-92-274 (May 4, 1992).

E3. Richard P. Jenks, Los Alamos National Laboratory, "Final Draft MELCOR Peer Review Summary Report," letter to SCDAP/RELAP5 Peer Review Committee members with attached draft report N-12-91-851 (December 3, 1991).

E4. B. D. Reagan, Idaho National Engineering Laboratory, "Transmittal of Information - BDR-2-91," letter to R. P. Jenks, Los Alamos National Laboratory, with "(4) additional documents" including "marked-up" code manual table of contents showing references for additional background (December 12, 1991).

E5. B. D. Reagan, Idaho National Engineering Laboratory, "Transmittal of Peer Review Action Items - BDR-02-92," letter to R. P. Jenks, Los Alamos National Laboratory, with 15 action items that follow (January 20, 1992):

- Marked-up SCDAP/RELAP5 Code Manual Table of Contents;
- "SCDAP/RELAP5 Decay Heat Deposition Model," writeup by R. A. Dimenna, Idaho National Engineering Laboratory;
- "Table 9. Comparison of Calculated and Measured Releases of Noble Fission Gases";

- "Assessment of Core Damage Models in SCDAP/RELAP5 During OECD LOFT LP-FP-2," E. W. Coryell, Idaho National Engineering Laboratory;
- Copy of presentation overheads of Art Shieh's talk on "SCDAP/RELAP5 Numerics," given at SCDAP/RELAP5 Peer Review Committee Meeting #1, Idaho Falls, Idaho (November 18–21, 1991);
- R. A. Lorenz, J. L. Collins, and A. P. Malinauskas, "Fission Product Source Terms for the Light Water Reactor Loss-of-Coolant Accident," *Nuclear Technology* **46**, 404 (Mid-Dec. 1979);
- J. Rest, "An Improved Model for Fission Product Behavior in Nuclear Fuel Under Normal and Accident Conditions," *Journal of Nuclear Materials* **120**, 195 (1984);
- List of 16 species tracked in fission-product transport model;
- Information on how particle mobility is calculated;
- Information on Hinds and Fuchs citations;
- "Table 8. Comparison of Calculated and Measured Meltdown of Test Bundles";
- Information on Stainless-Steel 304 oxidation model;
- Information on release of tellurium from fuel rods; and
- Information on potential code error that restricts maximum blockage to 90% of flow area.

E6. B. D. Reagan, Idaho National Engineering Laboratory, "Response to Fax 1-22-92 Concerning Peer Review Materials - BDR - 92," letter to R. P. Jenks, Los Alamos National Laboratory, with tables showing various test-problem matrixes (January 22, 1992):

- "Table C.1 Idealized Transient Verification Problems";
- "Table C.2 Other Verification Problems"; and
- "Table Code-to-Data Comparisons for SCDAP/RELAP5/MOD3[7X]."

E7. "A6889 Work Summary - 1991," summary of Idaho National Engineering Laboratory work performed under US Nuclear Regulatory Commission-funded Project A6889: SCDAP/RELAP5 Code Maintenance.

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E10. J. Hyvarinen, Technical Research Centre of Finland, "SCDAP/RELAP5 Papers & Reports," letter to C. Allison, EG&G Idaho, Inc., with two memos related to experience in Finland with SCDAP/RELAP5 code, specifically related to implementation on their VAX/VMS computers (January 21, 1992).

E11. C. Allison, EG&G Idaho, Inc., "TMI-2 Analysis Exercise Final Report," part of an OECD TMI-2 Joint Task Group report included as second attachment of letter CMA-02-92 to Y. Chen, US Nuclear Regulatory Commission (January 30, 1992).

E12. "Joint Meeting of the Subcommittees on Decay Heat Removal and Thermal Hydraulic Phenomena," transcript of the US Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards subcommittee hearing (portions pertaining to SCDAP/RELAP5 only) held in Idaho Falls, Idaho (August 23, 1990).

E13. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 User Survey," fax of March 10, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of UK's response to subject survey and cover letter from J. Lillington, AEA Technology, Winfrith, to Y-S. Chen, US Nuclear Regulatory Commission (February 3, 1992).

E14. R. Jenks, Los Alamos National Laboratory,"SCDAP/RELAP5 User Survey," fax of March 10, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of Switzerland's response to subject survey and cover letter from S. L. Chan, Swiss Federal Nuclear Safety Inspectorate, to Y-S. Chen, US Nuclear Regulatory Commission (February 3, 1992).

E15. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 User Survey," fax of March 10, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of a Finland response to subject survey and cover letter from J. Hyvarinen, Finnish Center for Radiation and Nuclear Safety, Helsinki, to Y-S. Chen, US Nuclear Regulatory Commission (January 27, 1992).

E16. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 User Survey," fax of March 10, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of Germany's response to subject survey and cover letter from Dr. Weisshaupl, Siemens, to Y-S. Chen, US Nuclear Regulatory Commission (February 4, 1992).

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E19. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 User Survey," fax of March 10, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of a Finland response to subject survey and cover letter from R. Sairanen, Technical Research Centre of Finland, to Y-S. Chen, US Nuclear Regulatory Commission (January 24, 1992).

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E21. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 User Survey," fax of March 16, 1992, to G. Price, Idaho National Engineering Laboratory, with copy of Spain's response to subject survey and cover letter from J.A. Martinez, Consejo de Seguridad Nuclear, to Y-S. Chen, US Nuclear Regulatory Commission (January 30, 1992).

E22. "Heat Conduction Model for Core Components," copy of presentation viewgraphs provided as additional informative handout to Committee at Meeting #2 (April 7–9, 1992).

E23. P. B. Abramson, H. Komoriya, and J. H. Baron, "Severe Accident Data Base: Summary Report and User's Manual," International Technical Services, Inc., report ITS/NRC/91-8 (September 1991).

E24. Bernard Berthet, CEA/EG&G, fax to R. Jenks, Los Alamos National Laboratory, with information on ICARE, CATHARE-2, and TRAP-FRANCE codes, as well as other packages and a brief description of PHEBUS facility (April 13, 1992).

E25. R. Jenks, Los Alamos National Laboratory, "Documentation Related to Code Capabilities," letter to SCDAP/RELAP5 Peer Review Committee members, N-12-92-447, July 29, 1992, including the following documents as attachments:

- D. A. Brownson, Idaho National Engineering Laboratory, "Selection of the SCDAP/RELAP5 Code for Severe Accident Analyses - DAB-12-92," letter to Y. S. Chen, US Nuclear Regulatory Commission (May 19, 1992);

- B. W. Sheron, NRC/RES, "Response to the ACRS Comments on SCDAP/RELAP5 Documentation," memorandum to R. F. Fraley, Advisory Committee on Reactor Safeguards (April 22, 1992); and

- L. J. Siefken, Idaho National Engineering Laboratory, "Transmittal of Information on Debris and Crucible Models in the SCDAP/RELAP5 Code - LJS-4-90," letter to Y. S. Chen, US Nuclear Regulatory Commission (November 16, 1990).

Committee Documents and Findings

F1. R. P. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #1 Handouts," internal memorandum to B. E. Boyack, N-12-91-856 (December 6, 1991).

F2. R. Jenks, Los Alamos National Laboratory, "Final Draft Meeting #1 Minutes - Related Materials," letter to Committee members, N-12-91-861 (December 12, 1991).

F3. R. Viskanta, Purdue University, "Questions Concerning SR5 to Be Addressed by INEL," letter to R. Jenks, Los Alamos National Laboratory, with 26 questions (January 6, 1992).

F4. R. Jenks, Los Alamos National Laboratory, "Additional Post Meeting #1 Information," letter to Committee members, N-12-92-29 (January 15, 1992).

F5. R. Jenks, Los Alamos National Laboratory, "Questions from Terry Heames," letter to C. Allison, EG&G Idaho, Inc., with three questions to be addressed by Idaho National Engineering Laboratory, N-12-92-52 (January 29, 1992).

F6. R. Jenks, Los Alamos National Laboratory, "Questions from Tim Haste," letter to C. Allison, EG&G Idaho, Inc., with three questions to be addressed by Idaho National Engineering Laboratory, N-12-92-67 (January 31, 1992).

F7. T. Haste, AEA Technology, Winfrith, "SCDAP/RELAP5 Peer Review: Comments on Meeting #1 and Related Matters," letter to R. Jenks, Los Alamos National Laboratory, Ref: JMB/TJH/Jenks2.14 (February 14, 1992).

F8. R. Jenks, Los Alamos National Laboratory, "Questions from Vijay Dhir," letter to C. Allison, EG&G Idaho, Inc., with 20 pages of questions and comments to be addressed by Idaho National Engineering Laboratory, N-12-92-114 (February 24, 1992).

F9. M. Khatib-Rahbar, Energy Research, Inc., "Preliminary Questions," fax to C. Allison, EG&G Idaho, Inc. (February 3, 1992).

F10. J. Kelly, Sandia National Laboratories, letter to C. Allison, EG&G Idaho, Inc., with request for additional information in three areas (January 30, 1992).

F11. R. Jenks, Los Alamos National Laboratory, "Notice: Peer Review Committee Meeting #2," letter to Committee with meeting notice, milestones, document database, and questions for Idaho National Engineering Laboratory from Committee members M. Khatib-Rahbar, T. Haste, T. Heames, and R. Viskanta, N-12-92-65 (February 3, 1992).

F12. R. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #2 Handouts," internal memorandum to B. E. Boyack, N-12-92-274 (May 4, 1992).

F13. R. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #3 Handouts," internal memorandum to B. E. Boyack, N-12-92-436 (July 25, 1992).

F14. Richard P. Jenks, Los Alamos National Laboratory, "CSARP Materials Related to SCDAP/RELAP5," letter to members of the SCDAP/RELAP5 Peer Review Committee, N-12-92-310 (May 20, 1992).

F15. R. Jenks, Los Alamos National Laboratory, "Final Draft Meeting #2 Minutes - Related Materials," letter to Committee members, N-12-92-275 (May 26, 1992).

F16. R. Jenks, Los Alamos National Laboratory, "Meeting #3 Summary," letter to Committee members, N-12-92-441 (July 21, 1992).

F17. R. Jenks, Los Alamos National Laboratory, "Meeting #4 Summary," letter to Committee members, N-12-92-533 (September 24, 1992).

F18. R. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #4 Handouts," internal memorandum to B. E. Boyack, N-12-92-522 (September 18, 1992).

F19. R. Jenks, Los Alamos National Laboratory, "SCDAP/RELAP5 Independent Peer Review Committee Meeting #5 - Summary," letter to Committee members, N-12-92-644 (December 4, 1992).

F20. R. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #5 Handouts," internal memorandum to B. E. Boyack, N-12-92-647 (December 8, 1992).

APPENDIX C

Decomposition of Dominant Phenomena

This appendix provides a list of dominant physical phenomena against which the existence, adequacy, and, when possible, fidelity of each SCDAP/RELAP5 model are assessed.

On a generic basis, the various top-level physical phenomena contributing to each phase of severe-accident progression are delineated for both BWRs and PWRs. The importance of individual phenomenon will vary depending on the specific accident sequence under consideration and the intended application.

The SCDAP/RELAP5 code is expected to be applicable to a wide spectrum of severe-accident conditions, including:

1. High- and low- [with respect to the reactor coolant system (RCS)] pressure sequences,

2. Scenarios leading to early [emergency core-cooling system (ECCS) fails early] and late (ECCS fails late) initiation of core degradation, and

3. Recoverable accidents.

The code should provide reasonable estimates of severe-accident behavior under these diverse accident conditions. (Ref. C-1).

Typically, severe-accident analyses are performed to better understand the behavior of plant and containment systems during postulated accident conditions. These studies are often conducted in support of PRAs or as added information for regulatory decision making (i.e., evaluation of potential severe-accident management strategies). As part of these studies, computer codes are exercised to evaluate key accident signatures, including some of the following (limited to in-vessel phase only):

1. Timing of key events (core uncovery, lower-plenum dryout, vessel breach, containment failure, etc.),

2. Important fission-product attributes (release from fuel, retention within RCSs, retention in pools, etc.),

3. Temperatures of RCS structures (lower head, hot leg, steam-generator tubes, etc.),

4. RCS pressure before vessel breach,

5. Mode and location of RCS failure (bottom head, hot leg, steam generator tubes, etc.),

6. Quantity of rate of hydrogen generation (in-vessel and ex-vessel), and

7. Core-debris quantity, composition, temperature, and rate of ejection into containment.

The decomposition proposed here is based on the premise that a complete mechanistic analysis must portray important phenomenological processes during the in-vessel phase of accidents for the following distinct time intervals.

Interval 1: Initial Transient, Depletion and Heatup Interval (Before Core Damage; $T_{exit} \leq T_{saturation}$)

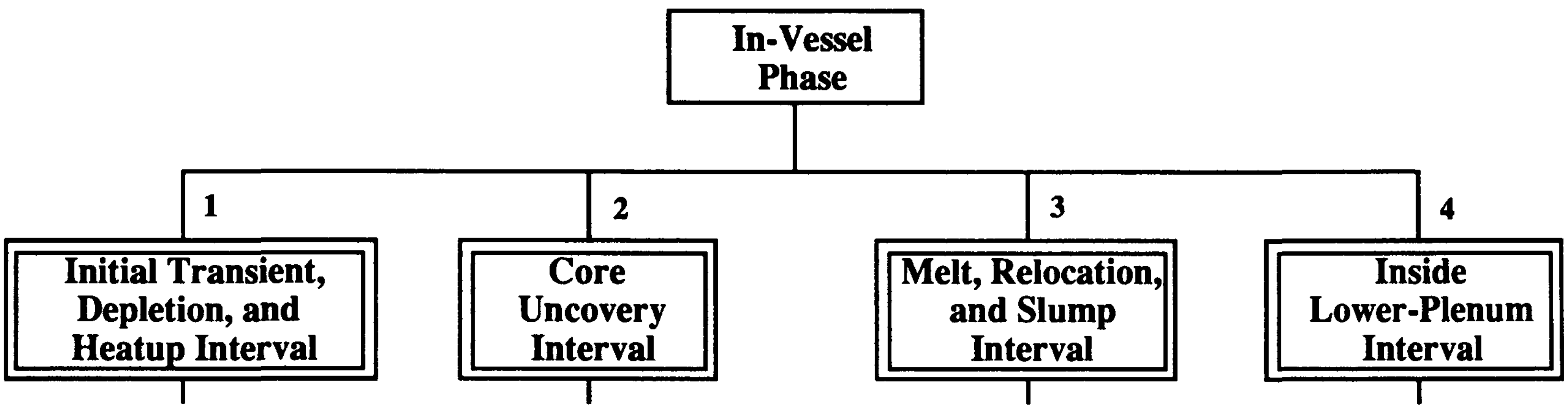
Interval 2: Core Uncovery Interval (Intact Geometry; T<1500 K)

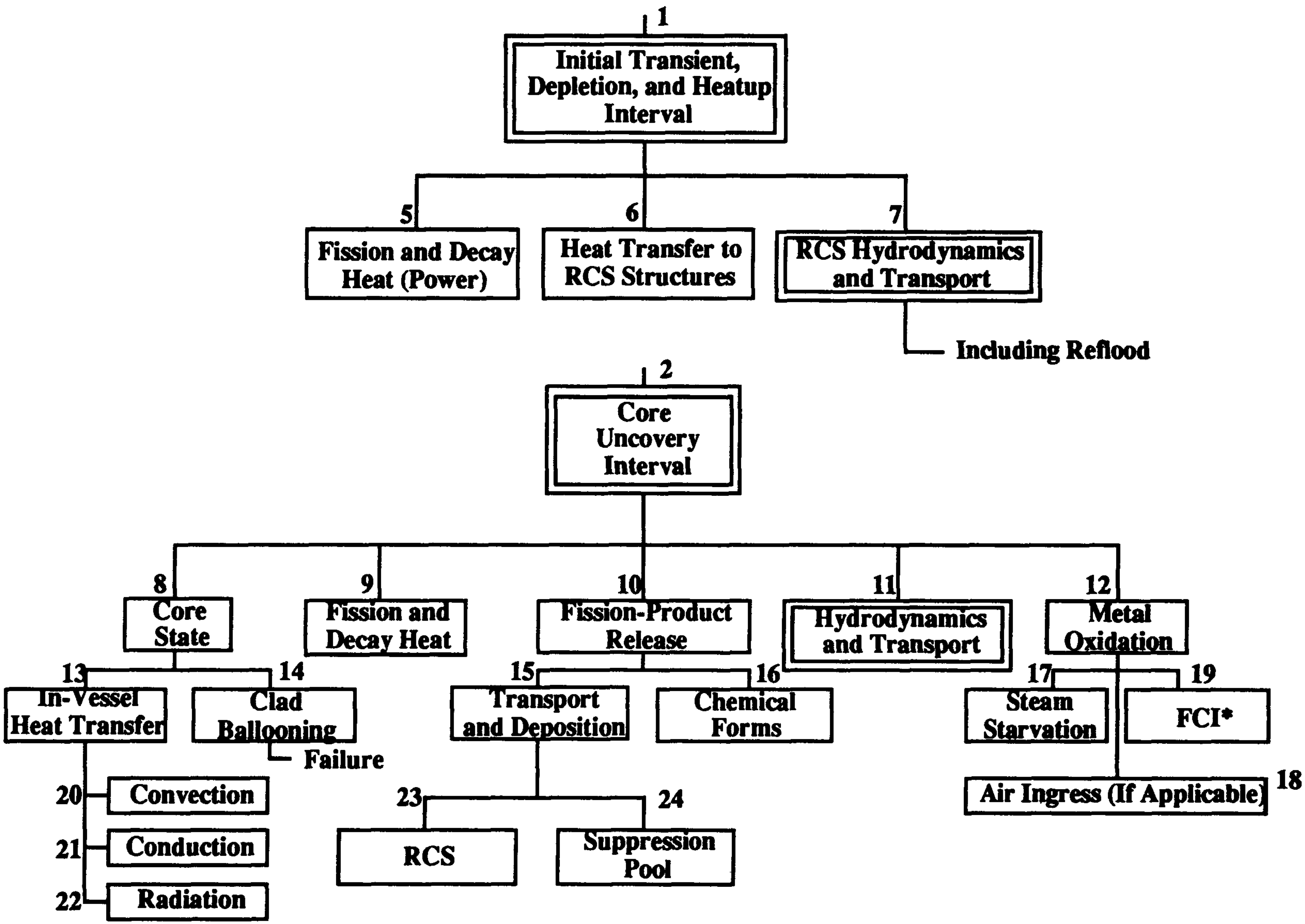
Interval 3: Melt Relocation and Slump Interval (Substantial Damage; T>1500 K)

Interval 4: Core-Debris Material Inside the Lower-Plenum Interval (Late In-Vessel Phase)

For each interval, key phenomenological issues impacting the evolution of the accident sequence are delineated. For the process to remain tractable, detailed subissues resulting from higher-order phenomena associated with the interaction of various physical and chemical processes are intentionally not shown. This should not mean that the dominance of some of these phenomena are to be ignored as part of SCDAP/RELAP 5 assessment. It is assumed that the individual Committee members cognizant of various phenomenological models will recognize these and address them as part of their review.

PART 1: BWR SEVERE-ACCIDENT DOMINANT PHENOMENA





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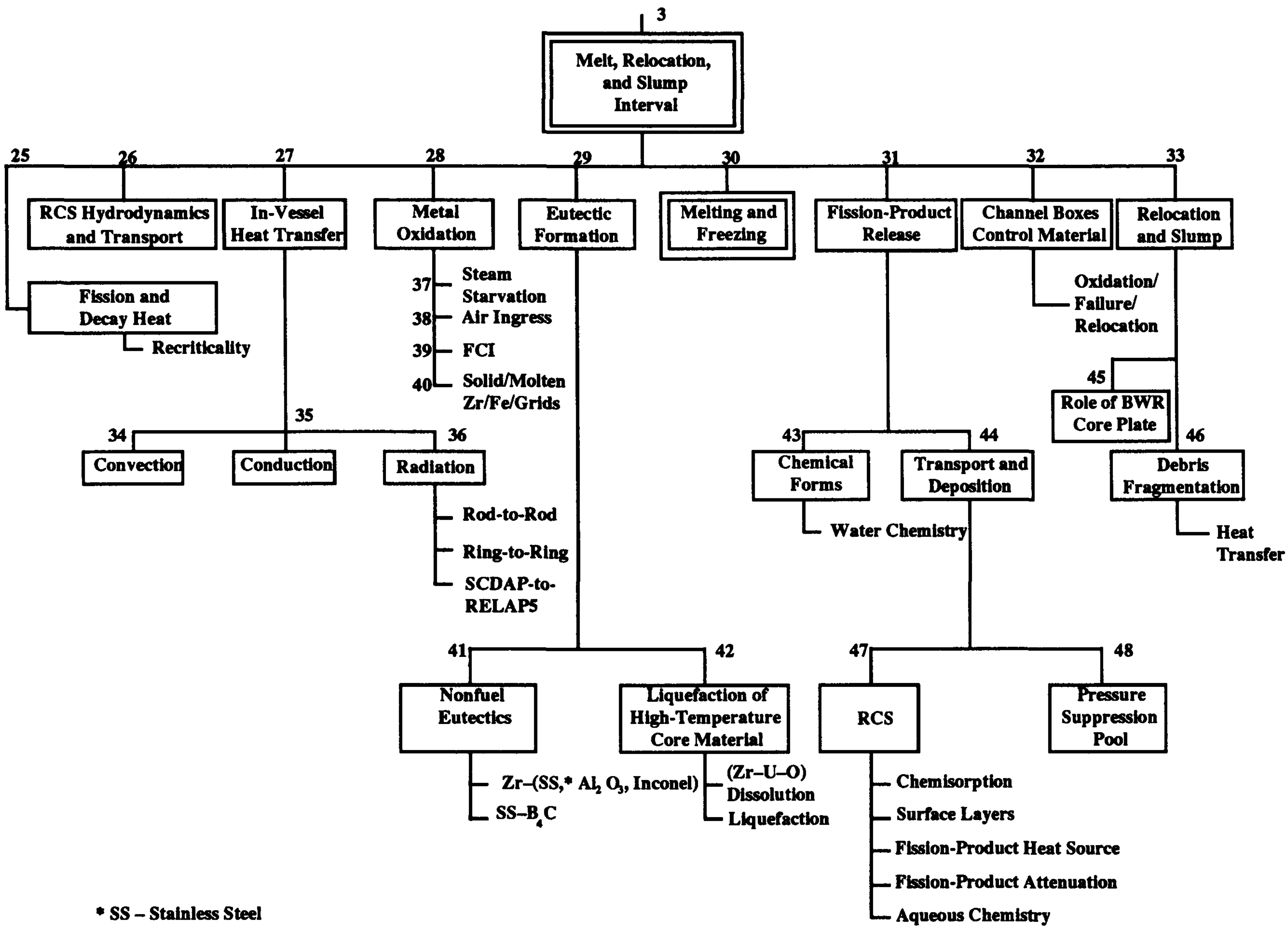
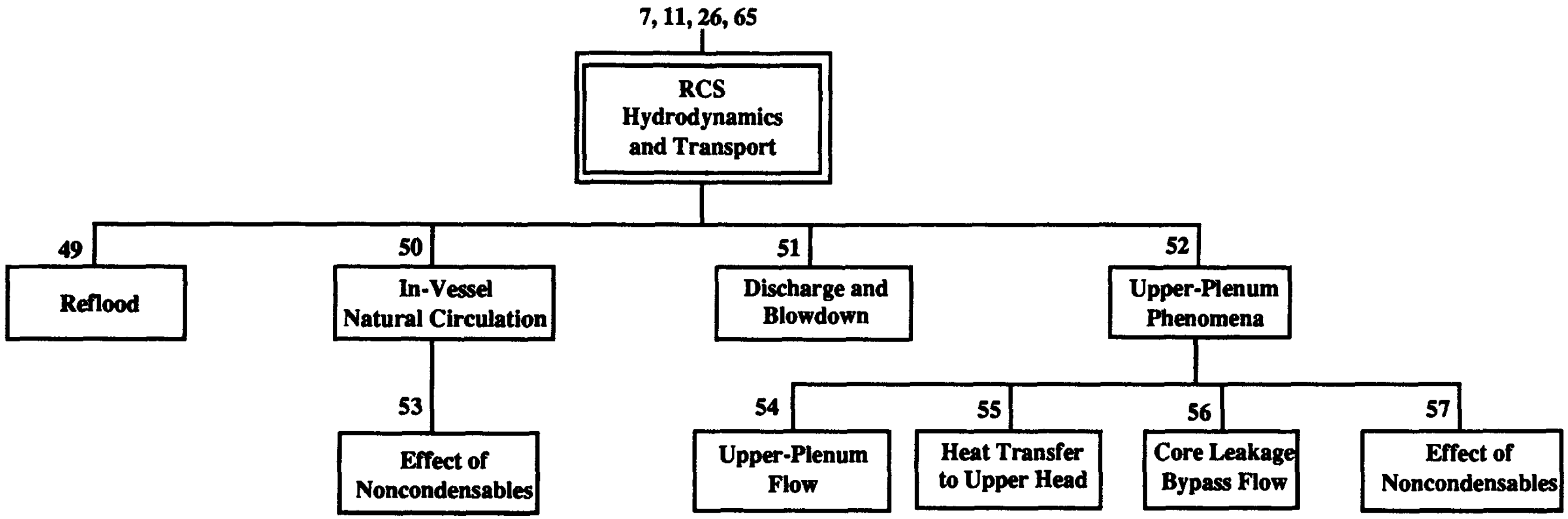
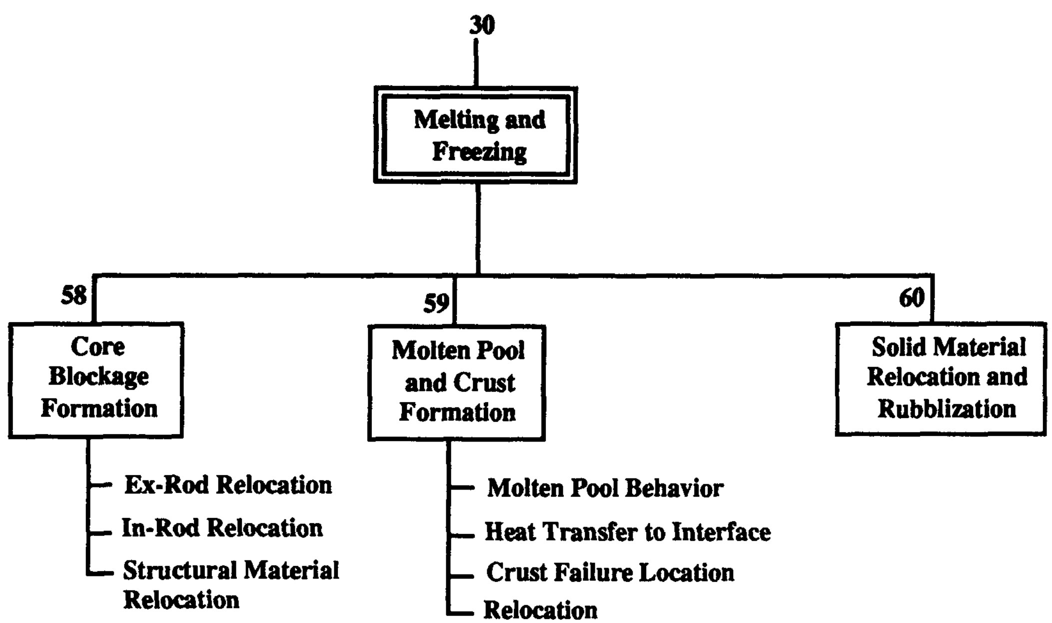
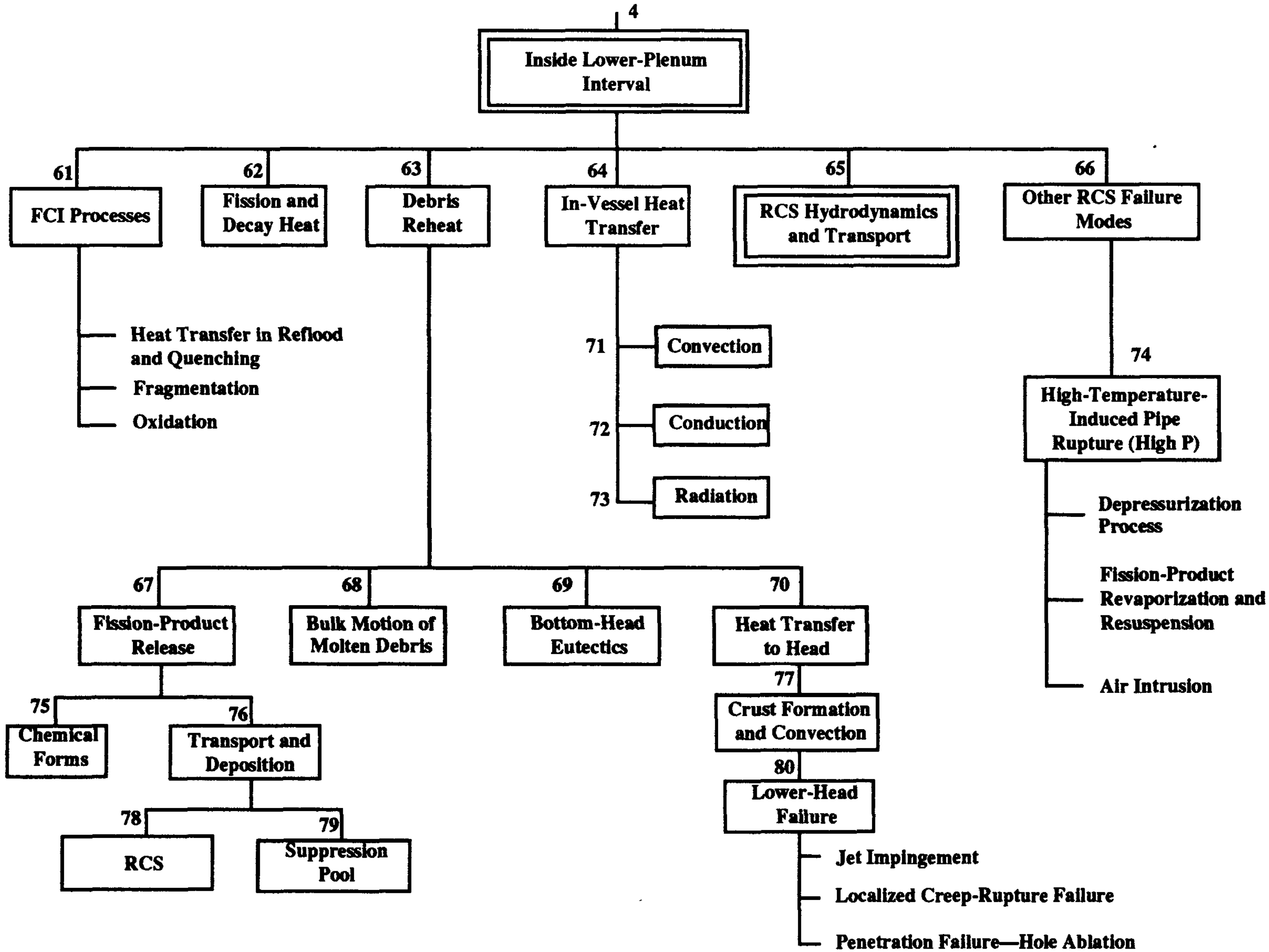


* SS – Stainless Steel


7, 11, 26, 65
RCS Hydrodynamics and Transport
49
Reflood
50
In-Vessel Natural Circulation
51
Discharge and Blowdown
52
Upper-Plenum Phenomena
53
Effect of Noncondensables
54
Upper-Plenum Flow
55
Heat Transfer to Upper Head
56
Core Leakage Bypass Flow
57
Effect of Noncondensables


30
Melting and Freezing
58
Core Blockage Formation
Ex-Rod Relocation
In-Rod Relocation
Structural Material Relocation
59
Molten Pool and Crust Formation
Molten Pool Behavior
Heat Transfer to Interface
Crust Failure Location
Relocation
60
Solid Material Relocation and Rubblization


4
Inside Lower-Plenum Interval
61
FCI Processes
62
Fission and Decay Heat
63
Debris Reheat
64
In-Vessel Heat Transfer
65
RCS Hydrodynamics and Transport
66
Other RCS Failure Modes
Heat Transfer in Reflood and Quenching
Fragmentation
Oxidation
71
Convection
72
Conduction
73
Radiation
74
High-Temperature-Induced Pipe Rupture (High P)
Depressurization Process
Fission-Product Revaporization and Resuspension
Air Intrusion
67
Fission-Product Release
68
Bulk Motion of Molten Debris
69
Bottom-Head Eutectics
70
Heat Transfer to Head
77
Crust Formation and Convection
80
Lower-Head Failure
Jet Impingement
Localized Creep-Rupture Failure
Penetration Failure—Hole Ablation
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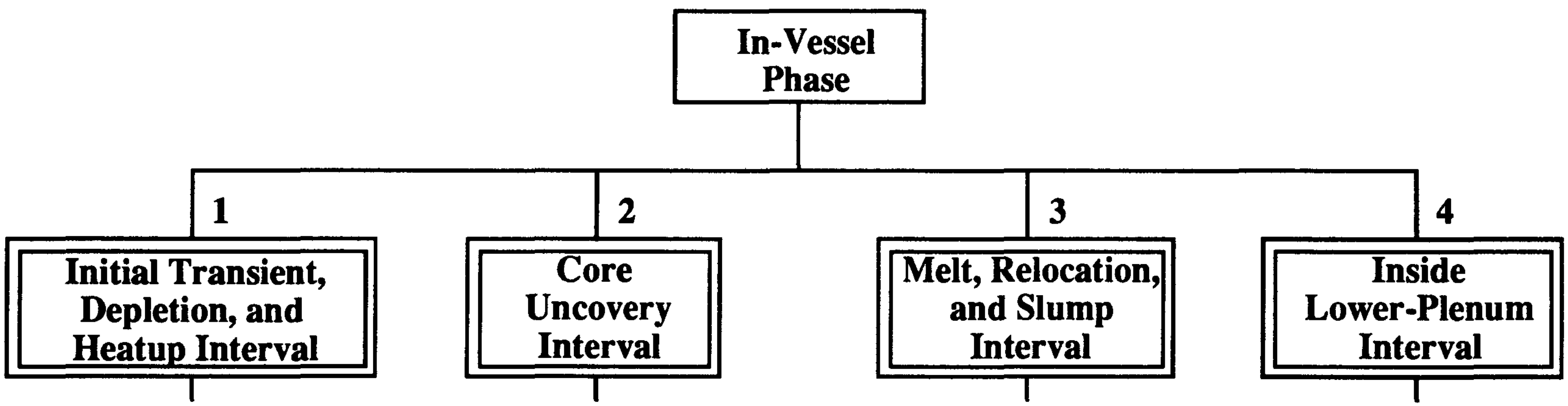
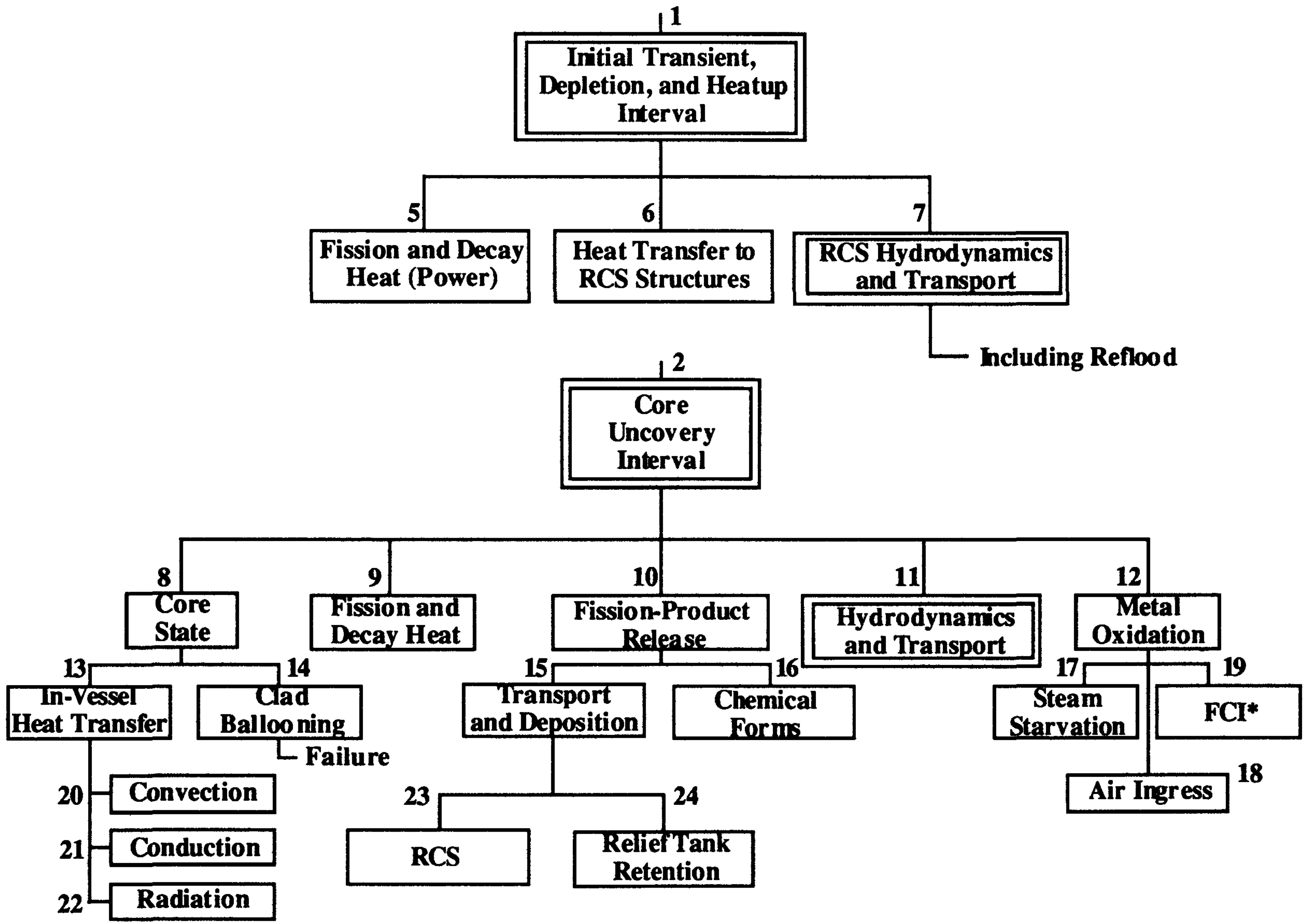
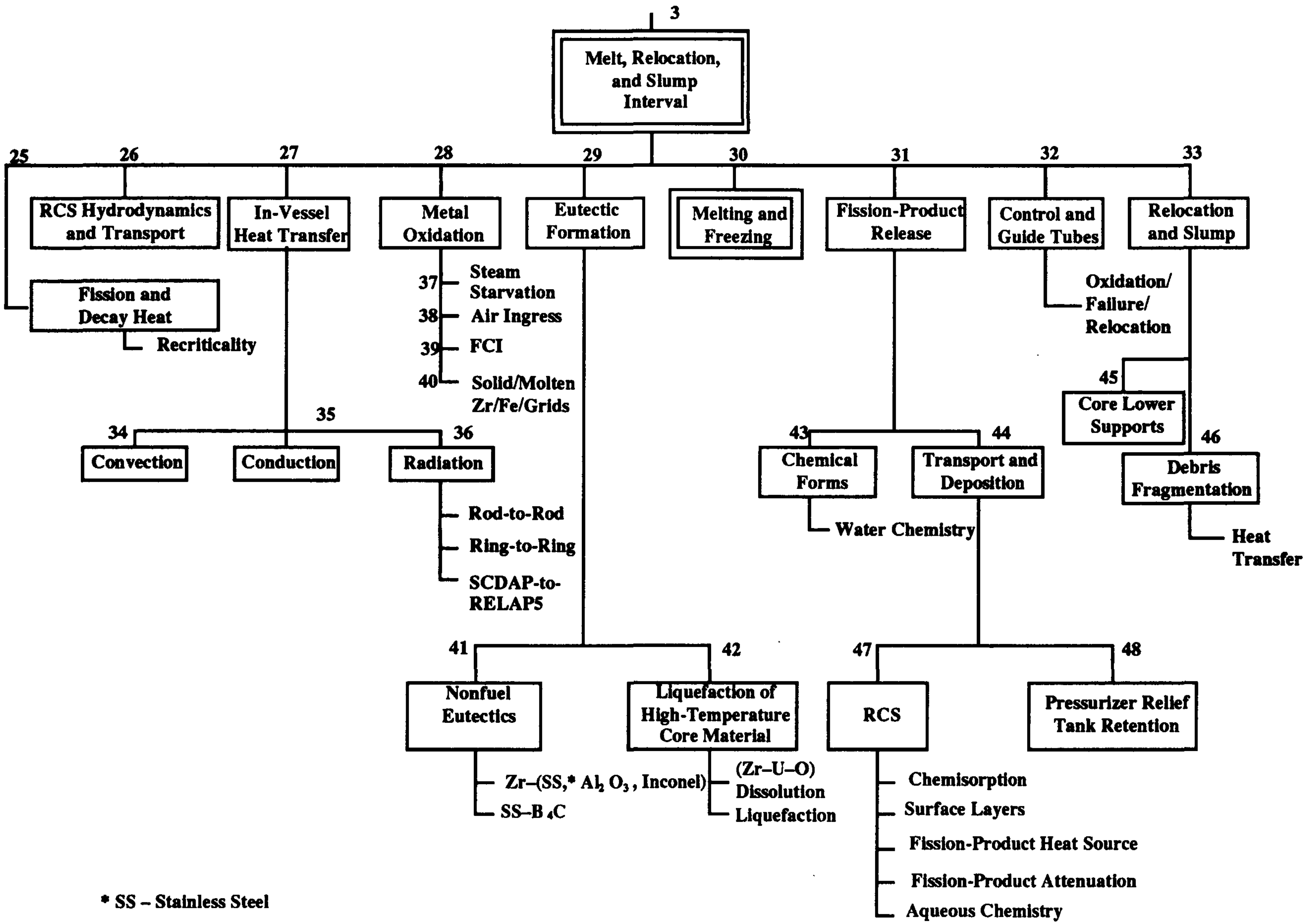
Chemical Forms
76
Transport and Deposition
78
RCS
79
Suppression Pool

PART II: PWR SEVERE-ACCIDENT DOMINANT PHENOMENA

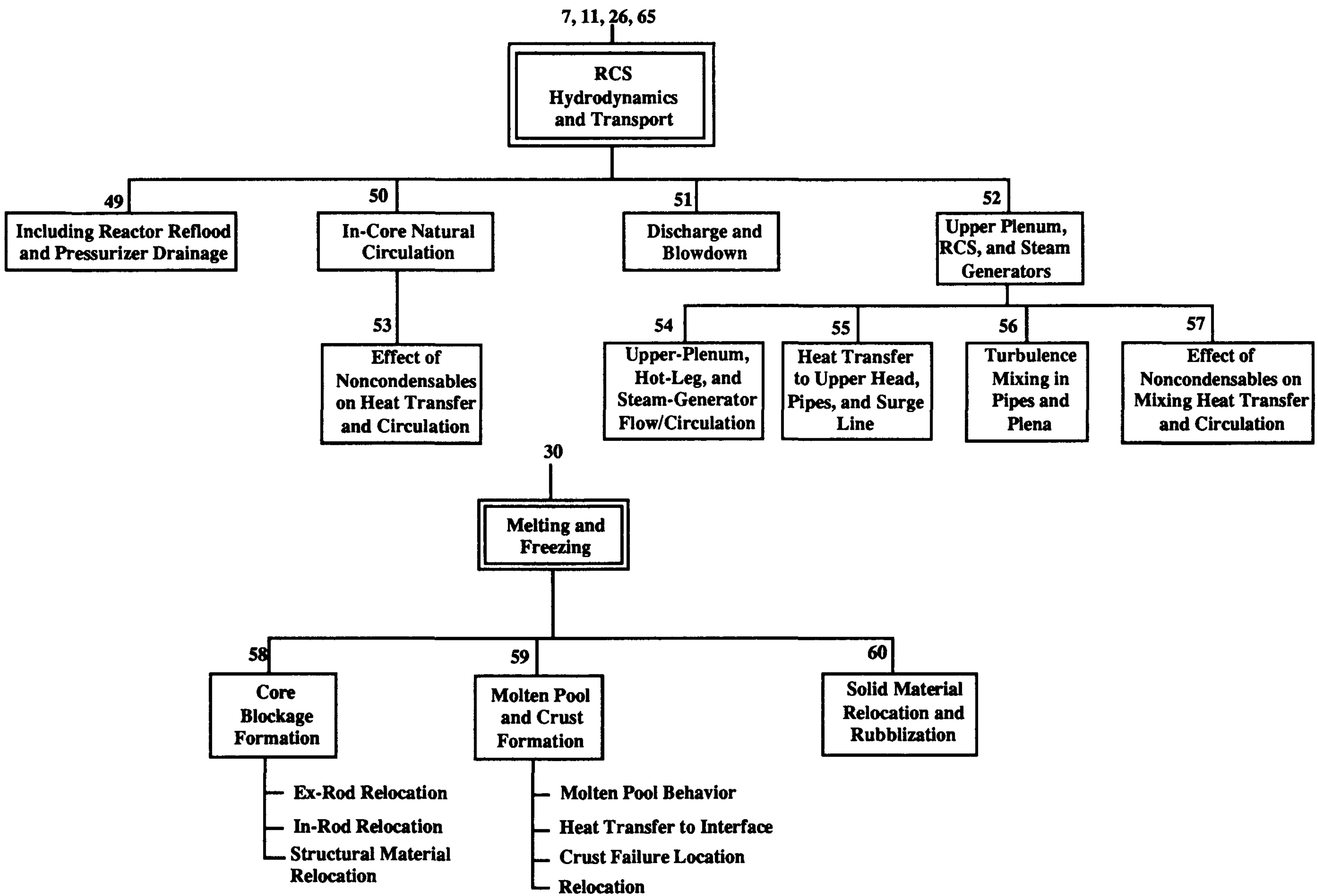




* Fuel-Coolant Interaction



* SS – Stainless Steel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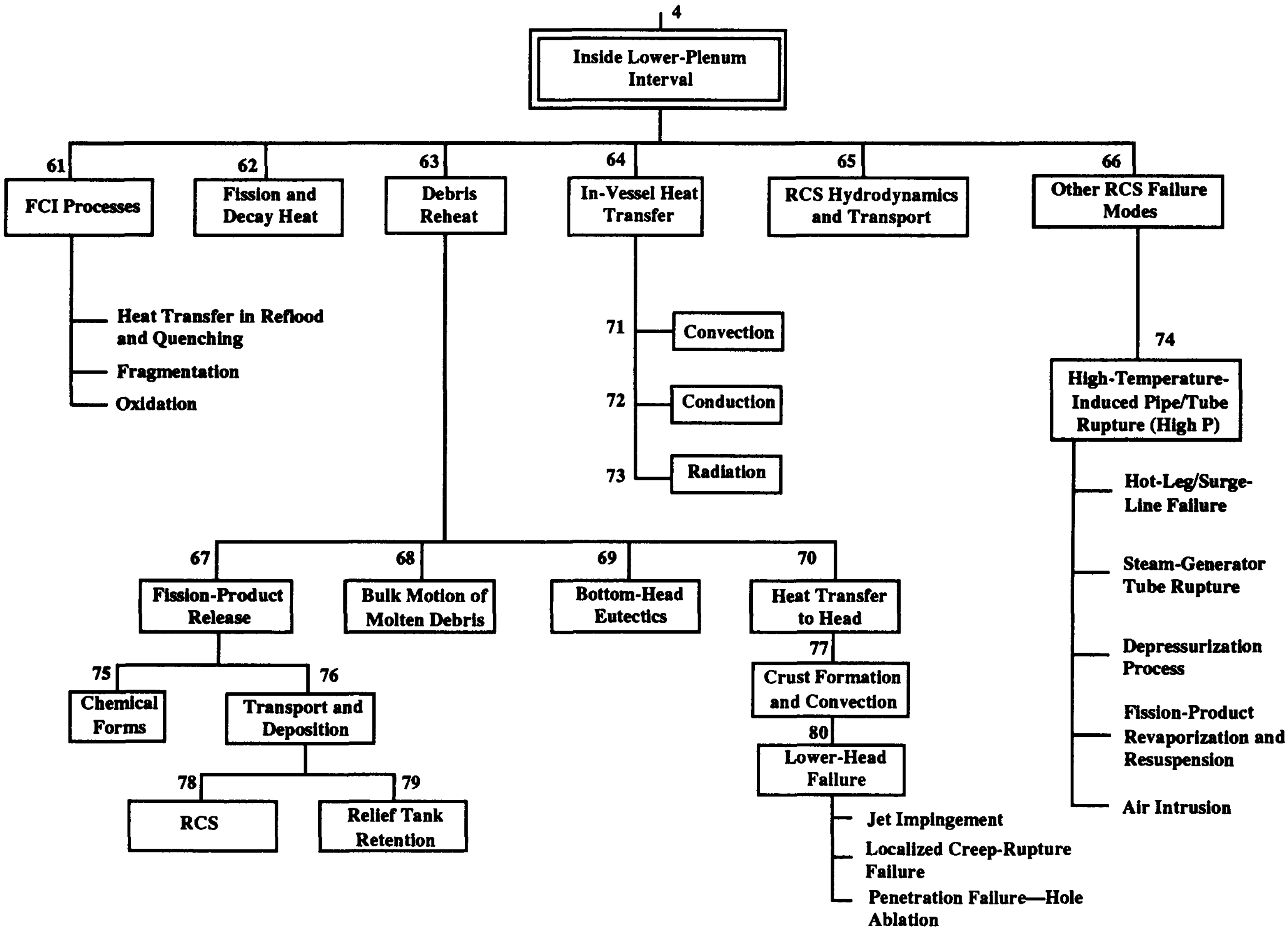
RCS Hydrodynamics and Transport
49
Including Reactor Reflood and Pressurizer Drainage
50
In-Core Natural Circulation
53
Effect of Noncondensables on Heat Transfer and Circulation
51
Discharge and Blowdown
52
Upper Plenum, RCS, and Steam Generators
54
Upper-Plenum, Hot-Leg, and Steam-Generator Flow/Circulation
55
Heat Transfer to Upper Head, Pipes, and Surge Line
56
Turbulence Mixing in Pipes and Plena
57
Effect of Noncondensables on Mixing Heat Transfer and Circulation
30
Melting and Freezing
58
Core Blockage Formation
Ex-Rod Relocation
In-Rod Relocation
Structural Material Relocation
59
Molten Pool and Crust Formation
Molten Pool Behavior
Heat Transfer to Interface
Crust Failure Location
Relocation
60
Solid Material Relocation and Rubblization


4
Inside Lower-Plenum Interval
61
FCI Processes
62
Fission and Decay Heat
63
Debris Reheat
64
In-Vessel Heat Transfer
65
RCS Hydrodynamics and Transport
66
Other RCS Failure Modes
Heat Transfer in Reflood and Quenching
Fragmentation
Oxidation
71
Convection
72
Conduction
73
Radiation
74
High-Temperature-Induced Pipe/Tube Rupture (High P)
Hot-Leg/Surge-Line Failure
Steam-Generator Tube Rupture
Depressurization Process
Fission-Product Revaporization and Resuspension
Air Intrusion
67
Fission-Product Release
68
Bulk Motion of Molten Debris
69
Bottom-Head Eutectics
70
Heat Transfer to Head
75
Chemical Forms
76
Transport and Deposition
78
RCS
79
Relief Tank Retention
77
Crust Formation and Convection
80
Lower-Head Failure
Jet Impingement
Localized Creep-Rupture Failure
Penetration Failure—Hole Ablation

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C-1. R. Jenks, Los Alamos National Laboratory, "Technical Reporting Deliverables 2A, 2B and 2C," letter to Y. S. Chen, US Nuclear Regulatory Commission, N-12-91-727 (October 22, 1991).

APPENDIX D

Ranking Severe-Accident Phenomena ("Hierarchy-by-Interval" Approach)

The Hierarchy-by-Interval Table (Table D-I) combines the results of the SCDAP/RELAP5 review with the decomposition of dominant phenomena block diagrams. The block diagrams provide a list of potential accident phenomena occurring within the four accident intervals:

Interval 1 - Initial transient, up to the start of core voiding.

Interval 2 - Core uncovery, up to a peak temperature of 1500 K or the time when a control-rod sheath would fail.

Interval 3 - Melt relocation and slump, up to the start of lower-plenum phenomena.

Interval 4 - Core debris in lower plenum, up to the point of vessel failure.

With this classification, some phenomena can occur within several intervals. For example, heat conduction will be listed in most intervals. This phenomena list is the basis for Table D-I, Column 1; the numbers and titles are taken from the decomposition block diagrams.

The initial task was to relate the SCDAP/RELAP5 code models with the dominant phenomena. The Committee, with help from INEL, generated Column 2 (where the numbers and titles refer to Section 2 of this document). The various models in Section 2 have been reviewed by the Committee. Therefore, this table relates the dominant phenomena to the particular SCDAP/RELAP5 models used to calculate that phenomena.

In Columns 4, 5, and 7, the results of the model review are tabulated using the criteria established in Chapter 1. Column 4 lists the modeling detail as described in Section 1. Column 5 gives the technical adequacy as defined in Section 1, Table 1-I, and discussed in Section 2. Column 7 indicates the present status of code validation. Note that many phenomena are modeled within the RELAP5 code but were not reviewed by this Committee, and other phenomena are not modeled by this code.

The next task, judging how well the basic physical processes are understood, was designed to evaluate the state of knowledge of the dominant phenomena. The Committee has chosen to use a three-level scheme of understood, questionable, and poorly understood, in keeping with Fig. 1-1. It is our view that processes that are poorly understood do not require the modeling detail of understood processes. The results of this task are shown in Column 3.

The next task was to determine a level of importance to the dominant phenomena. To do this, the Committee originally chose four figures of merit against which the phenomena could be rated (high, medium, or low) as contributing to its evaluation. The figures of merit chosen are conceptually measurable and will effect containment phenomena occurring during the severe accident. The figures of merit are:

Source Term:	The timing, magnitude, and phase of fission-product release to the containment.
Hydrogen Generation:	The timing and release rate of hydrogen to the containment.
Melt Eject Characteristics:	The composition of the corium released from the vessel. This includes the mass fractions, melt fractions, and temperature of the ejected material.
RCS Failure:	The timing and location of failure. This is primarily a function of the RCS temperature distribution and pressure history.

The Committee generated a figure-of-merit importance table using the phenomena list and these figures of merit. It was then decided that the last three figures of merit were all related to core damage; therefore, Column 6 shows a source-term average and an average of the other three figures of merit that we will call core damage for each of the dominant phenomena.

The last task was to resequence the table to further indicate importance. Therefore, within each interval the more important phenomena are listed first. For example, on the second page of the table, fission-product release and chemistry are listed first because of their great importance to the determination of the source term. Similarly, on the next page, metal oxidation is rated low because there is little oxidation below 1500 K.

This table can be used to provide a rationale for further work in the severe-accident area. Those phenomena judged to be highly or moderately important should be examined for completeness. For example, relocation phenomena were generally considered to be poorly understood but were of high importance; thus, work that has the goal of understanding the physics should be emphasized. In general, a low rating in knowledge of physics implies a need for fundamental work, a low rating in technical adequacy (4 or 5) implies a need for better computer models, and a low rating for validation status implies a need for further assessment work.

Table D-I

Hierarchy by Interval
(Interval 1) Ranking of Phenomena for Initial Transient Heatup
$T_{exit} \leq T$ Saturation

Phenomena	S/R5[a] Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance[b] ST	Importance[b] CD	Validation Status
5 Fission heat	2.2– Nuclear heat 2.5– Fuel-state models RELAP5	Understood Understood	0th Order 0th Order	7 7	M	ML	Validated Validation possible
5 Decay heat	2.22–Fission-product heat 2.5– Fuel-state models	Understood Understood	0th Order 0th Order	7 7	ML	L	Validated Validation possible
6 Heat transfer to RCS structures	RELAP5			Not reviewed	L	L	
7 RCS hydrogen and transport	RELAP5			Not reviewed	L	L	

[a]SCDAP/RELAP5

[b]ST and CD refer to Source Term and Core Damage, respectively. H, MH, M, ML, and L refer to High, Medium High, Medium, Medium Low, and Low, respectively.

Table D-I (cont.)

(Interval 2) Ranking of Phenomena for Core Uncovery T<1500 K

Phenomena		S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
11	Hydrodynamics and transport	RELAP5			Not reviewed	M	MH	
12	Metal oxidation zircaloy rods (see also 12–19)	2.1– Material oxidation	Understood	1st Order	1	ML	MH	Validated
9	Decay heat	2.22–Fission product 2.5– Fuel-state models	Understood Understood	0th Order 0th Order	4 4	MH	M	Validated
20	Convection	RELAP5			Not reviewed	M	M	
11	Hydrodynamics and transport steam-generator tube rupture	RELAP5			Not reviewed	M	M	
14	Clad ballooning and failure	2.7– Cladding deformation	Understood	0th Order	4	M	M	Validation possible
14	Clad ballooning–fuel-rod internal pressure	2.8– Fuel-rod internal pressure	Understood	0th Order	4	M	M	Validation possible
11	Hydrodynamics and transport Hot-leg/surge-line failure	RELAP5			Not reviewed	ML	M	
22	Radiation	2.12–Radiation heat transfer	Understood	1st Order	1	L	M	Validation possible

Table D-I (cont.)

(Interval 2) Ranking of Phenomena for Core Uncovery
T<1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
8 Core state (see 13–14)							
21 Conduction	2.6– Heat conduction	Understood	1st Order	1	M	ML	Partially validated
	2.4– Effective materials properties	Understood	1st Order	1			Partially validated
	2.5– Fuel state	Understood	0th Order	4			Validation possible
	2.11–Control rod and core structure	Understood	1st Order	1			Validation possible
	2.15–Lower-plenum debris heatup	Questionable	0th Order	4			Inadequate implementation
	2.21–Materials properties	Understood	1st Order	1			Validated
17 Metal oxidation steam starvation	2.1– Material oxidation	Understood	1st Order	1	ML	ML	Validated

Table D-I (cont.)

(Interval 2) Ranking of Phenomena for Core Uncovery T<1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
12 Metal oxidation in-core steel	2.11–Control rod and core structure	Understood	0th Order	4	L	ML	Validated
12 Metal oxidation RELAP5 structures		Understood		7	L	ML	Validation possible
13 In-vessel heat transfer (see 20–22)							
10 Fission-product release (see also 15–16)	2.10–Fission-product release	Understood	1st Order	1	H	L	Validation possible
16 Fission-product chemistry	2.19–Vapor evaporation/ condensation 2.21–Materials properties gases	Understood Understood	0th Order No Factors	7 7	MH	L	Validation possible
15 Fission-product transport and deposition (see also 23–24)	See 44	Understood	1st Order	1	MH	L	Validation possible
23 Fission-product deposition in RCS	See 43	Understood	1st Order	1	M	L	Validation possible

Table D-I (cont.)

(Interval 2) Ranking of Phenomena for Core Uncovery T<1500 K

Phenomena		S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
24	Aqueous chemistry suppression pool/ice condensers		Understood		7	M	L	Validation possible
19	Fuel-coolant interactions (fragmentation)		Questionable	0th Order	4	L	L	Insufficient data
9	Fission heat	2.2– Nuclear heat 2.5– Fuel-state models RELAP5	Understood Understood	0th Order 0th Order	7 7	L	L	Validated

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump
T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
49 Reflood	2.25–Severe-accident thermal hydraulics	Questionable		4	H	H	Insufficient data
34 Convection	RELAP5			Not reviewed	H	H	
30 Melting and freezing (see also 58–60)	2.4– Effective materials properties				M	H	
58 Core blockage formation	2.9– Liquefaction, flow, and solidification	Questionable	0th Order	4	M	H	Inadequate implemen-tation
	2.13–Core-region debris modeling	Poor	0th Order	3			Insufficient data

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
58 Core blockage formation ex-rod relocation	2.9– Liquefaction, flow, and solidification	Poor	0th Order	3	M	H	Insufficient data
	2.13–Core-region debris modeling	Poor	0th Order	3			Insufficient data
58 Core blockage formation in-rod relocation	2.26–Additional models being developed or upgraded	Poor		6	M	H	Insufficient data
28 Metal oxidation zircaloy rods (see also 37–40)	2.1–Material oxidation	Understood	1st Order	1	M	H	Validated

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump
T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	CD	Validation Status
45 Relocation and lower-core plate	2.14–Core slumping 2.15–Lower-plenum debris heatup 2.26–Additional model being developed or upgraded	Poor Poor	0th order 1st order	3 3	M	H	Insufficient data
33 Crucible relocation and slump (see 45–46)					ML	H	
50 In-vessel natural circulation (see also 53)	RELAP5			Not reviewed	H	MH	

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump
T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
32 Channel box and control rods relocation	2.11–Control rod and core structure	Questionable	0th Order	4	H	MH	
46 Relocation and debris fragmentation	2.13–Core-region debris modeling	Poor	0th Order	3	M	MH	Insufficient data
58 Core blockage formation structural material relocation	2.11–Control rod and core structure 2.26–Additional models being developed or upgraded	Questionable Questionable	0th Order	4	M	MH	Insufficient data
42 Eutectic Zr-U-O Dissolution T<Tmelt	2.9– Liquefaction, flow, and solidification 2.21–Materials properties	Questionable Understood	0th Order 1st Order	4 1	ML	MH	Validation possible

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	CD	Validation Status
42 Eutectic Zr-U-O Liquefaction T>Tmelt	2.9– Liquefaction, flow, and solidification 2.21–Materials properties	Questionable Understood	0th Order 1st Order	4 1	ML	MH	Validation possible
59 Molten pool behavior	2.13–Core-region debris modeling 2.26–Additional model being developed or upgraded	Questionable	0th order	4	ML	MH	Insufficient data
59 Molten pool heat transfer	2.13–Core-region debris modeling	Understood	0th order	4	ML	MH	Insufficient data
40 Metal oxidation molten zircaloy	2.26–Additional model being developed or upgraded	Questionable	No features	5	L	MH	Insufficient data
46 Debris heat transfer	2.13–Core-region debris modeling	Understood	No features	5	L	MH	Validation possible

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
54 Molten pool crust failure and relocation	2.13–Core-region debris modeling 2.14–Core slumping models	Questionable Questionable	0th Order 0th Order	4 4	L	MH	Insufficient data
25 Criticality	2.2– Nuclear heat RELAP5	Understood	No features	5	MH	M	Validation possible
26 RCS hydrodynamics and transport (see 47–50)					MH	M	
54 Upper-plenum, hot-leg, steam-generator flow	RELAP5			Not reviewed	MH	M	

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump
T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
35 Conduction	2.4– Effective materials properties	Understood	1st Order	1	MH		Validated
	2.5– Fuel state	Understood	0th Order	4	M		Validation possible
	2.6– Heat conduction	Understood	1st Order	1			Validated
	2.11–Control rod and core structure	Understood	1st Order	1			Validation possible
	2.15–Lower-plenum debris heatup	Questionable	0th Order	4			Inadequate implementation
32 Channel box and control rods oxidation/failure	2.11–Control rod and core structure 2.26–Additional models being developed or upgraded	Understood	No features	5	M	M	

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
32 Control structures heat transfer	2.11–Control rod and core structure 2.26–Additional models being developed or upgraded	Questionable	0th Order	4	M	M	Validation possible
25 Decay heat	2.22–Fission-product decay heat 2.5– Fuel-state models	Understood Understood	0th Order 0th Order	4 4	M	M	Validated Validation possible
29 Eutectic formation (see 41–42)					ML	M	
27 In-vessel heat transfer (see 34–36)					ML	M	
28 Metal oxidation in-core steel	2.11–Control rod and core structures	Understood	1st Order	1	L	M	Validated
37 Metal oxidation - steam starved	2.1– Material oxidation	Understood	1st Order	1	L	M	Validated
25 Fission heat	2.2– Nuclear heat 2.5– Fuel-state models RELAP5	Understood Understood	0th Order 0th Order	4 4	L	M	Validated Validation possible

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
36 Radiation rod to rod, rod to shroud	2.12–Radiation heat transfer	Understood	0th Order	4	L	M	Validation possible
36 Radiation ring to ring and axial		Understood	No features	5	L	M	Validation possible
36 Radiation SCDAP to RELAP		Understood	No features	5	L	M	Validation possible
40 Metal oxidation molten stainless steel		Poor		3	L	M	Insufficient data
40 Metal oxidation (grid spacers)	2.11–Control rod and core structure	Understood	No features	5	L	M	Validation possible
41 Eutectics nonfuel stainless steel - B4C	2.11–Control rod and core structure	Questionable	0th Order	4	L	M	Validation possible
60 Solid relocation (rubblization)	2.13–Core-region debris modeling	Poor	0th Order	3	L	M	Insufficient data
55 Heat transfer to upper heat structures and surge line	RELAP5			Not reviewed	L M		

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
39 Metal oxidation and fuel-coolant interaction (FCI)		Poor		3	L ML		Insufficient data
41 Eutectics nonfuel Zirconium - (stainless steel, Inconel, or Al_2O_3)	2.11–Control rod and core structure 2.26–Additional models being developed or upgraded	Questionable Understood	0th Order 1st Order	4 1	L ML		Validation possible
28 Metal oxidation RELAP5 stainless-steel structures		Understood		5	L	ML	Validation possible
28 Metal oxidation SCDAP stainless-steel structures	2.11–Control rod and core structure	Understood	No features	5	L	ML	Validation possible
31 Fission-product release (See also 43 and 44)	2.10–Fission-product release	Questionable	0th Order	4	H L		Inadequate implemen-tation

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump
T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
43 Fission-product chemistry	2.19–Vapor evaporation/ condensation	Understood	0th Order	7	H	L	Inadequate implemen-tation
44 Fission-product transport and agglomeration	2.17–Aerosol agglomeration	Understood	1st Order	1	H	L	Inadequate implemen-tation
44 Fission-product deposition in RCS (see also 47–48)	2.18–Aerosol particle deposition	Understood	1st Order	1	H	L	Inadequate implemen-tation
44 Fission-product deposition in RCS chemisorption in structures	2.20–Heterogeneous chemical reaction between chemical species and wall	Poor	0th Order	7	H	L	Insufficient data
47 Fission-product deposition in RCS, aqueous chemistry (solubility)		Questionable		7	H	L	Validation possible

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
48 Fission-product retention in pools (aqueous chemistry)		Questionable		7	H	L	Validation possible
51 Discharge and blowdown	RELAP5			Not reviewed	MH	L	
43 Fission-product deposition in RCS The effects of water chemistry		Questionable		7	M	L	Validation possible
47 Fission-product deposition in RCS heat source, decay heat, and gamma attenuation	2.23–Decay-heat energy deposition 2.22–Fission-product decay heat 2.24–decay-heat gamma attenuation complete absorption	Understood Understood Understood	0th Order 1st Order 0th Order	7 1 7	M	L	Insufficient data Insufficient data Insufficient data

Table D-I (cont.)

(Interval 3) Ranking of Phenomena for Melt Relocation and Slump T>1500 K

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
52 Upper-plenum, RCS, and steam-generator phenomena (see 53–56)							
53 Effect of noncondensables in core	RELAP5			Not reviewed	L	L	
55 Effect of noncondensables in plenum and RCS	RELAP5			Not reviewed	L	L	
56 Core bypass and turbulence in plenum and RCS	RELAP5			Not reviewed	L	L	

Table D-I (cont.)

(Interval 4) Ranking of Phenomena for Lower Plenum

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
66 RCS failure	Separate calculation	Understood	0th Order	4	MH	H	Validated
74 Depressurization	RELAP5	Not reviewed			H	MH	
61 FCI processes debris fragmentation	Input 2.14–Core-slumping models	Poor	Features	3	H	MH	Insufficient data
62 Fission and decay heat	2.15–Lower-plenum debris heatup	Understood	No features	5	M	MH	Validation possible
71 Convection	RELAP5	Not reviewed			M	MH	
61 FCI processes metal oxidation		Questionable		5	ML	MH	Insufficient data
61 FCI processes debris heat transfer to water	2.14–Core-slumping model	Poor	No features	6	L	MH	Insufficient data
63 Debris reheat lower-plenum molten pool formation	2.15–Lower-plenum debris heatup	Poor	0th Order	3	L	MH	Insufficient data

Table D-I (cont.)

(Interval 4) Ranking of Phenomena for Lower Plenum

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance		Validation Status
					ST	CD	
69 Bottom-head eutectics		Questionable		5	L	MH	Validation possible
77 Lower-plenum crust behavior	2.15–Lower-plenum debris heatup	Questionable	0th Order	4	L	MH	Insufficient data
70 Heat transfer to lower head (also see 80)	2.15–Lower-plenum debris heatup	Understood	1st Order	1	L	MH	
80 Lower-head failure jet impingement		Questionable		5	L	MH[a]	Validation possible
80 Lower-head failure localized creep rupture	2.16–Structural creep rupture	Questionable	0th Order	4	L	MH[a]	Validation possible
80 Lower-head failure penetration failure	2.16–Structural creep rupture	Questionable	0th Order	4	L	MH[a]	Insufficient data
80 Lower-head failure hole ablation		Questionable			L	M	Insufficient data
65 RCS hydrodynamics and transport (see 26)							

[a]These failure phenomena are judged to be of higher importance than their figure-of-merit ranking.

Table D-I (cont.)

(Interval 4) Ranking of Phenomena for Lower Plenum

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
64 In-vessel heat transfer (see 71–73)					L	M	
73 Radiation debris to structures (see 34)		Understood	No features	5	L	M	
68 Bulk motion of molten debris (layers)	2.14–Core-slumping models 2.15–Lower-plenum debris heatup	Questionable Understood	No features	5	L	ML	Validation possible
67 Fission-product release (see also 74–76)		Poor		7	H	L	Insufficient data
74 Fission-product resuspension		Questionable		7	H	L	Insufficient data
75 Fission-product chemistry		Questionable		7	H	L	Insufficient data
79 Suppression pool retention (aqueous chemistry)		Questionable	No features	7	H	L	

Table D-I (cont.)

(Interval 4) Ranking of Phenomena for Lower Plenum

Phenomena	S/R5 Code Model	Knowledge of Physics	S/R5 Physics	Technical Adequacy	Importance ST	Importance CD	Validation Status
74 Fission-product release and metal oxidation with air intrusion		Understood		7	H	L	Validation possible
61 FCI processes fission-product release		Questionable		7	MH	L	Insufficient data
76 Fission-product transport and deposition (see 43)				1	MH	L	Validation possible
72 In-core conduction (see 33)					M	L	

APPENDIX E

Detailed Review of Phenomenological Models in SCDAP/RELAP5

E.1. Material Oxidation Model

E.1.1. Model Description and Pedigree

The code treats the oxidation of zircaloy and stainless steel using parabolic rate equations with parameters derived from well-known models, e.g., Cathcart-Pawel and Urbanic-Heidrick. A steady-state model is used, except for zircaloy when the oxidation rate is high; in that case, a transient model is used that solves for temperatures and oxidation rates simultaneously. Hydrogen production and steam removal are treated. The oxidation process is limited by the availability of oxidizable material and steam.

Simplifying assumptions are that (1) volume changes on oxidation are considered only for zircaloy; (2) for structures with multiple layers, the layers oxidize in sequence; when a layer is completely oxidized, its presence is then ignored and oxidation of the next layer proceeds unhindered; and (3) oxidation of the inner surface of ballooned cladding (or failed shroud) can be treated simply by doubling the oxidation rate rather than by modeling the inner oxidation separately.

There is no treatment of oxidation in porous or cohesive debris beds, and the model for the oxidation of zircaloy by steam in the intact rod geometry is not directly coupled to that of the dissolution of UO_2 by zircaloy on the cladding inner surface. Dissolution of the outer zirconium oxide shell by metallic zircaloy is not mechanistically modeled.

The models for the oxidation of zircaloy and stainless steel are described in Refs E.1-1, E.1-2, E.1-3, and Presentation 1-8 of Ref. E.1-4.

The model for boron-carbide oxidation is reviewed in the section on control-rod and core-structure modeling.

E.1.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The kinetics of oxidation of solid and liquid zircaloy and of stainless steel are well established for well-defined geometries. Data on oxidation of zirconium-rich eutectics are less widely available. The kinetics mainly follow the parabolic form expected from diffusion theory; the transition from parabolic to cubic kinetics seen below 1000°C for zircaloy has not been mechanistically modeled, but empirical correlations are available.

There is no treatment of the cubic kinetics in the code; the parabolic correlations used (reasonably) at higher temperatures are extrapolated downward (to 1000 K). There are no published models in the reflood interval (Interval 1) to account for the sharp increase in oxidation in hydrogen production. It is likely that shattering existing oxide shells by thermal shock, thus exposing underlying metal that can then oxidize rapidly, is an important mechanism that remains to be treated. Relief of steam starvation may be a factor in some cases.

The overall treatment is reasonable for intact geometry, although there may be some error in using equations derived assuming an infinite geometry in a finite geometry (as when most of the cladding wall is consumed). The assumption of doubling of the oxidation rate to account for inner-surface oxidation on the cladding breach is not mechanistic but probably does not give rise to serious error.

E.1.3. Implementation Within the Code

Zircaloy oxidation is determined from the fuel-rod model, fuel-rod simulator, control-rod model, and flow-shroud model. However, the oxidation in the plenum regions of fuel, simulator, and control rods is not treated. Heat generation resulting from oxidation is passed into the relevant heat-conduction models. The generated hydrogen is released to the relevant coupled RELAP volumes from which steam consumed is correspondingly removed.

E.1.4. Results of Model Sensitivity Studies

None specific were found.

E.1.5. Results of Benchmarking/Validation Studies

The oxidation models have been validated in the course of analysis of many severe fuel-damage experiments (Refs. E.1-5 to E.1-7 and Presentation 1-8 of Ref. E.1-4) in the CORA series, the Power Burst Facility–Severe Fuel Damage series, LOFT LP-FP-2, the Annular Core Research Reactor "DF" series ISP-28 (PHEBUS B9+ test, Ref. E.1-8), and TMI-2. Typically, the total hydrogen production and, in some cases, the extent of cladding oxidation calculated by the code was compared with the experimental data. In some CORA tests, for example, on-line measurements of hydrogen production were available.

Where oxidation has taken place wholly (ISP-28) or largely (in some CORA tests) in an intact geometry, predictions are good. If there is significant oxidation further into the sequence involving relocated material, oxidation is underpredicted; this underprediction is particularly marked during reflood phases, e.g., for LP-FP-2, CORA-12, and TMI-2. In CORA, there is substantial oxidation in the plenum region (relatively large in this case) that is not modeled by the code. Oxidation of previously refrozen zirconium-rich melt is also observed in CORA tests but not modeled by the code.

In general, the acceleration in the heatup rate brought on by the autocatalytic zircaloy/steam reaction is adequately modeled, provided that the boundary conditions (particularly important in small bundles where radial heat losses can be high) are well simulated.

E.1.6. Identified Deficiencies and Options for Model Improvements

Intact geometry. Oxidation for intact fuel-rod geometries is adequately modeled, but the material inventory in the plenum region should be taken into account. Tracking the hydrogen production from intact steel-core structures should be considered (otherwise the total production will be underestimated), as well as introducing (Presentations 1-8 and 1-18 of Ref. E.1-4) a cubic kinetics model for zircaloy to operate in the 700–1000°C range and limiting of the rate of oxidation due to hydrogen blanketing (important only at high concentrations, ~90%). The mass balance needs to be checked for double-sided oxidation and for highly oxidized material.

Degraded geometry. Oxidation of zirconium-rich relocating and refrozen material is not adequately treated (Ref. E.1-7). This leads to underprediction of hydrogen production in the later phases of the accident sequences. The database, compared with that available for the undegraded cladding, is not extensive.

Reflood. The increase in hydrogen production observed under reflood conditions in several experiments is not calculated by the code, and theoretical understanding is lacking. This is a major deficiency, with implications for accident management, and is recognized by the code authors (Presentations 1-8 and 1-18 of Ref. E.1-4). It is understood that new models in this area are under development at INEL, but this work is presently unavailable for review.

E.1.7. Importance of Model to Prediction of Dominant Phenomena

Oxidation of zircaloy cladding by steam has several important effects on the progression of a severe accident (Ref. E.1-9), such as (1) increasing the rate of the temperature rise (a consequence of the highly exothermic nature of the reaction); (2) generating hydrogen; (3) forming an oxide shell that can hinder relocation of molten cladding and Zr-U-O eutectics; (4) embrittling the cladding, rendering it liable to fracture on thermal shock (e.g., during reflood); and (5) increasing the melting temperature of the core debris. These are all very significant phenomena; therefore, accurate modeling of oxidation is highly important.

Oxidation of steel is important only at higher temperatures (the rate exceeds that of zircaloy above ~1500 K), and the heat of reaction is much lower. Oxidation of core structures such as the upper plenum can contribute significantly to the total hydrogen production, e.g., if in-vessel natural circulation is present.

E.1.8. Technical Adequacy of Model

The modeling of oxidation largely reflects the state of knowledge in this important area. For intact geometry, the physics is generally well understood at temperatures relevant to severe accidents. As the core structure degrades, the knowledge of oxidation processes becomes more obscure, although it is known from experiments that oxidation processes are still significant. In quench situations, the mechanisms governing the excess hydrogen production are not understood. More attention needs to be paid to understanding the oxidation phenomena in these later phases.

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E.2. NUCLEAR-HEAT, FUEL-ROD MODEL

E.2.1. Model Description and Pedigree

The total power level is calculated as

$$Q(z,r,t)= Q_p(t)Z_p(z)R_p(r) + Q_d(t)Z_d(z)R_d(r) \quad ,$$

where

$Q(z,r,t)$	= Total nuclear heat at position (r, z) and at time t,
$Q_p(t)$	= Component average prompt neutron amplitude at time t,
$Q_d(t)$	= Component average decay-heat amplitude at time t,
$Z_i(z)$	= Axial power-peaking factor at position z (i=p and d), and
$R_i(r)$	= Radial peaking factor at position r.

The time-dependent prompt neutron power amplitude is provided as input to the code. The decay-heat amplitude is determined on the basis of the ANSI/ANS-5.1-1979 decay-heat standards, considering fission products with corrections to neutron capture (on the basis of an empirical relation) and contributions to decay heat due to the decay of ^{239}U and ^{239}Np.

For prompt neutrons, the axial power-peaking factors can also vary as a function of time, based on user-supplied distributions. On the other hand, the decay-heat axial power-peaking factors are allowed either to adjust themselves exponentially to the prompt power peaking factors or are supplied through separate user-input values. It is assumed in SCDAP/RELAP5 that radial power distributions remain time invariant.

The decay-heat radial peaking factor is assumed to be identical to that of prompt neutron power for nonfuel components, while for the fuel components, it is assumed that gamma-ray energy is ~one-half the decay power within fuel components; the radial peaking factor is then adjusted accordingly.

During severe accidents, reduction factors are applied to the fission-product decay heat due to a loss of volatile fission products from the fuel matrix. Justifications are not provided for neglecting the effect on decay heat because of the loss of lower-volatility species (i.e., tellurium, barium, strontium, etc.).

Adjustments to both prompt and decay-power contributions are made to correct for a significant movement of mass as a result of fuel-rod degradation and relocation. The decay-heat term is corrected for changes of fuel mass and density. The prompt power amplitude is multiplied by the ratio of current material density to initial material density as a way to account for any fuel disruption or phase change. This approximation is not expected to be valid during core disruption and is therefore not justified.

E.2.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The present nuclear-heat model, when modified to address the deficiencies listed below, should be adequate for use under the most severe accident conditions of interest. Of course, the limitations of the point kinetics models in RELAP5 restrict the code's application to situations where significant spatial variations in reactivity can arise. Accidents involving recriticality following an initially uncovered and damaged core and Anticipated Transients Without Scram (ATWS) are beyond the scope of SCDAP/RELAP5; however, if details of neutron kinetics for these accidents can be worked out via another computer code, then the spatial and temporal power variations should be usable as inputs to SCDAP/RELAP5 for determining consequences on the subsequent progression of accidents.

E.2.3. Implementation Within the Code

The code documentation does not provide an adequate description of the interface between the RELAP5 reactor kinetics package, decay-heat formulation, and SCDAP nuclear-heat model. In addition, when the fission-product transport models are not activated, all of the decay heat associated with the released fission products are lost from the system; therefore, energy is not conserved. This is a serious problem that must be remedied.

E.2.4. Results of Model Sensitivity Studies

None were reported.

E.2.5. Results of Benchmarking/Validation Studies

None were reported.

E.2.6. Identified Deficiencies and Options for Model Improvements

Several approximations have been noted that are not fully justified.

1. There is no application of G factors (neutron capture correction) by RELAP5. This creates inconsistencies between SCDAP and RELAP5 parts of the package.

2. The assumed recoverable energy of 195.33 MeV/fission is incorrect. Typically, there is ~6.8 MeV/fission more because of activation of structural material.

3. SCDAP neglected the effects of delayed neutrons after reactor shutdown. The statement that "user can force consideration of ... as an additional prompt neutron heat source" is wrong.

4. The SCDAP nuclear-heat model is inadequate for applications to high-burnup-fuel LWRs because it does not include fissioning of ^{238}U and ^{239}Pu.

5. The SCDAP nuclear-heat models rely on user inputs for determination of prompt and decay power. This is consistent with code objectives/targeted applications.

E.2.7. Importance of Model to Prediction of Dominant Phenomena

An accurate spatial and temporal variation of heat generation is required for all severe-accident conditions. However, accidents involving large spatial variations in reactivity are beyond the scope of SCDAP/RELAP5.

The physics of the nuclear-heat model is well understood, especially for accidents involving reactor scram. Uncertainties exist in the prediction of prompt power amplitude and distribution for ATWS, particularly following core uncovery and fuel and structural degradation.

E.2.8. Technical Adequacy of Model

The present models for the most part rely on user-specified inputs for determination of spatial power-generation rates during severe-accident conditions.

Overall, the nuclear-heat model appears reasonable. However, the above-noted inadequacies need to be addressed.

E.3. Electrically Heated Fuel-Rod Model

E.3.1. Model Description and Pedigree

A model is provided for the electrically heated fuel-rod simulators of the type used in the CORA facility at KfK. The simulator consists of a tungsten-resistive heating element surrounded by a stack of annular UO_2 pellets encircled by standard zircaloy fuel-rod cladding. Electrical power is fed to the tungsten heater through copper and molybdenum electrodes in series at each end; the voltage and current are measured at the outer ends of the copper electrodes, which are cooled in water baths. A significant fraction of the power (typically 10–15%) is dissipated in the electrodes, and there is a strong positive feedback between temperature and power distribution due to the temperature dependence of the tungsten electrical resistance.

The models treat the power generation, taking into account the partition of energy generation between the tungsten and electrodes in calculating the energy released in the heated section. Heat conduction (axial and radial) is calculated in the heated section using the standard SCDAP/RELAP5 two-dimensional model. Oxidation and ballooning of the cladding are treated in the same way as for standard fuel rods. There is no treatment of the material in the plenum regions (e.g., no modeling of any cladding oxidation, heat generation in the electrodes, nor axial heat conduction to the water baths). Only the tungsten-heated regions can be connected to RELAP volumes.

The model is described in Ref. E.3-1 and is based on a treatment developed by W. Hering at KfK for the standalone SCDAP code; in this treatment, the heat-conduction model is extended into the plenum regions.

E.3.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The model is a physically reasonable first-order treatment of well-understood phenomena, as far as the heated section is concerned. The main difficulty is that the same treatment is not extended into the plenum regions, so that heat generated there electrically and by cladding oxidation is lost to the system. Because the heat-conduction model does not extend as far as the heat sinks (water baths), it is difficult to model precisely the true axial boundary conditions.

E.3.3. Implementation Within the Code

The implementation of the model within the code is very similar to that of the standard fuel-rod model. A submodel provides the resistance of the electrical conductors (including the tungsten) as a function of temperature and forms the basis of the treatment of the partition of electrical heat in the simulator. The diameter of the heating element is read in as data, but quantities depending directly on this diameter are also hardwired in the code, providing the potential for error if a nonstandard value for this parameter is chosen. The presence of the heating element is not taken into account in the UO_2 dissolution model.

E.3.4. Results of Model Sensitivity Studies

None were found.

E.3.5. Results of Benchmarking/Validation Studies

The model has been indirectly tested through the extensive analyses of CORA experiments carried out for validation purposes (Refs. E.3-2, E.3-3, and Presentation 2-33 of Ref. E.3-4). The model generally behaves well, though the lack of a model for heat conduction through the electrodes in the plenum regions tends to result in overprediction of the temperatures at the ends of the heated section (particularly at the lower end).

Furthermore, the lack of a model for the plenum region leads to an underprediction of hydrogen production (Ref. E.3-3) because in most CORA experiments, the flame front extends into the upper-plenum region. (This additional hydrogen production can, however, be estimated by hand calculations from observed temperature data.) The deficiencies mean that care must be taken in evaluating results of these analyses, though they are unlikely to lead to major misinterpretations of the data.

The model is also being evaluated (Presentation 2-32 of Ref. E.3-4) in the context of ISP 31 (based on the CORA-13 experiment), which is currently in progress.

E.3.6. Identified Deficiencies and Options for Model Improvements

The deficiencies relating to the lack of modeling for the plenum regions and of possible inconsistencies between input and hardwired geometrical data for the heating element have been recognized by EG&G (Presentation 1-18 of Ref. E.3-5), and it is recommended that these deficiencies receive attention. The fact that the presence of the heater rod is not taken into account in the UO_2 dissolution model needs to be considered if there is very extensive UO_2 dissolution (not the case in many CORA tests).

A further deficiency is that the later-interval models cannot be invoked in a CORA environment. This means that quench-induced shattering observed in the CORA quench tests cannot be modeled (e.g., in the ISP-31 exercise). However, the lack of modeling in later severe-accident intervals is not a significant problem in the analysis of the other tests, where debris beds are not generally formed.

E.3.7. Importance of Model to Prediction of Dominant Phenomena

The CORA experiments provide an important database for development and validation of early-interval melt progression models, and mechanistic treatment of the heater rods is essential in their analysis. It is particularly necessary to treat the distribution of electrical heat generation in the heater rods because this is strongly temperature dependent.

E.3.8. Technical Adequacy of Model

The model is basically sound. The corrections needed to resolve outstanding deficiencies are judged to be reasonably straightforward.

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E.4. Effective Materials Properties

E.4.1. Model Description and Pedigree

A scheme for transforming properties of heterogeneous layers into homogeneous properties has been developed. The properties considered are thermal conductivity, volumetric heat capacity, and heat-generation rate and are employed in the finite element approach used to solve for temperature profiles in the solids. The heterogeneities result from the presence of different materials (e.g., UO_2, Zr-U-O, ZrO_2, structural materials, and absorber materials) in different phases (solid and liquid). Volume and temperature averaging is used to obtain effective volumetric heat capacity and inverse thermal conductivity. Local energy generation due to chemical reaction or decay heat is also averaged over the volume.

The effect of melting or solidification is accounted for by correcting the specific heat for the latent heat of fusion. Layers specified by geometry-independent and geometry-dependent properties can be considered in the model. Both axial and radial variations in slab cylindrical and spherical geometries can be considered. The effect of the burnup on thermal conductivity is considered through the FRAP-CON2 code. These parameters are used in the transient, one-dimensional, heat-conduction equation for calculating the temperature in the heat structures.

Parallel and Series Resistance. The well-established concept of parallel and series resistance is used to determine effective thermophysical properties of materials for the heat-conduction volume element in both one-dimensional planar and cylindrical geometry. Layers specified by geometry-independent and geometry-dependent properties are considered to exist. The former include unirradiated fuel, zircaloy cladding, ZrO_2, liquid Zr-U-O, frozen Zr-U-O, and structural and absorber materials. The latter include cracked fuel, relocated fuel, and gaps, as well as voids. Oxidation of zirconium by O_2 is not considered.

Volumetric Heat Generation. The effective volumetric heat-generation rate due to both volumetric and surface (i.e., oxidation and dissolution) contributions in a heat-conduction element with several material layers is computed by averaging the contributions across the layers from the component center to the component surface.

Integral Transformation. The concept of effective heat capacity is used to treat phase transformation at the phase-change front. To account for the change in enthalpy during the transformation at a front, a product of an effective volumetric heat capacity and a small temperature jump (which is equivalent to the latent heat of fusion) is introduced. This heat capacity is then used in the corresponding heat-conduction element to compute the temperature when phase change occurs in the element. This is a procedure well established in the literature.

E.4.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics is well known. The approach is consistent with a first-order model.

E.4.3. Implementation Within the Code

The model is called as required from the heat-conduction model for the core components. Implementation within the code is appropriate.

E.4.4. Results of Model Sensitivity Studies

Some sensitivity studies have been reported (Presentation 1-8c of Ref. E.4-1); however, the source of the figures for Table 11.1, found in Ref. E.4-1, was not provided and will have to be obtained and discussed.

E.4.5. Results of Benchmarking/Validation Studies

Some cases to test the correct implementation of the models have been run, but they do not represent validation.

E.4.6. Identified Deficiencies and Options for Model Improvements

The method of how voids, cracks, etc., are treated in the code is unclear. Because feedback exists between properties and temperature, an iterative procedure will be required. It is not clear if such a procedure is being used. Although Zr-U-O melts are considered, oxidation of zircaloy by UO_2 was not allowed in the zircaloy oxidation models. Use of an average reciprocal of thermal conductivity is not consistent with Fourier's law. A parameter used to account for burnup does not approach unity for the zero burnup case.

The principal deficiency identified is the need for some additional discussion and clarification of the parallel and series-resistance approach (Ref. E.4-2, pp. 3-21 to 3-22). For example, are certain equations specified in Reference E.4-2 appropriate for series, parallel, or both arrangements?

The use of either the series or parallel thermal resistance concepts for layers specified by geometry-dependent effective thermal conductivity needs to be assessed. On strictly theoretical grounds, a correct effective thermal conductivity cannot be computed using the volume-averaging approach, particularly for the parallel-resistance concept when there is a large difference in the thermal conductivity between adjacent material layers in the element of volume. Finally, the average thermal conductivity is not calculated correctly; rather, an inverse of the thermal conductivity is calculated that relates to the thermal resistance.

E.4.7. Importance of Model to Prediction of Dominant Phenomena

A model is important in predicting the temperature of the heat structures, but the model-related uncertainties will play a smaller role in comparison to other uncertainties that exist with respect to configuration of the core in the degraded state and materials properties.

E.4.8. Technical Adequacy of Model

The model is considered to be first order. The treatments are mostly state of the art, and there are no serious errors.

Two main concerns with respect to this model are the lack of iterative procedure for determination of dependence of properties on temperatures and the manner in which average thermal resistance is evaluated.

On the whole, the model is judged to be adequate. The model is described in sufficient detail, and the principal need is for some additional discussion and clarification of the resistance concept. The model does not calculate the properties of debris beds and therefore is not applicable to late phases of an accident.

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E.5. Fuel-State Models

E.5.1. Model Description and Pedigree

The fuel-state model (Refs. E.5-1 and E.5-2) defines the material layers and composition of each layer at each node of a fuel-rod component, modifies the axial power profile to account for relocation of material, and calculates the temperatures of the grid spacers. It accepts relevant fuel data from the liquefaction, oxidation, and ballooning models.

The method for calculating the axial peaking factor in relocated crusts is to form a weighted average, based on the cross-sectional areas of fuel and mixtures of fuel and cladding, of the peaking factors in the axial nodes from where the material came. These local peaking factors are used only for delayed sources of heat generation; peaking factors for prompt sources are provided by the user. The nuclear heat is then recalculated for nodes in which a frozen crust is present. The definition of material layers, etc., consists simply of logically sorting data from the relevant fuel behavior models. Grid temperatures are set to the outer cladding temperatures of the corresponding axial nodes, averaging if on a node boundary. No information regarding the pedigree of the models was found in the documentation.

E.5.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics involved in the fuel-state model is well understood, but the treatment in the code is judged zeroeth order by comparison with the assessment for the nuclear-heat model.

E.5.3. Implementation Within the Code

The fuel-state model is called from within the fuel rod and fuel-rod simulator models.

Although the documentation (Refs. E.5-1 and E.5-2) refers to modeling the change in power profile due to fuel fragments slumping into a ballooned region ("freloc" model), no evidence can be found of such modeling in the current code.

E.5.4. Results of Model Sensitivity Studies

None were found.

E.5.5. Results of Benchmarking/Validation Studies

None were found.

E.5.6. Identified Deficiencies and Options for Model Improvements

The code authors have identified (Presentations 1-8 and 1-18 of Ref. E.5-3) that the model may be inappropriate for high burnups, where radial peaking factors may be large. It is also accepted that the model is inappropriate in cases of recriticality.

E.5.7. Importance of Model to Prediction of Dominant Phenomena

A calculation of the axial power distribution is required to calculate the fuel temperature under all accident conditions. It is important to allow for the changes in this distribution because of material relocation.

E.5.8. Technical Adequacy of Model

The model is considered zeroeth order. It is not, however, likely to be a limiting factor in most severe-accident calculations. The modeling for prompt power is unlikely to be justified for severe-accident conditions (see conclusions to the nuclear-heat model review). Consideration should be given to reintroducing the modeling of fuel slumping into a severely ballooned region (e.g., strains >~50%). If scenarios involving high burnup fuel are to be analyzed, the model will need to be revised to deal with the radial peaking factor problem.

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E.6. Heat-Conduction Model

E.6.1. Model Description and Pedigree

A finite element approach is used to solve one-dimensional transient conduction equations in fuel rods and structures (Refs. E.6-1 to E.6-3). Both plate-type and cylindrical geometries are analyzed. Any of the three types of boundary conditions can be applied. In obtaining the solution for the temperature field, the Galerkin method of weighted residuals and temperature-dependent effective thermal properties are used. The volumetric heat-generation rate is also included in the model. The model is only applicable to intact structures. Heat-transfer coefficients at the structure surfaces are obtained from the RELAP5 code. Recently, the model has been upgraded to two dimensions. An alternating-direction implicit method scheme is used to solve for the temperatures in axial and radial directions.

E.6.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The model appears to be physically reasonable as long as the boundary conditions are appropriately applied. Unfortunately, the correlations or models used for the convective heat-transfer coefficients could not be reviewed.

E.6.3. Implementation Within the Code

The geometry, physical dimensions, and material of the structures are specified by the user. The package interacts with RELAP5 for boundary conditions.

E.6.4. Results of Model Sensitivity Studies

None were found.

E.6.5. Results of Benchmarking/Validation Studies

Simple steady-state and transient solutions obtained from the numerical calculations are compared with analytical solutions. Such an effort only gives confidence in numerics but tells nothing about the appropriateness of the chosen boundary conditions in a physical situation.

E.6.6. Identified Deficiencies and Options for Model Improvements

It has not been possible to assess the appropriateness of the boundary conditions (convective and radiative) that are applied to solve for the temperature field. No information could be found with respect to the robustness of the method to handle very rapid transients involving chemical reactions. The method does not appear to include failed and partially molten structures in regions where debris beds are formed.

E.6.7. Importance of Model to Prediction of Dominant Phenomena

The temperature of cladding material, UO_2, control rods, and structures is a key parameter that will determine the core-degradation process.

E.6.8. Technical Adequacy of Model

The model appears to be adequate for steady-state and slow transients. However, significant concern exists as to the type of boundary conditions that are applied during the core-degradation process.

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E.6-3. L. Siefken, EG&G Idaho, Inc., personal communication (September 9, 1992).

E.7. Cladding-Deformation Models

E.7.1. Model Description and Pedigree

The code incorporates a model for the ballooning of zircaloy cladding very similar to that found in the FRAP-T6 code (Ref. E.7-1). At low strains, an axisymmetric model is used on the basis of the standard Lame-Clapeyron equations for stresses in a thick-walled tube, von Mises definitions of equivalent stress and equivalent strain, and Prandtl-Reuss flow rules for plastic strain. The anisotropy of alpha-phase zircaloy is taken into account in the calculation of plastic strain and thermal expansion. The physical properties for the modeling of the cladding strain are taken from MATPRO. Deformation is assumed to stop at rod-to-rod contact (~33% strain). A limit strain may be set by the user (no implication of cladding rupture).

A localized nonaxisymmetric model is used at high strains, with a prescribed variation of fuel surface temperature in the axial and circumferential directions. Axial, as well as circumferential, curvatures of the cladding are considered. The localized model is applied to the node where the hoop strain is a maximum (greater than 5%) and where the axial temperature gradient exceeds a preset limit; this criterion can be overridden with a user-defined strain. Flow-area reduction is calculated with the FAR1 probabilistic model, which takes into account noncoplanarity of strain among rods.

The rupture criterion can be either a user-defined strain (e.g., obtained from a detailed ballooning analysis with another code or from relevant experimental data) or the temperature-dependent stress criterion from MATPRO. The MATPRO correlation is an empirical relationship derived from analysis of single-rod, tube-burst data.

The models for cladding deformation are described in Refs. E.7-2 to E.7-4 and Presentation 1-8 of Ref. E.7-5.

E.7.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The modeling is generally based on well-established, reasonable, theoretical principles for the deformation of zircaloy cladding in a nonoxidizing atmosphere. The modeling of rupture is empirical; however, it is judged reasonable in the context of a severe-accident code. The treatment of flow blockage is empirical; however, it is near state of the art and judged reasonable. However, there is no first-order modeling available for the treatment of the effect of oxidation on cladding strain.

The physical processes involved in cladding deformation are now well understood (mainly from design basis studies).

E.7.3. Implementation Within the Code

The deformation models are called from the fuel rod and fuel-rod simulator models taking as input the previous cladding state, cladding temperature, internal pressure, and external pressure at each timestep. The new cladding state appropriate to the end of the timestep and any indication of rupture, etc., is returned. The method of implementation is judged reasonable.

E.7.4. Results of Model Sensitivity Studies

None were found.

E.7.5. Results of Benchmarking/Validation Studies

The ballooning model has been validated in various calculations (Presentations 1-7 and 1-8 of Refs. E.7-5 to E.7-8) of tests in experimental facilities such as LOFT, PBF, and CORA. It is stated (e.g., in Presentation 1-7 of Ref. E-5) that ballooning temperatures are typically within experimental variations.

UK analysis of CORA experiments (Ref. E.7-8) showed that the effects of ballooning on core degradation could be well simulated in that case; e.g., the importance of double-sided oxidation in advancing the timing of the oxidation excursion was demonstrated in calculations for CORA-15, where the internal rod pressure was high. However, ballooning was predicted unexpectedly for CORA tests with a very low cladding pressure differential, which was inconsistent with the data at temperatures in the high alpha phase (700–800°C).

Furthermore, in calculations for ISP-28 (PHEBUS B9+), ballooning inconsistent with the data was again predicted (Ref. E.7-9); in the experiment, ballooning was prevented by the use of a fusible plug; however, the code calculated ballooning at temperatures below that at which the plug fused. In the UK submission (with MOD2.5), it was found necessary to turn the ballooning model off completely to obtain a sensible calculation for the rest of the transient.

E.7.6. Identified Deficiencies and Options for Model Improvements

Limitations of the model (Refs. E.7-7, E.7-8, and Presentations 1-8 and 1-18 of Ref. E.7-5) are that (1) the effect of oxidation in strengthening the cladding and limiting the rupture strain is not explicitly modeled (important if ballooning takes place over ~950°C), (2) the constitutive equation does not correctly account for strain rate, and (3) the cladding strength model is based on early data so that the strength of modern cladding is underestimated (important in the high alpha-phase region, particularly at low stresses). These limitations are consistent with the reviewer's own experience with the code, some of which is discussed above.

In some cases, these deficiencies can be avoided by choosing suitable input data; for example, the oxidation effects can be approximately simulated through the use of a limit strain and/or a user-defined rupture strain, and inappropriate low-temperature ballooning can be prevented by reducing the helium inventory or even turning the ballooning model off altogether. Obviously, this is not an ideal situation.

It is recommended that the modeling be improved in the above areas. In the meantime, the limitations noted and ways of avoiding them should be indicated in the user documentation.

E.7.7. Importance of Model to Prediction of Dominant Phenomena

Cladding-deformation models are needed to determine (1) the reduction in flow area due to ballooning and its effect on bypass flows, (2) the increase in cladding surface area available for oxidation (and also for heat transfer to the coolant), and (3) the timing of early release of the gap fission-product inventory through any ballooning-induced rupture. Such a rupture also allows an ingress of steam to the inner surface of the cladding and thus increases the total rate of oxidation.

These effects are judged to be of moderate significance in a severe-accident analysis.

E.7.8. Technical Adequacy of Model

Much of the modeling is first order; however, the absence of modeling for the effects of oxidation on deformation and the calculation of unreasonably large cladding strains when cladding stresses are low are important deficiencies. The overall treatment is therefore judged to be incomplete. However, because a knowledgeable user may limit the impact of these deficiencies by a suitable choice of parametric input, the model is assigned to a zeroeth order category.

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E.8. Fuel-Rod, Internal-Gas Pressure

E.8.1. Model Description and Pedigree

The fuel-rod, internal-gas pressure is calculated as a function of temperature and gas volume using

$$P_i = N_m R/\Sigma(V_i/T_i) \quad ,$$

the ideal gas law, where P is the fuel-rod, internal-gas pressure, N_m is the moles of gas in void volumes, R is the universal gas constant, V_i is the i-th volume, and T_i is the gas temperature in the i-th volume.

The different volumes considered include:

1. Fuel-rod plenum volume;
2. Fuel-void volumes (cracks, voids, etc.);
3. Gap volume; and
4. Additional gap volume due to cladding ballooning.

The gap and void gas temperature is assumed to be at the average fuel-rod temperature in the i-th axial node, while the gas temperature in the fuel-rod plenum is assumed to be the coolant temperature (+6 K) at the top of the rod, which is consistent with the FRAP-T6 model. This last assumption is not justified.

The hot void volumes are calculated either by interpolation of user-specified tables providing relative void volumes as functions of average fuel temperatures or based on PWR and BWR-specific correlations to the FRAPCON-2 code calculations. Using the former option (tabular inputs) to define external volumes with prescribed temperature histories, the internal plenum is then ignored for that component (Ref. E.8-1). This option was introduced to model ballooning in the CORA facility, where most of the gas inventory resides in pressure gauges and capillary tubes outside the heated section and where the use of a conventional option gives gross errors in the internal pressure. This option is also useful for sensitivity studies.

These correlations have been developed to account for the impact of fuel burnup and temperature on fuel-void volumes. The FRAPCON-2 calculations were performed for a constant system pressure of 15.51×10^6 N/m^2 and 7.14×10^6 N/m^2 for PWRs and BWRs, respectively, over the following temperature and burnup range:

PWR:	$500 \leq$ Temp. ≤ 1100 K
	$0.1 \leq$ Burnup ≤ 30 MWd/kg-U
BWR:	$500 \leq$ Temp. ≤ 1200 K
	$0.1 \leq$ Burnup ≤ 30 MWd/kg-U

Corrections to the calculated void volumes due to variations from standard BWR and PWR designs are made by assuming that variations in the area are independent of the inner cladding radius and that variations in area are dependent only on the temperature difference between hot and cold fuel rods. Furthermore, it is assumed that the relative fuel-void volume calculated from empirical correlations holds for varying as-fabricated geometries.

E.8.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The assumption of setting the gas temperature in the fuel-rod plenum to that of the coolant temperature at the top of the rod (+6 K) is arbitrary and easily can be remedied. The approach of correlating void volumes to FRAPCON-2 calculations is not based on first principals, although it is acceptable for the intended applications of SCDAP/RELAP5, especially when considering the larger phenomenological uncertainties inherent in the simulation of other more important issues relevant to severe accidents. The corrections to nonstandard designs are not unreasonable; however, no benchmarks are presented to confirm technical acceptability.

E.8.3. Implementation Within the Code

It is difficult to determine the adequacy of model implementation within the SCDAP/RELAP5 code architecture from the documentation provided. The interface between FPRESS (fuel-rod, internal-gas-pressure model) and the other interacting subroutines in SCDAP/RELAP5 cannot conclusively be established at this time.

E.8.4. Results of Model Sensitivity Studies

Sensitivity studies using the SCDAP/RELAP5 fuel-rod, internal-gas-pressure model have not been performed (or have been performed but have not been reported to the Peer Review Committee).

E.8.5. Results of Benchmarking/Validation Studies

Comparisons of FPRESS and FRAPCON-2 results are presented. These comparisons only reveal the quality of the FPRESS fits to FRAPCON-2 calculations. This, of course, cannot be considered as validation of the FPRESS approach.

E.8.6. Identified Deficiencies and Options for Model Improvements

The documentation does not describe the procedure for calculating the number of moles of fission gas. Model parameters are for initial coolant pressure greater than the internal rod pressure. The impact of depressurization on the calculated correction factor is not described. Feedback between expansion of fuel and fuel porosity is not considered, and the correlations are only valid up to 1100–1200 K. It is not clear if vapor pressure curves for control-rod materials have also been developed. A justification for assuming a uniform rod temperature for temperatures above 750 K is not provided.

E.8.7. Importance of Model to Prediction of Dominant Phenomena

The fuel-rod, internal-gas pressure has a direct impact on the prediction of fuel-cladding deformation and failure, initiation and rate of fission gas release, and component heat transfer.

The physics of the fuel-rod, internal-gas-pressure model is relatively well understood; however, uncertainties exist regarding the impact of burnup and temperature on fuel behavior.

E.8.8. Technical Adequacy of Model

The present model as incorporated in the SCDAP/RELAP5 is at most zeroeth order.

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E.9. Liquefaction, Flow, and Solidification

E.9.1. Model Description and Pedigree

The fuel liquefaction, relocation, and solidification processes are calculated based on the following steps (Refs. E.9-1 to E.9-3):

1. Fuel- and zirconium-cladding liquefaction,
2. Cladding-oxide-shell failure,
3. Relocation and solidification of liquefied Zr-U-O from the breached element, and
4. Reliquefaction of previously solidified material.

It is assumed that liquefied zirconium will instantly dissolve the UO_2 fuel that it contacts. The dissolution rate is calculated from an empirical relation attributed to Hofmann et al. (Ref. E.9-4), which is based on steady-state uniform heating of ZrO_2-Zr-UO_2; however, the calculated amount of fuel dissolution is limited to that based on an equilibrium ternary-phase diagram. The effect of heat of formation is not included. The limit on the maximum amount of liquefaction is set to the solidus temperature because of a better agreement with experimental data. The breach of the cladding surface ZrO_2 layer will lead to relocation of liquefied fuel and cladding over the fuel rod's outer surface.

The liquefied fuel-cladding mixture is assumed to spill out of the breached cladding and flow downward on the same (failed) fuel pin. The flowing mixture takes the form of a slug ring with an initial thickness assumed to equal the average thickness of the *in situ* liquefied mixture. The spilling effects of the liquefied fuel-cladding mixture on the surrounding rods is not modeled.

The slug-ring velocity is calculated by numerical integration of an equation of motion for the liquefied mixture. A steady-state pipe friction factor is used to calculate the liquid slug-ring drag as a function of flow-regime (turbulent and laminar) Reynolds numbers.

A slug-ring drop distance is determined from the calculated slug-ring velocity; however, if a grid spacer is encountered, the slug-ring velocity is arbitrarily set to zero. This arbitrary assumption has been found to dominate the predicted melt relocation behavior. In ISP-28 (PHEBUS B9+), the grid spacers relocate themselves long before the melt arrives. In

CORA, on the other hand, the grid spacers dominate matters in the PWR experiments, and a grid spacer model is essential if blockages are to be accurately calculated.

Heat transfer from the flowing slug ring to the cladding surface is modeled by convection based on steady-state, heat-transfer coefficients for both laminar and turbulent flow. The convection heat-transfer coefficient for turbulent flow is calculated based on the Reynolds analogy.

The rate of formation of a solid crust on the outside surface of cladding is calculated through a transient-energy equation that balances the heat loss to the cladding via convection and heat gained from the liquid mixture via a set of parametric equations to simulate molecular conduction and turbulence effects.

If the slug-ring contact is much shorter than the temperature front propagation time into the cladding, the heat transfer from the crust of solidified Zr-U-O into the fuel-rod cladding is calculated assuming the cladding is a semi-infinite medium subject to a uniform temperature boundary. Otherwise, the cladding or cladding-oxide midplane temperature is calculated through a lumped-parameter, transient-energy equation, with conduction across the cladding and/or the oxide layer. The lumped mass consists of the entire zirconium cladding mass in a given axial mesh plus one-half of the adjacent oxide layer.

The previously solidified crust is allowed to remelt upon reaching the melting temperature of the Zr-U-O mixture. The reliquefied crust is assumed to flow downward because of gravity and flow into the closest axial mesh with a temperature below the melting temperature of the Zr-U-O mixture; the film then is assumed to refreeze.

E.9.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

In general, the LIQSOL models are not totally unreasonable; however, a number of unsupported approximations and assumptions limit the code applicability to a mechanistic prediction of severe accidents as summarized below:

1. The assumption of intimate contact between fuel and cladding material dictates eutectic dissolution as the only mode of fuel and clad failure.

2. The LIQSOL oxide-shell failure criteria are based on achieving either a user-specified failure temperature and cladding-oxide fraction or an oxide-shell melting temperature. Mechanistic models are not included to account for possible effects of internal pressure (especially for low-pressure accident sequences) and structural weakening of the fuel rods. The amount of relocation calculated can be very sensitive to the choice of oxide-shell breach temperature. No single value gives the best agreement in all experiments. These parametric fuel-rod failure criteria are expected to dominate the SCDAP/RELAP5 predictions of fuel failure conditions during severe accidents. This is probably the most important uncertainty associated with the early-interval melt progression and fission-product release, where the SCDAP/RELAP5 models are inadequate.

3. In a ballooned geometry, some of the melt may run down inside the balloon rather than form a crust on the outer surface. Thus, the code will overpredict blockage in such circumstances because it is assumed that all melt will freeze on the outside. Furthermore, the UO_2/zircaloy reaction will be inhibited, at least early on, by the enlarged pellet/clad gap and the internal oxidation of the cladding by steam. These effects are not treated. However, the effects of ballooning in CORA-15 (limiting damage to the top half of the bundle and advancing the oxidation excursion) were well predicted.

4. There is no proper coupling between the models for the reactions between zircaloy/steam on the outside of the cladding and zircaloy/UO_2 on the inside. A coupled model should be able to take into account dissolution of the oxide shell from the inside cladding surface, the oxygen availability at the outer cladding surface, etc.

5. The Zr-U-O mixture relocation is based on a gravity-driven mixture slug-ring flow over the outside surface of the same failed fuel rod.

6. Because Zr-U-O slumping is treated on a node-by-node basis, strong nodalization dependencies are expected.

7. The heat-transfer models for slug-ring relocation are very deficient. Only conduction (steady state) from the flowing Zr-U-O mixture to the solid crust

has been considered. Heat transfer by convection to the coolant (steam, and in the case of reflood, water) and radiation to the coolant and surrounding structures have been neglected. Large convection and radiation fluxes could potentially lead to rapid freezing of the moving Zr-U-O mixture.

8. Rivulet rather than slug flow (referred to as film flow in SCDAP/RELAP5 documents) has been established experimentally as the dominant relocation process (seen especially clearly in the CORA tests). Again, this process is not modeled.

9. Heat generation within the relocating material is neglected. Furthermore, additional zirconium oxidation during Zr-U-O relocation is not modeled. Surface renewal processes could potentially enhance metal oxidation, and thereby the relocation process, due to additional heat generation.

10. The effect of grid spacers on the slug-ring relocation is based on an arbitrary assumption. This usually has been found to dominate the SCDAP/RELAP5 predicted relocation.

E.9.3. Implementation Within the Code

The computational procedures incorporated into the LIQSOL submodel are clearly outlined as part of Ref. E.9-1; however, LIQSOL implementation within the SCDAP or SCDAP/RELAP5 program architecture cannot be established based on the documentation currently available.

E.9.4. Results of Model Sensitivity Studies

Sensitivity results showing how the calculated results will be affected by the choice of the molten mixture relocation and thermal transport models have not been reported.

E.9.5. Results of Benchmarking/Validation Studies

Extensive calculations have been reported where SCDAP/RELAP5 results have been compared with several integral severe fuel-damage experiments. The reported studies include comparisons of calculated and measured temperatures, hydrogen generation, and other global quantities, which are not indicative of the adequacy of the SCDAP/RELAP5 meltdown and relocation models.

The SCDAP/RELAP5 simulation of ISP-28 (Refs. E.9-5 and E.9-6) showed a gross overprediction of the amount of eutectic formation and thus the amount of blockage in a transient where the time at temperatures over ~1500°C was ~6000 s (in a helium atmosphere). By contrast, in CORA facility simulations where the corresponding time was ~1000 s, good agreement was obtained. In both cases, predicted temperatures were in quite good agreement with the experimental measurements. This indicates that there is some problem with the dissolution model—either the Zr-U-O phase diagram is improperly understood or modeled or there is some mechanism that is not being treated (e.g., slurrification of fuel by attack of liquid zircaloy along UO_2 grain boundaries, forming an inhomogeneous mixture or the effect of a reduced oxygen potential at the cladding outer surface).

E.9.6. Identified Deficiencies and Options for Model Improvements

The present fuel liquefaction, relocation, and solidification models suffer from several modeling and phenomenological inadequacies listed previously. In general, many of the stated assumptions and selected models are without adequate physical basis and are sometimes arbitrary. The choice of these models has sometimes forced more complications than are actually necessary.

E.9.7. Importance of Model to Prediction of Dominant Phenomena

This is one of the most important models impacting the prognostication of the early-phase, fuel-cladding behavior, melt relocation, zirconium oxidation (during relocation), and melt freezing and blockage formation during severe-accident conditions.

Uncertainties exist in understanding the physical and chemical phenomena related to fuel-cladding meltdown, relocation, oxidation, crust formation, crust liquefaction and resolidification, and blockage formation.

E.9.8. Technical Adequacy of Model

The present fuel liquefaction, relocation, and solidification models are not mechanistic; they are for the most part parametric. The documentation of the models leaves much to be desired. Inconsistencies, typographical errors, and lack of clarity exist in the various documents (Refs. E.9-1 to E.9-3). The models are more appropriate for parametric studies useful to PRAs than for application to mechanistic predictions and accident management studies.

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E.10. Fuel-Fission-Product Release

E.10.1. Model Description and Pedigree

Release Model for Intact Fuel. The fuel-fission-product-release model is based on the PARAGRASS/FASTGRASS and CORSOR-M computer codes.

The release of volatile noble gases (xenon and krypton), cesium, iodine, and tellurium are based on the PARAGRASS model, whereas the release of semirefractory and refractory species is based on the CORSOR-M model.

The PARAGRASS model was developed for noble-gas release from fuel. In the version integrated into SCDAP/RELAP5, the release of volatiles like iodine, cesium, and tellurium is accomplished by a combination of diffusion through the fuel matrix and noble-gas bubble capture of individual atoms. In the gas bubble capture technique, a chemical equilibrium is calculated. The gas diffusion (to grain boundaries) is treated via the solution of time-dependent diffusion equations. The modeling of other processes (e.g., gas atom re-solution, coalescence, and trapping by gas bubbles) impacting fission-product gas behavior is accomplished through the solution of a second-order, time-dependent balance equation.

Following their transport into the gap (as predicted by PARAGRASS), the release of fission products to the coolant is based on the following approach:

Noble gases are released instantaneously upon cladding failure. Additional releases of noble gases to the coolant are defined as the PARAGRASS-calculated, noble-gas releases into the gap region.

Cesium and iodine are released based on a model that includes (1) a burst component that accompanies the initial cladding breach and depressurization, and (2) a diffusion component that describes the subsequent time-dependent releases of the remaining iodine and cesium species. These two components are assumed to be independent.

The model assumes all of the iodine will react with available cesium to form CsI, with any leftover iodine being released as I_2 or any leftover cesium reacting with water to form CsOH and hydrogen.

The release of fission-product tellurium appears to be identical to that of iodine and cesium; however, when zirconium cladding is less than 90% oxidized, the tellurium release is reduced to 1/40 of the calculated release (based on ORNL experimental observations). Otherwise, the tellurium release is unimpeded.

The release of nonvolatile species is based on the first-order-rate equations of CORSOR-M with Arrhenius-type rate constants fitted to the fission-product-release data.

A simple mass-transfer-driven, first-order release model is used to calculate the release of tin. In addition, a simple empirical approach (a function of temperature only) is used to calculate the control-rod release of silver and cadmium.

<u>Release Model During UO_2 Liquefaction and Fragmentation.</u> The release of volatile fission products of xenon, krypton, cesium, and iodine during UO_2 liquefaction and fragmentation is calculated by assuming (1) an instantaneous release into the gap from all calculated liquefied fuel, and (2) an instantaneous release of calculated accumulated grain boundary material following a calculated fuel-fragmentation process.

Subsequent release within the rubble bed is stated to be controlled by the intragrain processes; however, it does not appear from the documents provided that any releases from the rubble bed are actually calculated.

The release of other less-volatile, fission-product species is assumed to be unaffected by UO_2 liquefaction and fragmentation processes.

The fuel-fission-product-release model also calculates the enthalpy of released gases.

E.10.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

<u>Intact Fuel.</u> The present model for the release of fission products before fuel liquefaction, fragmentation, and slump is based on an adaptation of a first-order model developed as part of the GRASS series of computer codes at Argonne National Laboratory (ANL). The PARAGRASS models are only applicable to the prediction of volatile fission-product releases, while the CORSOR-M model is used to calculate the release of less-volatile species.

The burst release model, which is based on a relatively outdated empirical approach developed at ORNL for analysis of LOCAs for typical PWRs, is used as part of the present model. In addition, a longer-term diffusion release is included using an empirical first-order release model. The experiments on which these models are based used short sections of fuel rods with low gap inventories. These models are used for all temperatures in SCDAP, whereas the adjustable parameters are based on experimental data in the temperature range of 970–1170 K. In general, these models have been found to represent the data on which they were based within a factor of three.

The CORSOR-M code parameters (pre-exponential and activation energy terms) are based on experimental data, except for refractory species, where the heat vaporization has been substituted for the so-called activation energy. CORSOR-M is only used in SCDAP/RELAP5 to predict the release of less-volatile fission products for which CORSOR-M applicability is questionable. In addition, the influence of changes in surface-to-volume ratios during meltdown are not included.

The control-rod release is based on the purely empirical, nonmechanistic relations of the CORSOR code (Ref. E.10-1).

In the absence of eutectic interactions, release of volatile fission products will be enhanced as result of continued fuel heatup. Furthermore, the present model does not appear to treat melt releases (both during melting and relocation). Therefore, these inadequacies become more consequential.

During UO_2 Liquefaction and Fragmentation. The liquefaction and fragmentation release model suffers from similar shortcomings. There are no models currently present in the code to calculate releases within a rubble bed configuration.

E.10.3. Implementation Within the Code

Changes have been made to the models and numerics of the original PARAGRASS model that are not well documented as part of the SCDAP/RELAP5 MOD3 Manual (Ref. E.10-2). The delivered PARAGRASS model had errors in the chemical equilibrium package, which were modified as directed by ANL. The basic concept of two fission products interacting within the fuel matrix has been abandoned by ANL. The version of the model that has been integrated does not reflect the state of the art.

The interface between the fission-product release, decay heat, and fission-product transport and deposition models is not very clearly defined in the current documentation.

E.10.4. Results of Model Sensitivity Studies

Sensitivity studies using the SCDAP/RELAP5 fuel-fission-product-release model have not been performed (or have been performed but have not been reported to the Peer Review Committee). This model is always active during the SCDAP/RELAP5 calculations; nevertheless, very little attempt has been made to determine phenomenological sensitivities.

E.10.5. Results of Benchmarking/Validation Studies

Some comparisons to the PBF/SFD test results have been reported. However, the present models have not been validated with experimental data (either separate effects or integral data).

E.10.6. Identified Deficiencies and Options for Model Improvements

The present SCDAP/RELAP5 fission-product-release model for intact and liquefied fuel is based on outdated versions of the PARAGRASS/FASTGRASS and CORSOR-M computer codes. Models do not appear to have been included for fission-product release resulting from fuel-fragmentation and rapid oxidation processes. In addition, the model for control-rod release of silver and cadmium is based on a temperature fit to experimental data.

Little effort is being expended by INEL in improving or assessing the fission-product-release (and transport) models within the code; therefore, a technical basis for the adequacy of these models has not yet been established.

The fission-product-release models could significantly be improved by using both an up-to-date version of the PARAGRASS computer code that is under development at ANL and the recently completed CORSOR-Booth model developed at ORNL.

E.10.7. Importance of Model to Prediction of Dominant Phenomena

The prediction of release quantity, state, chemical form, and transient characteristics of fission products is one of the most important issues for accident management studies. It is the Committee's understanding that other codes, notably VICTORIA, will be used in conjunction with SCDAP/RELAP5 to address issues where the magnitude of the source term is particularly important for analysis.

The detailed mechanisms of fission-product-release behavior during various phases of a severe accident are not well understood. A relatively large experimental database related to intact (undamaged) fuel, heatup, and meltdown phases of severe accidents exist. However, theoretical models are still relatively immature and are often tailored to match experimental observations.

E.10.8. Technical Adequacy of Model

Overall, the fission-product modeling approach is excessively unbalanced. The model to predict release from intact fuel is fairly detailed. It accounts for several important mechanisms of lattice migration and transport. On the other hand, the remaining models are for the most part highly empirical and somewhat outdated.

The documentation also leaves a lot to be desired. Many typographical errors exist in the equations, and most of the support information is not readily accessible. For instance, M_B should be replaced by M, and the gap inventory in Eqs. (3-143) and (3-153) of Ref. E.10-2 does not appear to be correct. References are also made to the wrong equations.

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E.11. Control-Rod And Core Structure

E.11.1. Model Description and Pedigree

Models are provided (Refs. E.11-1 and E.11-2) for PWR control rods, BWR control blades, flow shrouds, and spacer grids.

A PWR control rod is modeled as a multilayer cylinder, using the same treatment of heat conduction as for fuel rods. Oxidation of the zircaloy guide tubes is considered, and the hydrogen production and heat generation are taken into account. User-defined gamma heating is modeled. Relocation of the stainless steel, unoxidized zircaloy, and molten (Ag-In-Cd) absorber material is assumed to take place when the stainless steel has reached its melting temperature [taken as 1700 K, hardwired in the code; however, some studies have indicated failure temperatures as low as 1500 K in low-pressure transients due to eutectic formation (see Ref. E.11-3)]. Further slumpings above the breach location occur at the (Ag-In-Cd) melt temperature (taken as 1100 K). The molten Ag-In-Cd freezes when it reaches an elevation at a temperature 200 K less than its solidus temperature. Molten stainless steel and zircaloy are assumed to relocate inside the zircaloy oxide shell. There is no calculation of the contribution of relocated control-rod materials to the blockage formation.

The complex structure for a BWR control rod is modeled with a cylindrical geometry approximation. An equivalent stainless-steel outer radius and thickness are used to represent the tube cladding's outer radius, thickness, and associated portion of the stainless-steel sheath. The outer radius of the B_4C is also input. The model uses a one-dimensional radial conduction solution (axial conduction is not modeled) and treats B_4C and stainless-steel oxidation using empirical correlations that are parabolic for stainless steel (correlation parameters are hardwired into an input routine, not obtained from MATPRO). Gamma heating is also considered via input data. Relocation is treated using a model for incompressible viscous flow over a cylindrical geometry. This model is being replaced by new models under development at ORNL, as described in Section E.26.1.

No model is provided for zircaloy-clad B_4C/Al_2O_3 burnable poison rods found in some PWRs (including TMI-2). Alumina is incompatible with zircaloy above ~1620 K. It is unclear, however, what gross effects in severe-accident transients would result from the presence of these control rods.

A general-purpose model is provided for flow shrouds, etc., where the material composition is user defined. There is an option for user-specified material properties, e.g., thermal insulators as used in experimental facilities. Zircaloy oxidation for fuel rods and relocation of unoxidized zircaloy are considered (the material relocates until it reaches a surface at a temperature 200 K less than its melting point). No significant deficiencies have been identified. Problems have been identified (Ref. E.11-4) in the modeling of thermal radiation between nested shroud components, but these are more likely caused by logical errors in the radiation model rather than faults in the shroud model, per se. One user (Ref. E.11-5) found that zircaloy in a flow shroud failed to slump as expected.

The simple model provided for spacer grids (Inconel and zircaloy) considers them only (while intact) as an obstruction for flowing melt or debris regions. Only one type of grid is allowed in each analysis. The grid melt temperatures (2150 K for zircaloy and 1670 K for Inconel) are hardwired in the code (in an input routine). There is no treatment of chemical reactions involving grid materials, and the metal inventory in the grids is not considered. The grid removal model is described separately in Ref. E.11-6.

Some advances in the modeling have been reported recently. In a more detailed grid model (Ref. E.11-7), the material inventory in grids is taken into account, there is a more detailed heat-transfer model, the composition of the grids can be different at different axial elevations, and the zircaloy/Inconel interaction is treated. A brief assessment of this model is given in Section E.26.2. Similar developments are planned for control-rod models in 1992/93.

E.11.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics involved in these areas is now reasonably well known. Kinetics data are available for important eutectic reactions such as Ag-In-Cd/zircaloy, B_4C/stainless steel etc., and the effect of zircaloy oxide shells (such as would be formed during normal reactor operation) is also quantified. However, this state of knowledge is not currently reflected in the modeling in the code, and the treatment of control rods and grids (as described in the main references) is parametric at best. The newly developed model for spacer grids appears to be first order, but more evidence of its performance is needed before it can be properly assessed.

E.11.3. Implementation Within the Code

The models are called as required from the intact core module, etc., in a reasonably logical way. There are some inconsistencies in the ways in which material data used in the modeling are defined (see above); some tidying up in this area would be appropriate.

E.11.4. Results of Model Sensitivity Studies

None were found.

E.11.5. Results of Benchmarking/Validation Studies

Validation of these models has taken place in the context of analysis of experiments such as those in the CORA and PBF series as reported, for example, in Refs. E.11-8 and E.11-9. In general, the timing of key events (i.e., control-rod and channel-box failure) is reasonably well predicted. However, the need for materials interactions models is demonstrated, particularly for the BWR case. The effect of the intact grids in holding up melt flow in the CORA tests was well simulated by the code (Ref. E.11-9).

E.11.6. Identified Deficiencies and Options for Model Improvements

PWR control-rod model. The basic structure of the control rod is adequately treated; however, degradation processes involving control-rod material are inadequately modeled. Deficiencies identified (Presentations 1-9 and 1-18 of Ref. E.11-10) are that the radial spreading of absorber material is not modeled, the Ag-In-Cd eutectic reaction with zircaloy (Ref. E.11-11), which promotes early fuel relocation, is not treated, and there is no mechanistic treatment of the hydrodynamics and heat transfer of the slumping material. These deficiencies need to be addressed.

BWR control-rod model. Similar remarks apply to the BWR control-rod model. As with the PWR absorber model, there is no treatment (Presentations 1-9 and 1-18 of Ref. E.11-10) of radial spreading of melt or of the eutectic reactions involving the control-rod materials, which have a significant effect (Ref. E.11-12) in initiating melt progression (e.g., B_4C/stainless steel and B_4C/zircaloy). These reactions, especially the former, remove B_4C from the system and prevent the formation of HBO_2, which could react to change the fission-product species CsOH and CsI (Ref. E.11-3). Detailed studies under way at ORNL (Ref. E.11-13) may remove some deficiencies in the areas of BWR-specific modeling.

Flow shroud model. The model is basically sufficient for its intended applications. The operation of the zircaloy slumping model should be checked.

Grid model. In the model described in the main references, there is no treatment of the reaction of Inconel with zircaloy cladding (fuel or control rod); this reaction has been shown (Ref. E.11-14) to promote early relocation because of the formation of low-melting-point eutectics. However, the recently developed, more detailed grid model takes this reaction into account; the temperature distribution in the grid now also is calculated. Assessment of this new model against experimental data is now required.

General. Sensitivity studies on relocation temperatures, etc. (e.g., to simulate early rod failure due to eutectic formation), are only possible by altering the coding, not by varying input data.

Work in progress may aid in remedying the identified deficiencies in the general area of control-rod and structure models.

E.11.7. Importance of Model to Prediction of Dominant Phenomena

The presence of absorber material can significantly affect the nature and timing of both PWR and BWR core-melt progression, and therefore, models for the relevant types of control rod should be included. Grid spacers can have an important effect on blockage formation, and therefore, modeling of these structures is needed, as well. The flow-shroud model is essential to model experimental facilities where such structures are often present; they define the radial boundary conditions and also often contribute significantly to the hydrogen generation.

E.11.8. Technical Adequacy of Model

The control-rod models currently describe their basic geometry and thermophysical behavior of these core components, but do not treat some important chemical/eutectic reactions involving their constituents. The grid model described in the main documentation and assessed against experiments is parametric in nature; a newly developed model appears to be first order but cannot yet be formally categorized, owing to a lack of assessment against experiments. The shroud model is adequate for its purpose.

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E.12. Radiation Heat-Transfer Model

E.12.1. Model Description and Pedigree

The thermal radiation heat-transfer model analyzes radiation exchange between various components in the core, as well as between components and the coolant. The coolant is considered to absorb and emit radiation, but the scattering from aerosol particles is neglected. The analysis is based on the zonal approximation of radiation heat transfer, which is well established in the literature and is similar to the analyses that were developed earlier for nuclear reactor applications with intact core geometry.

The model calculates the net (leaving minus incident) radiation heat-transfer rate at a component surface and the net (emitted minus absorbed) radiation heat-exchange rate by the coolant zone (finite volume). These rates are then used in thermal boundary conditions of any vessel component such as the fuel and control rods or flow shrouds.

The configuration (view, angle) factors of the intact geometry are calculated from analytical expressions given in the documents for cylindrical rods of infinite length. This approximation appears to be well justified. Dynamic adjustment of these factors is not attempted, but disappearance of components for a highly degraded core is considered. An assumption is made in the calculations that if a shroud is present around a rod bundle, only the outermost rows and columns of the rods are considered to exchange radiant energy with the shroud.

The emissivities of surfaces are obtained from MATPRO (Ref. E.12-1) and are computed as a function of temperature. The variation of emissivity of zircaloy with a zirconium oxide layer thickness is considered. Because a gray radiation model is used in the analysis of radiation heat exchange, the surface emissivity equals absorptivity. The variation of steel emissivity with oxidation or any difference in emissivity between wet and dry surfaces is not considered.

The emittance and absorptance of the gaseous coolant is obtained in a manner similar to that used by TRAC-BD1. Only steam is considered in calculating the emittance of the coolant, and the gaseous nonradiating species such as hydrogen and aerosols are ignored. The absorption spectrum of water vapor is considered to consist of six major absorption bands.

The dependence of the spectral absorption coefficient with temperature is accounted for, but its dependence on pressure is ignored.

The integration over the spectrum to obtain a mean-wavelength independent emittance or absorptance is carried out by ignoring the detailed band shape and using a "top hat" (rectangular) band model to approximate the true shape of the band. The mean beam length needed to evaluate the spectral emittance is calculated approximately. First, the path length between any two rods is obtained; the average path length between two component groups then is calculated by weighing with respect to the corresponding configuration factor. This is equivalent to assuming that the coolant is not capable of absorbing and emitting thermal radiation. There is neither discussion of nor references cited to justify the use of the approach for calculating the mean beam length.

E.12.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The calculation of radiation heat transfer can be very time consuming because of the long-range and spectral nature of the phenomena in complex and changing geometries such as the core and upper plenum. The radiation heat-transfer model in SCDAP/RELAP5 represents a reasonable compromise between reality and computational effort in predicting dominant phenomena. In developing the model, several assumptions are made: (1) axial radiation heat transfer is negligible, (2) all surfaces are gray and diffuse emitters, and (3) the coolant is a gray radiator. Of the three assumptions, Items 2 and 3 are the most serious. Because the transmittance of the coolant (τ_g) is assumed equal to one minus the coolant emittance (ε_g); this assumption models the coolant as a gray gas.

The assumption that water vapor has some strong absorption bands, and the gray gas assumption cause the net radiative heat-transfer rate to steam to be underestimated and the net rate to, for example, the shroud wall, to be overestimated. To avoid using a gray gas assumption, Edwards (Ref. E.12-2) has suggested using a gas transmittance of 1 between the bands and a value of τ_g within the bands; however, this very simple but more realistic approach of calculating radiation heat transfer in the core and upper plenum has not been adopted for SCDAP/RELAP5.

E.12.3. Implementation Within the Code

The implementation of the model in the code is in serious question. Ring-to-ring radiation heat exchange is missing, and there are also several questionable assumptions and arbitrary specifications of model parameters. For example, there is no justification provided in the manual as to why half of the radiation incident on a component is reflected backwards diffusely. There is no supporting evidence in the literature to suggest that the diffuse-plus-specular reflection model for radiation heat exchange is superior to the diffuse reflection model. If the diffuse-plus-specular radiation heat-transfer model is to be implemented correctly, exchange factors need to be computed from the knowledge of the radiation property data and system geometry and not specified arbitrarily.

E.12.4. Results of Model Sensitivity Studies

No model or parameter sensitivity studies have been identified.

E.12.5. Results of Benchmarking/Validation Studies

Some validation studies against test data have been identified. A comparison of the TRAC-BD1 radiation heat-transfer model predictions with Göta Radiation Test 27 data (Ref. E.12-3) is not conclusive. The good agreement between test data and model predictions using an anisotropically reflected radiation of 0.5 (μ_i = 0.5) may be fortuitous and coincidence. There is no physical or theoretical justification for using such an arbitrary value of μ_i because bidirectional or directional reflectances of zircaloy or stainless steel have not been measured. More importantly, oxidation of zircaloy or steel would change the reflectance from partially anisotropic (specular) plus isotropic (diffuse) to purely isotropic, and this is not accounted for in the model. Some additional independent comparisons have been reported by the code developers (Presentation 1-9c of Ref. E.12-4). The results were presented for a 7-x-7 rod bundle and showed that the maximum surface-radiation heat flux difference between the current method and TRAC-BD is less than 14%. The difference was attributed to the SCDAP model seeing only two rows of rods in the bundle.

E.12.6. Identified Deficiencies and Options for Model Improvements

The radiation heat-transfer model is used only for intact core geometry, but after the core degrades or when the rods go to the debris, the model is deactivated and radiation heat transfer is turned off. This is a deficiency that becomes more serious at elevated temperatures when the core becomes "open" and the convective heat transfer tends to be relatively less important in comparison to radiation (i.e., the hydraulic diameter becomes larger as rods go into debris, and thus, the convective heat-transfer coefficient becomes smaller).

Rod-to-rod radiation heat exchange is considered, but the ring-to-ring exchange is missing, and axial radiation heat exchange is neglected. These are serious deficiencies of the code. There is no discussion in the documentation of if and how radiation heat exchange between RELAP5 and SCDAP heat structures is coupled in the code. It appears that this has been neglected and represents a major deficiency of the analysis. No planned model improvements have been identified.

E.12.7. Importance of Model to Prediction of Dominant Phenomena

Thermal radiation is the dominant mode of heat transfer at elevated temperatures in the later stages of some postulated severe-accident scenarios, and therefore, accurate modeling is very important.

E.12.8. Technical Adequacy of Model

The model is considered first order. However, treatment of isotropic and anisotropic "outgoing radiation" in the code (Ref. 12-5, pp. 3-140 to 3-142) is different from the technically accepted approaches published in radiation heat-transfer textbooks. One should not arbitrarily multiply the reflected radiation from a surface i by a user-specified factor μ_i (taken to be 0.5) to account for anisotropically reflected contribution. This fraction depends not only on the radiation properties of the surfaces in the "enclosure" exchanging heat by radiation but also on the geometry considered. Some features of the model do not reflect the state of knowledge in this important area. The physics for intact geometry is generally well understood at temperatures relevant to severe-accident conditions; however, the radiation surface properties of materials and water vapor emittance (absorption-emission) data may not be available at high pressures of operating LWRs. Implementation of some model features (i.e., ring-to-ring exchange and coupling of SCDAP to RELAP) is missing, and model validation is practically nonexistent.

The model for coolant emittance is not adequate. The presence of hydrogen in the coolant mixture needs to be considered when calculating the emittance. In addition, the approximation of the spectral absorption bands of steam by a rectangular band model, which ignores the wings of the bands, is not adequate.

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E.13. Core-Region Debris Modeling

E.13.1. Model Description and Pedigree

The core-region debris model is based on several severe fuel-damage experiments and the TMI-2 accident. These experiments have shown that the reactor-core damage proceeds in several stages before the core slumps to the lower head. The model postulates three core configurations during a severe core-disruptive accident and describes the thermal response of the core. The phenomena considered are very complex and not fully understood, and the model is essentially parametric in nature.

Formation and Heatup of Nonporous Debris. This stage of the core-damage progression is caused by the melting of stainless-steel-clad control rods with Ag-In-Cd absorber material and occurs at a temperature of ~1500 K. The meltdown becomes widespread when the reactor-core region exceeds the melting temperature (~2200 K) of the fuel-rod cladding. The submodel describes heatup of the core and subsequent slumping and solidification of the relocated metallic material into a nonporous cohesive debris bed with embedded, intact fuel rods.

The debris layer, supported by intact fuel rods, may extend radially across the core, and its thickness may vary with the radial position (i.e., from one component group to another) and time. Heat conduction in the axial direction is accounted for but is ignored in the radial direction between component groups. Heat generation in the embedded fuel rods is taken into account, as is heat transfer by convection and radiation from the bottom and top surfaces of the nonporous debris bed. The heatup of the debris bed in each component group is calculated using a heat-balance integral method.

Formation and Heatup of Porous Debris. Two mechanisms considered to form porous debris are: (1) thermal shocking by reflood water of a reactor core embrittled by oxidation, and (2) the core fragmentation instigated by the melting of cladding with a very thin oxide layer and a small amount of dissolved oxygen that will not melt the fuel. In the first formation mechanism, the embrittled fuel rods are considered to break up into particles during cooldown when the cladding temperature is decreased below the coolant saturation temperature plus a temperature increment. This coolant saturation temperature is a function of the rate of cooldown.

The second formation mechanism is assumed to occur when the metallic cladding temperature is ~2000 K and, simultaneously, the cladding-oxide layer thickness is less than 0.01 mm. Each time, a map is constructed of the debris regions resulting from degeneration of segments of fuel rods into the porous debris. An arbitrary distribution of porous debris throughout the core is considered. The heatup of the debris region is calculated from an instantaneous, lumped mass energy balance. Volumetric heat generation is considered, but oxidation of particles is neglected. Convective heat transfer is assumed to be the only mechanism for heat removal from the porous debris, and conduction and radiation heat transfer between particles in the debris bed are neglected.

Molten Pool. A uniform mixture of liquefied metallic melt and previously porous solid debris is assumed to be supported and contained by the nonporous debris to form a melt pool. Heat transfer from the interior of the melt pool to the exterior is transferred by natural convection. The pool geometry is mapped into an idealized hemispherical configuration, and the convective heat-transfer coefficients are calculated from empirical correlations published in the literature.

The change in temperature of the molten pool during a timestep is calculated from an instantaneous overall energy balance, which accounts for volumetric heat generation, heat transfer to the nonporous crust that supports the pool, and heat transfer to the crust above the pool. In calculating the temperature distribution in the nonporous debris supporting the molten pool, the boundary condition reflects the melting of the top surface of the nonporous debris due to the contact with the molten pool. The thickening and thinning of the crust above the pool is coupled thermally to the molten pool and the thermal-hydraulic conditions in the core above the molten pool.

E.13.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The phenomena of core-damage progression under severe-accident conditions are poorly understood. As a consequence, the treatment in the code is parametric at best. For example, the embrittled fuel rods are assumed to break up into particles during cooldown where the cladding temperature is decreased below the coolant saturation temperature plus a temperature increment that ranges from 50–1273 K (defined by the user).

The particle size and porosity of the debris resulting from fragmentation is assumed to be the same as that formed by thermal shock during the TMI-2 accident. An average particle diameter is assumed to be 0.87 mm, with an average porosity of 0.54. No spatial variation of either the average particle diameter or porosity across the core is considered. Because test data are lacking (except for a postmortem interpretation of TMI-2 accident), some phenomena described in the model may represent only the code developers' views of the core-degradation phenomena.

E.13.3. Implementation Within the Code

The models calculate the changes in the configuration of the reactor core as damage progresses. The models also calculate the heatup progression and melting in the damaged regions of the core. The method of implementation of the model is judged to be reasonable.

E.13.4. Results of Model Sensitivity Studies

None were found.

E.13.5. Results of Benchmarking/Validation Studies

None were found.

E.13.6. Identified Deficiencies and Options for Model Improvements

The major limitation of the model is that it is parametric, not mechanistic. The deficiencies of the model that have been identified are: (1) heat transfer between molten metal drops and the fuel rod in the cooler core region is neglected; (2) dynamics of the drops to move and solidify into nonporous debris is not considered; (3) conduction and radiation heat transfer in the porous debris bed are neglected in comparison to convection; (4) liquid in the melt pool is perfectly mixed, and no account is taken of possible components having different densities; (5) there is no crust formation model on the upper side of premolten core material; and (6) the documentation is in many places either incomplete or inconsistent. A number of core-region debris modeling deficiencies have been identified by the code developers (Presentation 1-15a of Ref. E.13-1) and need not be repeated here; however, there is no indication that there are any plans to address these deficiencies in the code.

The documentation leaves something to be desired. A few examples of documentation deficiencies are: (1) inconsistency in the description of how the incoherent debris was modified (Ref. E.13-2, p. 3-160); (2) no discussion of what correlations were employed to calculate the convective heat-transfer coefficients h_{conv} and h_{conv2} to define the convective heat fluxes from the top and bottom of the cohesive debris bed (Ref. E.13-2, p. 3-162); (3) no discussion in the documentation of how the thermal diffusivity [for use in Eq. (3-227) on p. 3-160 of Ref. E.13-2] of a cohesive debris bed consisting of fuel rods and solidified molten metallic debris is to be calculated; (4) no justification is provided as to why the heat-transfer coefficient between the debris particle surface and the fluid, h_s, should be 1000 W/m^2 K (Ref. E.13-2, p. 3-172); and (5) no discussion of how the emissivity of the cohesive nonporous debris (which may be an alloy of several relocated metals) is determined (Ref. E.13-2, p. 3-162). It is recommended that these and many other identified documentation deficiencies be improved.

E.13.7. Importance of Model to Prediction of Dominant Phenomena

The core degradation and core-region debris formation can greatly affect the nature and timing of both PWR and BWR core-melt progression; therefore, the models are very important for all applications.

E.13.8. Technical Adequacy of Model

The modeling of core-region debris formation largely reflects the generally poor state of knowledge in this very important area. For severe-accident conditions, the physico-chemical processes of core degradation are poorly understood. The core-region debris model suffers from numerous inadequacies: (1) formation of the cohesive debris neglects the thermal interaction of the molten metals with the coolant; (2) the lack of adequate models for debris fragmentation represents a significant limitation of modeling severe-accident phenomena; (3) the lack of a transient natural circulation model in the melt pool limits the capability to predict sudden crust failure and spillage of a large mass of molten debris into the lower head; (4) some of the models are incomplete and are not capable of describing physical processes of core-melt progression and debris formation; (5) model assumptions are not fully articulated in the documentation, and many important idealizations are not justified; (6) sensitivity, benchmarking, and validation studies do not exist; and (7) parametric models are deficient in time resolution and dynamics.

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E.14. Core-Slumping Models

E.14.1. Model Description and Pedigree

The model describing the slumping of the material from the molten pool in the core to the lower head of the reactor vessel is parametric in nature. The mechanisms that are considered to trigger the slumping are: (1) meltthrough and failure of the nonporous debris that supports the molten pool, and (2) meltthrough and failure of the upper crust and the resulting displacement of liquid from the molten pool by the solid material that falls into the pool. The criteria for slumping from the molten pool are parametric and user specified. For example, if the crust supporting the molten pool melts to the point that its thickness is less than some user-defined value, then slumping is assumed to occur.

Thus, the thickness of the crust is used to determine the time of failure of the crust that surrounds the molten pool. If melting causes the lower crust to thin to a value of less than 25 mm, then the crust is considered to fail and release all of the liquid in the molten pool that is above the point of failure to the lower head. The crust on top of the molten pool that supports the solid debris is considered to fail when its thickness becomes less than 0.5 mm. If some liquid has drained from the molten pool because of the failure of the lower crust, then the upper crust is considered to fail when its thickness becomes less than 25 mm.

E.14.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The crust failure criteria used are based on calculation of stresses under conditions estimated for the TMI-2 accident, but there are large uncertainties associated with the calculations. Large uncertainties in configuration, composition, load, and temperature during evolution of a severe accident require that the failure criteria be treated by the user as a sensitivity parameter. For example, breakup of slumping material such as the upper crust is controlled by the code user. The degree of interaction of the slumping material with the water through which it falls is also defined by the code user.

E.14.3. Implementation Within the Code

A core-slumping model is implemented by a set of user-specified input parameters (i.e., nonporous debris thickness and upper-crust thickness). The nonporous debris layer that supports the molten pool is considered to fail by meltthrough, but the model for calculating the thinning of the layer is not described, and no references to documents are provided where the details can be found. Is the model used similar to the one described in Subsection 3.12.2, "Formation and Heatup of Nonporous Debris," in the SCDAP/RELAP5 code manual (Ref. E.14-1)? Apparently a model is available for predicting the lower- and upper-crust layer thicknesses because the code developers have reported some results of numerical calculations (Presentation 1-15b of Ref. E.14-2). No logic inconsistencies have been identified.

E.14.4. Results of Model Sensitivity Studies

None were found.

E.14.5. Results of Benchmarking/Validation Studies

None were identified.

E.14.6. Identified Deficiencies and Options for Model Improvements

The major limitation of the model is that it is parametric. The input parameters are controlled by the user, and there is no guidance provided in the documentation of how to determine the failure criteria. It is briefly stated (Ref. E.14-1, p. 3-181) that "These failure criteria are based on calculations of stress under conditions estimated by TMI-2 accident...." The question arises as to whether the criteria would be the same for a PWR under a different accident scenario or for a BWR. In brief, it is not clear if the failure criteria are based on some simple zero-order models or on educated guesses that are consistent with the TMI-2 accident.

The model also does not account for a number of important physical processes that are expected to occur: (1) after meltthrough of the lower nonporous debris layer, the enlargement of the opening is neglected; (2) natural convection in the molten pool could greatly impact the timing and location of the crust failure; (3) the interaction of the molten material with water is neglected, and refreezing of the slumping material on the cold, lower part of the core and on the core baffle plate region is neglected; (4) dynamics of molten material relocation are not modeled after either the crust or the cohesive debris fails and the molten materials start relocating into the lower plenum; (5) creep rupture (failure) of the lower crust supporting the molten pool is not considered; (6) there is no modeling of the steam explosion; and (7) the melting of structures and/or enlargement of flow openings by the molten material are not considered. Some of these deficiencies are recognized by the code developers, but no plans for making model improvements have been identified.

E.14.7. Importance of Model to Prediction of Dominant Phenomena

The core-slumping model is one of the more important models. It impacts the timing of the molten material relocation into the lower head and late stages of accident progression.

E.14.8. Technical Adequacy of Model

The model is parametric and the crust (e.g., containing the molten material) failure is based on user-supplied criteria. The timing of events may be very sensitive to the choice of the crust failure criteria, but no sensitivity studies appear to have been performed to assess this assumption. The submodels for the physical phenomena that have been accounted for are invalidated and, in the absence of confirmation, may only represent the code developers' views of core slumping during a severe LWR accident. The model does not describe some physical events, and assumptions made in constructing the model are not articulated in the documentation.

REFERENCES

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E.14-2. R. P. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #1 Handouts," internal memorandum to B. E. Boyack, N-12-91-856 (December 6, 1991).

E.15. Lower-Plenum Debris Heatup

E.15.1. Model Description and Pedigree

The model calculates the heatup of the debris that slumps into the lower plenum and the heatup of the vessel wall. The most important output of this model calculation is the vessel wall temperature distribution from which the time for creep, rupture, or melting of the lower head can be estimated. The lower-plenum debris and head are modeled using a time-dependent, two-dimensional (radial and axial directions), heat-conduction equation.

The model accounts for a time-dependent-growing debris bed and considers spatially varying decay heat, porosity and initial internal energy, thermal conduction, and radiation heat transfer in the porous material. The model also considers debris quench, dryout and melting, and creep-rupture failure or melting of the lower head. The fallen debris is assumed to segregate in some unspecified way into a nonporous metallic debris layer in contact with the vessel wall. Above this denser debris, overlaid with water, there is a porous ceramic debris layer. There are several submodels of the lower-head heatup model that are highlighted.

COUPLE Code. The general two-dimensional, steady-state, transient conduction and advection code (Ref. E.15-1) for planar and axisymmetric geometries has been developed and is used. The finite element method is capable of handling anisotropic heat conduction and advection; however, for SCDAP/RELAP5, advection is neglected, and only the pure heat-conduction and isotropic effective conductivity version has been adopted. The transient temperature of the porous bed is predicted from an energy equation that is based on a thermal equilibrium assumption, i.e., that the solid is at the same temperature as the fluid. The continuum Fourier-Biot heat-conduction law for an inhomogeneous and anisotropic debris bed applies.

Porosity Model. The local porosity of the debris bed is not calculated but is specified by the user as a function of position and time. The local thermal characteristics of the porous bed are calculated by a simple volume-averaging method based on the local volume-fraction-weighted solid and fluid. This type of averaging is appropriate for the density and specific heat but not for the thermal conductivity.

Thermal Conductivity Model. The total effective thermal conductivity of a dry porous bed is represented as a sum of conductive and radiative contributions, calculated using the Imura-Takegoshi and the Vortmeyer models, respectively. The limitation of the models is that they have been developed for homogeneous, uniform-diameter, spherical particle beds and not for inhomogeneous, irregular shaped, nonuniform-diameter particle beds.

Phase Change. The enthalpy method is used to compute phase transformation of a material. This is an accepted procedure and requires the assumption that instead of a discrete temperature (i.e., for a pure substance), the phase change occurs over a small temperature difference. The predicted results were found to be insensitive to the difference chosen in the calculations.

Dryout of Debris. The Lipinski correction is used to calculate dryout as a function of pertinent variables (e.g., debris power density, depth, porosity, particle size, and coolant properties). If debris is quenched, the COUPLE model calculates debris heatup considering liquid water in the voids. If the debris is in a state of dryout, the COUPLE code calculates debris heatup considering that the voids in the debris bed are filled with steam. Heat transfer from the debris to coolant and volumetric vapor generation rate in the control volumes of the lower plenum are handled by RELAP5.

E.15.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics of debris formation and accumulation in the lower head as the accident progresses is poorly understood; therefore, the treatment is parametric. The segregation of the fallen debris into a nonporous metallic layer in contact with the vessel wall and a porous ceramic layer above it occurs in some physically unspecified manner.

E.15.3. Implementation Within the Code

The growth of the bed is dynamic. The user determines the finite element mesh for both the debris and the lower-head structures for use with the COUPLE code. The heatup of the lower head is coupled with the core-slumping and system thermal-hydraulic models. The implementation of the model into the code appears to be appropriate. For example, the amount of heat transferred from the debris to the coolant is added to the energy term for the RELAP5 control volume modeling the lower plenum.

E.15.4. Results of Model Sensitivity Studies

No sensitivity studies with the intent to assess the performance of the model for reasonableness have been performed.

E.15.5. Results of Benchmarking/Validation Studies

Few model validation studies have been conducted. A reference is made (Presentation 1-15c of Ref. E.15-2) to the comparison of temperature distribution in solid-state debris with a textbook solution, but neither the solution nor the published source where the comparison has been made is cited. The calculated temperatures of debris and the vessel lower head in TMI-2 are reported to be within the range of possible temperatures determined by postaccident examination, but a source for this comparison is not given.

E.15.6. Identified Deficiencies and Options for Model Improvements

There are several limitations of the lower-plenum debris heatup model, which include: (1) an inadequate description of the effective thermal conductivity model of the porous bed; (2) the lack of treatment of quenching, dryout, and melting of debris; (3) the lack of a model to predict buoyancy-induced, multidimensional flow in the bed and buoyancy-assisted melting of nonporous metallic and porous ceramic debris; (4) the lack of a model to describe the rise of vapor or migration of liquid in the debris bed; (5) neglect of oxidation of the debris bed and no release of fission products in the debris bed; (6) an inadequate description of gap conductance between the nonporous metallic debris layer and the vessel wall; and (7) the absence of a lower-head-failure model for instrument penetrations. For example, migration of liquefied debris within the solid debris bed is neglected. This movement would change the relative amounts of metals and oxides within each control volume and would therefore change local mixture density, specific heat, and fusion temperature. This composition dependence of the nonporous metallic debris in contact with the vessel would impact the heatup of the structure. These and other model deficiencies identified by the code developers (Presentation 1-15c of Ref. E.15-2) have the potential of impacting not only the heatup of the debris in the lower plenum but, more importantly, the temperature of the structure and its failure.

The documentation is deficient in many respects. The empirical correlation used to calculate the effective thermal conductivity undergoing quenching is not given in the documents (Refs. E.15-1 and E.15-3) nor is a reference made to the published literature. The radiation exchange factor (η) in the radiative conductivity model of Vortmeyer (Ref. E.15-3, p. 3-188) is expected to depend on the debris particle packing and emissivity of the bed material; however, no information is provided in the documentation of how this is being handled. The time-dependent change of the debris bed in each control volume impacts the heatup of the debris through the variation in its composition with time. This change also affects the thermophysical properties of the debris bed in the control volume, but this dependence has not been accounted for in the model or discussed in the documentation. There is no discussion either in the code manual (Ref. E.15-3) or the documentation (Presentation 1-15c of Ref. E.15-2) as to how the boundary conditions needed by the COUPLE model are to be determined. The gap conductance between the solidified debris and structure is a user-defined parameter, but no guidance is provided of how the parameter can be estimated. The convective heat-transfer coefficient between the structure and metallic melt was estimated presumably using the correlation of Jahn and Reinecke (Ref. 3.67 within Ref. E.15-3), which was developed on the basis of experimental data for water and not for a liquid metal or solid debris and a liquid metal.

E.15.7. Importance of Model to Prediction of Dominant Phenomena

The lower-plenum debris and vessel wall heatup play a very important role in the progression of a severe accident. Model-related uncertainties will affect the timing of the lower-head breach and depressurization of the reactor vessel, fission-product release into the containment, direct containment heating, and survival of the containment.

E.15.8. Technical Adequacy of Model

The state of knowledge with respect to key phenomenological issues related to lower-plenum core debris quench, dryout, and vessel wall heatup are not adequate. The lower-plenum debris bed and vessel wall heatup model are semiparametric and may be subject to large uncertainties because user-specified input parameters are required. The model is inadequate because molten material is not relocated but remains where it is formed, and buoyancy-driven convection in the molten material is not considered.

The lower-head debris model is inadequate because the COUPLE code does not treat quenching, dryout, melting (and possibly refreezing) of the debris, and circulation of the molten debris. Because a realistic core-slumping model is also absent and the initial and boundary conditions that impact the lower-head heatup are somewhat arbitrary, lower-head-failure analysis using COUPLE is beyond the capability of the SCDAP/RELAP5 code.

There is also a need to improve documentation. No guidance is provided to the user in the manual (Ref. E.15-3, p. 3-186) as to how the porosity variation of the debris bed is varied as a function of time in each bed control volume. Does the porosity take on the user-specified values of 1 or 0 only? This functional dependence is important because the thermophysical properties (density, specific heat, and effective thermal conductivity) depend on the porosity.

REFERENCES

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E.15-2. R. P. Jenks, Los Alamos National Laboratory, "Copies of SCDAP/RELAP5 Peer Review Committee Meeting #1 Handouts," internal memorandum to B. E. Boyack, N-12-91-856 (December 6, 1991).

E.15-3. C. M. Allison, G. A. Berna, T. C. Cheng, D. L. Hagrman, G. W. Johnson, D. M. Kiser, C. S. Miller, V. H. Ransom, R. A. Riemke, A. S. Shieh, L. J. Siefken, J. A. Trapp, R. J. Wagner, and E. C. Johnson (editor), "SCDAP/RELAP5/MOD3 Code Manual Volume II: SCDAP Code Structure, System Models, and Solution Methods (DRAFT)," EG&G report EGG-2555, Vol. II, Rev. 1 (NUREG/CR-5273) (September 1990).

E.16. Structural Creep-Rupture Model

E.16.1. Model Description and Pedigree

The two-dimensional temperature field in the structures or vessel wall is obtained by using the COUPLE code (Ref. E.16-1). The principle stresses for a given system pressure are calculated by using a thin-shell approach (uniform tangential stress and no shear stress). The equivalent stress is calculated using the Distortion Energy Theory. The equivalent stress reduces the two- or three-dimensional principal stress state to a one-dimensional stress state. The equivalent stress is related to the temperature and creep-rupture time through the Larson-Miller parameter or through the Manson-Hafred theory. For 316 Stainless Steel and Inconel 600, the Larson-Miller parameter is used. However, in the lower range of stresses in A-508 Class 2 carbon steel, the Manson-Hafred theory is used, whereas in the higher range of stresses, the Larson-Miller theory is used.

To allow for variations in temperature with time, a creep-damage parameter is introduced. The damage that occurs during the time a structure remains at a given temperature is obtained by dividing this time by the rupture time obtained from the Larson-Miller parameter. These damage fractions are added, and when the sum reaches a value of unity, the structure is assumed to fail by creep rupture.

E.16.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The model is based on a thin-shell approach and utilizes the creep-rupture database obtained under an isothermal condition. In several severe-accident scenarios, the temperature in the vessel wall will be nonuniform. The effect of the presence of holes in the structures on creep-rupture times is also not considered.

E.16.3. Implementation Within the Code

The temperature input is given by the SCDAP/RELAP5 COUPLE model.

E.16.4. Results of Model Sensitivity Studies

None were found.

E.16.5. Results of Benchmarking/Validation Studies

None were found.

E.16.6. Identified Deficiencies and Options for Model Improvements

Key uncertainties exist with respect to the heat transfer and the mechanical boundary conditions imposed on the structure. Not much database exists with respect to creep rupture under nonisothermal conditions. Also, analysis does not reveal how openings in the structure wall will be treated and how the mechanical boundary conditions are specified. No information is given on the formation and healing of cracks.

E.16.7. Importance of Model to Prediction of Dominant Phenomena

Results of analysis will be helpful in understanding the modes in which the various structure components can fail.

E.16.8. Technical Adequacy of Model

The approach is correct. However, concern exists with respect to the application of the creep-rupture criterion when structure wall temperature is nonuniform and openings exist in the structures. The thin-shell theory may not be valid in all structure types and situations. In cases where the theory is not valid, it is recommended that a detailed finite-element analysis be performed instead using such codes as ABAQUS and PATRAN.

REFERENCES

E.16-1. L. Siefken, EG&G Idaho, Inc., "Model for Creep Rupture of Structural Components," personal communication (September 9, 1992)

E.17. Aerosol Agglomeration Models

The aerosol and fission-product behavior models were implemented to resolve the source-term problem. In particular, the models should be able to follow the radionuclides from the time they leave the fuel until they either leave the primary loop and become the source term to the containment or else permanently deposit upon the structural surfaces within the primary loop.

To resolve the problem, the TRAP-MELT code (Ref. E.17-1) was integrated into the SCDAP/RELAP5 code system. In the process of integration, the developers made various improvements to the code, which involved a temperature-dependent agglomeration kernel and the use of the MATPRO material properties routines.

TRAP-MELT, a parametric nonmechanistic model of fission-product behavior in the reactor coolant system, is part of the STCP. It is difficult to believe that the addition of a Level-2 parametric package into a Level-1 mechanistic code would automatically upgrade the models to a mechanistic level. However, the addition is a first pass and gives the SCDAP/RELAP5 code system additional capabilities. Unfortunately, the addition was not successfully implemented; thus, the SCDAP/RELAP5 code system is unable to adequately calculate the source term.

E.17.1. Aerosol Behavior

E.17.1.1. Model Description and Pedigree. The aerosol mass-distribution calculation is:

$$\frac{\partial q_k}{\partial t}(m,t) = \int_0^{\infty} \phi\ (m-\mu,\mu)\, q_k(\mu,t)\, C(m-\mu,t)\, d\mu$$

$$- q_k(m,t) \int_0^{\infty} \phi(m,\mu,t)\, C(\mu,t)\, d\mu$$

$$- R(m,t)\, q_k(m,t) + S_k(m,t)$$

$$+ \delta_{lk}\xi(m,t)\, C(t) - \frac{\partial}{\partial m}\left[\xi(m,t)\, q_k(m,t)\right] \quad ,$$

where

$q_k(m,t)$	=	Mass distribution of k'th component,
$\phi(m,\mu,t)$	=	Agglomeration kernel for particles of mass m and μ,
$C(m,t)$	=	Particle number concentration of distribution,
$R(m,t)$	=	Removal rate (deposition) for particles of mass m,
$\xi(m,t)$	=	Mass condensation rate onto particles of mass m,
$S_k(m,t)$	=	Mass source rate of k'th component,
δ_{lk}	=	1 k = 1 (water component), and
		0 k ≠ 1.

Implicit in this equation are four assumptions: (1) the aerosol is well mixed, (2) particles are characterized by their mass; that is, particles of a given mass size all have the same shape, (3) boundary layers do not effect deposition, and (4) particles do not break up into smaller particles.

E.17.1.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The model is used in several industries and should be able to predict dominant phenomena.

E.17.1.3. Implementation Within the Code. The equation can be transformed into a series of ordinary differential equations by using a mass discretization scheme. This is done by the MAEROS model in CONTAIN and MELCOR, by the CHARM model in VICTORIA, and by the MATSA model in TRAP-MELT with varying degrees of detail. The SCDAP developers have chosen an alternative method that solves the agglomeration integrals explicitly at each timestep. The first-order modeling comes from the time dependency of the agglomeration kernel. The SCDAP developers supply two levels of sophistication in the DERN model to reduce computational time. The technique is described in Appendix A of the SCDAP manual and consists of a determination of the particle size resulting from the agglomeration of particles.

E.17.1.4. Results of Model Sensitivity Studies. Comparisons of the SCDAP technique with other techniques are not known.

E.17.1.5. Results of Benchmarking/Validation Studies. Comparisons of the SCDAP technique with other techniques are not known. The Marviken Aerosol Transport Tests have been analyzed with other codes and should be considered as a validation study. The code should be examined for the agglomeration/deposition timestep such that within a volume there is sufficient time for aerosols to both agglomerate and deposit before being transported to another cell.

E.17.1.6. Identified Deficiencies and Options for Model Improvements. The physics and numerics necessary to solve this equation is in the open literature (Refs. E.17-2 and E.17-3).

E.17.1.7. Importance of Model to Prediction of Dominant Phenomena. An accurate calculation of the aerosol mass distribution is needed to predict the radio nuclide behavior (source term); thus, the model is necessary.

E.17.1.8. Technical Adequacy of Model. The physics and numerics necessary to solve this equation is in the open literature (Refs. E.17-2 and E.17-3). The SCDAP developers have chosen to develop their own first-order model. If the model is computationally faster and reasonably accurate, then it is a step forward in aerosol analysis. The only use of the model indicates that it is not fast enough for SCDAP users. It is not known if the problem is the model, implementation, accident sequence, or terms in the equation. Because the model has not been used, it has a validation class of inadequate implementation.

E.17.2. Agglomeration Kernel

E.17.2.1. Model Description and Pedigree. There are three types of agglomeration processes described in the SCDAP manual: (1) Brownian motion, (2) gravity, and (3) turbulence. The three processes are combined as:

$$\phi_{Total} + \phi_B + \left[\phi_G^2 + \phi_T^2\right]^{1/2} \quad .$$

The above expression was derived by Saffman and Turner (Ref. E.17-4) using mathematical arguments; however, for flow regimes where both the gravity and turbulence rates are of the same order of magnitude, there is insufficient data to justify the combination. SCDAP addresses the problem by allowing the user to choose between the above combination and one where the turbulence terms are ignored.

Brownian agglomeration is computed by employing the standard expression derived by Schmoluchowski (Ref. E.17-5). Both MAEROS and CHARM use this same formulation with a correction due to Fuchs (Ref. E.17-5) for particle sizes less than the mean free path. SCDAP does not include the Fuchs correction, any sticking efficiency, or shape factors.

The **gravitational agglomeration** is computed using the expression derived by Saffman and Turner (Ref. E.17-4). The SCDAP implementation includes a velocity correction for buoyancy but does not include the shape factors and sticking efficiencies.

The **turbulent agglomeration,** resulting both from shear and inertia, is computed using the derivation of Saffman and Turner (Ref. E.17-4). The SCDAP implementation includes the buoyancy correction and neglects the shape factor and sticking efficiencies.

E.17.2.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The models chosen describe the known agglomeration processes anticipated in reactor scenarios. The flow friction and Nusselt correlations should be available from RELAP rather than assuming forced-convection turbulent flow within SCDAP.

E.17.2.3. Implementation Within the Code. It is not known how the rate terms were actually implemented; however, the expressions given in the manual have several errors.

E.17.2.4. Results of Model Sensitivity Studies. There are no published calculations using these models. The original TRAP-MELT2 models have been compared with experiments by Williams (Refs. E.17-6 and E.17-7). He noted that the inclusion of the Fuchs correction to the Brownian agglomeration term changed the total deposition for the Marviken analysis significantly. It changed calculated wall and floor depositions from being in error by a factor of two to a calculated error of 10%. He also notes that the constant multipliers in the agglomeration kernels used by TRAP-MELT and SCDAP can be off by factors of two to four using different correlations. In the MELCOR Peer Review (Ref. E.17-8), Gieseke noted that the Fuchs collision efficiency term gives better results if divided by three. This was also noted by Williams.

E.17.2.5. Results of Benchmarking/Validation Studies. The code has not been benchmarked against either its predecessor TRAP-MELT or experimental data.

E.17.2.6. Identified Deficiencies and Options for Model Improvements. The physics and numerics necessary to solve this equation is in the open literature (Refs. E.17-2 and E.17-3).

E.17.2.7. Importance of Model to Prediction of Dominant Phenomena. The model is used to predict fission-product retention in the reactor coolant system. It may be considered as a major source-term reduction model.

E.17.2.8. Technical Adequacy of Model. This is a case where reasonable models were integrated, but the open literature, where modifications to improve capabilities were discussed, was ignored.

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E.17-3. F. Gelbard, "MAEROS User Manual," Sandia National Laboratories report SAND80-0822 (NUREG/CR-1391) (1982).

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E.17-7. D. A. Williams, "Further Analysis of the Marviken Aerosol Transport Tests 1, 2a and 2b Using the TRAPMELT2-UK Computer Code," Atomic Energy Establishment Winfrith report AEEW-M2298 (April 1986).

E.17-8. B. E. Boyack, V. K. Dhir, J. A. Gieseke, T. J. Haste, M. A. Kenton, M. Khatib-Rahbar, M. T. Leonard, and R. Viskanta, "MELCOR Peer Review," Los Alamos National Laboratory report LA-12240 (March 1992).

E.18. Aerosol Particle Deposition

E.18.1. Model Description and Pedigree

There are five types of deposition processes described in the SCDAP/RELAP5 manual: (1) gravitational settling, (2) thermophoresis, (3) turbulent flow deposition, (4) laminar flow deposition, and (5) deposition in bends. These processes are combined additively as is typical of aerosol deposition codes.

Gravitational settling is calculated as:

$$U_G = gmB \quad ,$$

where B is the particle mobility corrected for buoyancy and the non-Stokesian settling term. This is a settling term inappropriate for settling in turbulent flow through horizontal pipes.

Thermophoretic deposition is calculated as derived by Brock (Ref. E.18-1). The SCDAP implementation uses the original values for the coefficients, and Loyalka (Ref. E.18-2) uses slightly different coefficients to extend the range for all particle sizes. These latter coefficients will change the velocities by at most 40% (Ref. E.18-3). The SCDAP implementation for the boundary-layer thickness used in the temperature gradient assumes that the flow is turbulent.

Turbulent flow deposition by particle diffusion is calculated as derived by Davis (Ref. E.18-4). There are other formulations that could have been chosen [such as Friedlander (Ref. E.18-5)], which use heat-mass-transfer correlations.

The deposition due to impaction from the turbulent flow of supermicron particles is also modeled in SCDAP as derived by Friedlander and Johnstone (Ref. E.18-5). The Liu and Agarwal (Ref. E.18-6) correlation is more widely used for this deposition mechanism.

The deposition from laminar flow by diffusion, Re< 2300, is implemented as described by Gormley and Kennedy (Ref. E.18-7). The SCDAP implementation uses this value to replace the deposition due to impaction for laminar flows. (It would be more logical to replace the turbulent flow deposition with this model and set the impaction to zero when the flow is laminar.) In addition, the original correlations have been reduced by a factor of four and a low velocity limit is introduced, neither of which are referenced.

The deposition in a bend is a model derived by Newman. This model is unpublished and has not been correlated with available data. Other codes use a model developed by Pui (Ref. E.18-8) for this phenomena.

E.18.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The set of flow friction and Nusselt correlations should be available from RELAP rather than assuming a set in SCDAP. Parozzi (Ref. E.18-9) also was concerned with this. His calculations indicate that significantly more deposition would be found in natural circulation situations than SCDAP would calculate.

E.18.3. Implementation Within the Code

It is not known how the rate terms were implemented in the code, but the expressions given in the manual have several errors.

E.18.4. Results of Model Sensitivity Studies

There are no published calculations using these models. The original TRAP-MELT deposition models have been compared with experimental data by Kuhlman (Ref. E.18-10), Parozzi (Ref. E.18-9), Williams (Refs. E.18-3 and E.18-11), and Wright (Ref. E.18-12). The general conclusions are that deposition on vertical walls is underestimated and deposition on horizontal pipes is overestimated.

Williams has also run VICTORIA in these same tests and was able to duplicate the experimental results when an intervolume settling was allowed. VICTORIA uses similar models but also allows the agglomeration kernel to be time dependent. The supermicron deposition was replaced in VICTORIA with the development due to Sehmel (Ref. E.18-13) because the use of the Perry friction correlation with the particle-relaxation-dependent correlations in TRAP-MELT gave negative depositions at low turbulent Reynolds numbers (Ref. E.18-3).

E.18.5. Results of Benchmarking/Validation Studies

The code has not been benchmarked either against its predecessor TRAP-MELT or against experimental data.

E.18.6. Identified Deficiencies and Options for Model Improvements

Most of the models represent some of the standard deposition equations used in the aerosol industry. The significant differences are in the bend model, the impaction model where there is no evidence of sensitivity calculations, gravitational settling in horizontal pipes, and diffusive deposition due to vapor condensation on the walls. Intravolume settling and the possibility of resuspension should also be accounted for. It should be noted that there are more sophisticated deposition models available in the open literature. It is not obvious at this time that they are needed for reactor-safety, source-term analysis. The errors in the manual are significant, and the code should be checked for accuracy.

E.18.7. Importance of Model to Prediction of Dominant Phenomena

The model is used to predict fission-product retention in the reactor coolant system. It may be considered as a major source-term reduction model.

E.18.8. Technical Adequacy of Model

This is an area where the models were integrated, but improvements available in the open literature were ignored.

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E.18-5. S. K. Friedlander and H. F. Johnstone, "Deposition of Suspended Particles from Turbulent Gas Steams," *Industrial and Engineering Chemistry* **49**, 1151–1156 (1957).

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E.19. Vapor Evaporation/Condensation

E.19.1. General Vapor Evaporation and Condensation Models

E.19.1.1. Model Description and Pedigree. The vapor condensation and evaporation models used in SCDAP were taken from the TRAP-MELT2 code.

Inherent in the model equations and their implementation are the following assumptions:

1. Particles are assumed to be at the gas temperature.

2. Particle size (curvature) has no effect on vapor pressure at particle surface.

3. The dilution of species in the deposited layer does not affect partial pressure above the surface (surface temperature is the only factor).

4. The species in gas above the surface is in equilibrium with the same compound in the surface layer.

5. The particle motion through gas does not affect mass transfer ($k_p = D/r$).

6. Particles are assumed spherical and all of same composition.

The mass-transfer coefficients are determined as:

$$K = \frac{D_v(t)}{\delta_D} \quad ,$$

where D_v is the diffusivity of the species in the gas at the appropriate temperature and d_D is the diffusion boundary-layer thickness. For particles from bin L, d_D is equal to the particle radius; for flow along a wall, the expression used is from Keller (Ref. E.19-1):

$$\delta_{DW} = \frac{D_e}{NuSc^{.333}} \quad .$$

The implementation in SCDAP sets Nu equal to the forced-convection turbulence flow situation for all cases.

E.19.1.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The equations for the rate of change have been reviewed by others and found to be reasonable. The choice of mass-transfer coefficients is the key term, and the solution of the stiff equations is the numerical problem. The developers of the RAFT code (Ref. E.19-2) believe that this growth mechanism is the dominant phenomena in high-pressure-reactor scenarios.

E.19.1.3. Implementation Within the Code. The model appears to be implemented correctly. However, the use of the turbulent flow model for all wall scenarios is unreasonable; something better should be available from RELAP.

E.19.1.4. Results of Model Sensitivity Studies. There are no published results from this part of the code.

E.19.1.5. Results of Benchmarking/Validation Studies. The code has not been benchmarked either against its predecessor TRAP-MELT or against experimental data.

E.19.1.6. Identified Deficiencies and Options for Model Improvements. The primary deficiency is the small number of species involved. Detailed speciation models use over a hundred species. Perhaps one of those models should be included.

E.19.1.7. Importance of Model to Prediction of Dominant Phenomena. If the fission product condenses, it will be able to deposit in the RCS. If it remains vapor, it will eventually release to the containment and become part of the source term.

E.19.1.8. Technical Adequacy of Model. The zeroeth model appears reasonable, but without data comparisons to justify the limited number of mass-transfer coefficients, it is not technically acceptable.

E.19.2. Derivation of Equations for Equilibrium Vapor Mass Concentrations from Vapor Pressure Information

E.19.2.1. Model Description and Pedigree. The developers of SCDAP have taken a Van der Waals equation-of-state formula to derive a relationship between partial pressure information and mass concentrations.

This is the chemistry calculation for SCDAP. It covers the following species: Iz, CsI, CsOH, Te, Ag, Cd, and Sn. This indicates that all cesium is either CsOH or CsI and that tellurium does not react with anything. These assumptions are consistent with TRAP-MELT. The question is whether these few species are the only ones of interest.

E.19.2.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. In the Swiss analysis of the INEL LOFT-FP2 experiment (Ref. E.19-3), the Swiss were unable to use this model in either its TRAP-MELT form or in its PULSE form until the chemistry speciation was modified to include AgI.

E.19.2.3. Implementation Within the Code. The model appears to be implemented correctly.

E.19.2.4. Results of Model Sensitivity Studies. There are no published results from this part of the code.

E.19.2.5. Results of Benchmarking/Validation Studies. The code has not been benchmarked either against its predecessor TRAP-MELT or against experimental data.

E.19.2.6. Identified Deficiencies and Options for Model Improvements. The primary deficiency is the small number of species involved. Detailed speciation models use over a hundred species. Perhaps one of those models should be included.

E.19.2.7. Importance of Model to Prediction of Dominant Phenomena. If the fission product condenses, it will be able to deposit in the RCS. If it remains vapor, it will eventually release to the containment and become part of the source term.

E.19.2.8. Technical Adequacy of Model. This is not a mechanistic treatment of well-known physics. Without benchmarking studies to justify the zeroeth-order model, it is not acceptable.

REFERENCES

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E.19-2. K. H. Im, R. K. Ahluwalia, and H. C. Lin, "The RAFT Computer Code for Calculating Aerosol Formation and Transport in Severe LWR Accidents," Electric Power Research Institute report NP-5287-CCM (July 1987).

E.19-3. S. Guntay, E. W. Coryell, and M. L. Carboneau, "A Post Test Analysis of the OECD LOFT Experiment LP-FP-2 Using the Computer Programs SCDAP/RELAP5, TRAP-MELT2.2 and PULSE," *PSI-Bericht* **95** (April 1991).

E.20. Heterogeneous Chemical Reaction between Chemical Species and Wall

E.20.1. Model Description and Pedigree

The SCDAP developers have chosen to model this phenomena with a simple deposition velocity. The literature indicates a more complex phenomena that accounts for oxide layers, chemical speciation, and concentration gradients, as well as the time and temperature.

The correlations used are not referenced, and there is no indication as to whether the species can be resuspended. Parozzi (Ref. E.20-1) indicates that the 1-cm/s deposition velocity of Te_2 is an order of magnitude too high and should be shut off as temperatures exceed 850 K.

E.20.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

This can be an important mechanism for the removal of fission products from the system. It has also been found that this phenomena can cause the CsI to disassociate with the cesium reacting with the silicon in the steel and the iodine reacting with the bulk gas as HI.

E.20.3. Implementation Within the Code

The model appears to be implemented correctly.

E.20.4. Results of Model Sensitivity Studies

There are no published results from this part of the code.

E.20.5. Results of Benchmarking/Validation Studies

The code has not been benchmarked either against its predecessor TRAP-MELT or against experimental data.

E.20.6. Identified Deficiencies and Options for Model Improvements

The use of a constant sorption velocity independent of temperature is unreasonable. It is most likely that one effect will be in control when the wall temperatures allow condensation and another when the interaction takes place above the boiling point. In addition, the literature indicates that sorption is reversible.

E.20.7. Importance of Model to Prediction of Dominant Phenomena

If fission products are absorbed by the structural surfaces, then the source term can potentially be reduced to very small values.

E.20.8. Technical Adequacy of Model

This is an area of questionable physics being treated parametrically but not adequately.

REFERENCES

E.20-1. F. Parozzi and G. Sandrelli, "Italian Contribution to TRAP-MELT Code Development," in "Proceedings of an International Symposium on Severe Accidents in Nuclear Power Plants," International Atomic Energy Agency report IAEA-SM-296/58P (1988).

E.21. Materials Properties

E.21.1. MATPRO, Material Properties Library—Solids and Liquids

E.21.1.1. Model Description and Pedigree. Table E-I is a compilation of the uncertainties in the correlations used. As shown, 10 material groups are examined for 11 different properties. MATPRO provides some data for all materials—more data for the more significant materials (UO_2) and less data for the less significant materials (Inconel 718). Table E-I gives the temperature range in which the correlation should be used for an average error. In most cases, the errors have been provided in the documentation and appear reasonable. Note that several error estimates are large (viscosities for UO_2 and B_4C, for example); these are due to a lack of experimental data.

Table E-1 includes an entry as core components. These are mixtures of the previous materials that may be necessary during melt progression. The code uses atomic fractions to determine mixture properties, but based on Section 3, it appears that the more common mass averages for heat capacity and enthalpy are used with the inverse for thermal conductivity. The mole fraction system used for the solidus-liquidus temperature curves are detailed in MATPRO.

Table E-II lists additional correlation uncertainties.

Table E-I
Uncertainty Estimates for Selected Functions in MATPRO, the Materials Properties Library for SCDAP/RELAP5/MOD2

	Melt Temp. (T_m)	Specific Heat (C_p)	Enthalpy (h)	Thermal Cond. (k)	Emis-sivity (ε)	Thermal Strain ($\partial L / L_0$)	Density (ρ)	Young's Modulus (E)	Poisson's Ratio	Viscos-ity	Vapor Pres-sure
Uranium Dioxide UO_2+(U,Pu)O_2	X	300–3000 ±3%	300–3000 ±3%	300–3000 ±7%	300–2400 ±6.8%	300–3000 ±10%	300–3000 ±1%	450–1600 ±3.5%	300–1000 ±34%	300–3113 ±67%	999–4000 ±2%
Uranium Metal		300–3500 NE	300–3500 NE	300–1406 ±20%		300–1132 ±10%	300–1132 NE				
Zircaloy	X	300–1090 ±3%	300–1300 NE	300–2098 ±10%		300–1244 ±12%	300–1090 ±10%	300–822 ±10%			
Zircaloy Oxides ZrO_2	X	300–3300 ±20%	300–3300 ±20%	300–2900 ±10%	300–1575 ±3%	300–3300 ±5%	300–3300 NE	300–2810 ±20%	300–2810 ±20%		
Cont. Rod Clad 304 Stainless Steel	X	300–1558 ±10%	300–1800 ±10%	300–1800 ±2%		300–1800 ±10%	300–1800 ±10%				
Ag-In-Cd	X	300–1500 ±10%	300–1500 ±10%	300–1500 ±1%		300–1300 ±10%	300–1300 ±2%			999–3000 ±80%	

Table E-I (cont.)
Uncertainty Estimates for Selected Functions in MATPRO, the Materials Properties Library for SCDAP/RELAP5/MOD2

Boron Carbide B4C	X	300–2800 ±10%	300–2800 ±10%	300–2800 ±50%		300–2800 ±20%	300–2800 ±30%			1050–3000 ±80%	
Stainless Oxides (Average)		300–3300 NE	300–3300 NE	300–1300 ±20%		300–1500 NE	300–1500 ±0.5%				
Spacer Grids (Inconel 718)	X										
Core Material Zr-U-O Mix	X	300–3300 ±10%	300–3300 ±10%	300–3300 ±25%		300–3300 ±5%	300–3300 ±5%			2900–3500 ±80%	

Note: 1. First entry in each square indicates valid temperature range in degrees Kelvin.
2. Second entry in each square indicates uncertainty estimate in percent.
3. All core component values are functions of component concentration as well as temperature.

Table E-II

MATPRO Function Uncertainties Not Included in Tables

Uranium Dioxide creep (N.E.[a]), densification (N.E.), swelling (N.E.), pressure sintering (±0.5%), restructuring (N.E.), fracture strength (N.E.)
Uranium Metal oxidation parabolic rate constant (N.E.)
Zircaloy hydride prevention temperature (N.E.), shear modulus (±30%), axial growth (313-633K, ±10%), creep (N.E.), plastic deformation (N.E.), annealing (N.E.), mechanical limits and embrittlement (N.E.), cyclic fatigue (N.E.), collapse pressure (N.E.), Meyer hardness (N.E.)
Zirconium Dioxide mechanical limits and embrittlement (±70%)
Ag-In-Cd, B_4C surface tension (±667%)
Core Components solution and precipitation (N.E.), friction coefficient (±90%), interfacial surface tension (±220%), heat of solution (±100%), heat of fusion (N.E.)
Ag-Zr solubility of zircaloy in Ag-In-Cd (N.E.)
Gases - I/I_2, Cs-I, CsOH, Te, Cd, Ag, H_2Te, HI, Sn, SnTe, H_2O, ZrO_2, UO_2, $C/C_2/...C_6$, AgI equilibrium vapor concentrations (N.E.)
Chemical Reactions fuel oxidation (N.E.), zircaloy oxidation (N.E.), cladding H_2 uptake (N.E.), stainless-steel oxidation in steam (±25%), rate of dissolution of UO_2 in Zr-U-O (N.E.)
Utilities linear interpolation (N.A.[b]), texture factors (N.A.), collected heats of fusion (N.A.), mass fraction-mole fraction conversion (N.A.), integral of the reciprocal of thermal conductivity (N.A.), atomic fraction (N.A.)
Creep-Rupture Failure rupture time (N.E.), creep damage term (N.E.)

[a]N.E. = No or Nonconclusive Uncertainty Estimate.
[b]N.A. = Not Applicable.

E.21.1.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. This is a reasonable method of including material property variations into the code. It is not known if the detailed variations are necessary.

E.21.1.3. Implementation Within the Code. The correlations are implemented correctly.

E.21.1.4. Results of Model Sensitivity Studies. The MATPRO correlations and documentation have been available for several years. The correlations are widely used, and there are no known published reviews.

E.21.1.5. Results of Benchmarking/Validation Studies. The MATPRO correlations and documentation have been available for several years. The correlations are widely used, and there are no known published reviews.

E.21.1.6. Identified Deficiencies and Options for Model Improvements. Because the code was designed for use as an experimental analysis tool, it is imperative that mass and energy be calculated within acceptable bounds. The primary properties that allow this are specific heat, density, and thermal conductivity. MATPRO covers these values reasonably well; however, there are some missing values. In particular:

<u>At higher temperatures (but less than 2500–3000 K)</u>
Zirconium—Specific heat and density @ T>1090 K
Stainless steel—Specific heat, conductivity, and density @ T>1800 K
Ag-In-Cd—Specific heat, conductivity, and density @ T>1500 K

<u>At all temperatures (but less than 2500–3000 K)</u>
Inconel—Specific heat, conductivity, and density @ T>300 K

Because the code should analyze the relocation of core materials, it is also important to handle not only mixtures but also eutectics. In particular, phase diagrams and heat-of-fusion data are needed to address liquefaction and solidification issues. MATPRO covers the fuel-zircaloy and silver-zircaloy eutectics; however, there are some missing combinations:

Zirconium/Inconel—Cladding-grid space liquefaction @ T = 1500–1600 K

Zirconium-stainless steel—Cladding-control-rod sheath liquefaction @T = 1500–1600 K

Stainless steel-B_4C—Control-rod, material-sheath liquefaction @ T = 1450–1500 K

The rate of relocation of the molten materials is dictated by the model chosen and the previous properties, as well as by the viscosity. The viscosity of the core materials has a known error of 60–80%. A reduction of this error, especially for the Zr-U-O eutectic, would increase the confidence in a particular relocation model substantially.

E.21.1.7. Importance of Model to Prediction of Dominant Phenomena. Values for the thermophysical properties are essential to the correct prediction of phenomena. However, it is not clear if highly detailed variations in pressure, temperature, or mass fractions are as essential to this prediction.

E.21.1.8. Technical Adequacy of Model. The property packages provide the effect on thermophysical properties due to changes in pressure, temperature, and composition. In the SCDAP/RELAP5 code, the fuel rod, control rod, structure surface, and many noncondensable gas properties are contained in the MATPRO documentation. The correlations are reasonable.

E.21.2. MATPRO—Gases

E.21.2.1. Model Description and Pedigree. The developers have supplied some properties for the 10 noncondensable gases in the code system. The properties and known error margin for these gases are given in Table E-III. The correlations chosen by the developers come from basic and dated texts such as Zemansky (Ref. E.21-1), Bird (Ref. E.21-2), and the *Handbook of Chemistry and Physics* (Ref. E.21-3). The form of the equations and the errors determined are reasonable. The developers have also supplied the partial pressure correlations for the seven species used in the chemistry calculation (I_2, CsI, CsOH, Te, Ag, Cd, and Sn). These are the same species chosen by the TRAP-MELT developers and indicate the limited chemistry capabilities of the code.

Table E-III

Uncertainties of Noncondensable Gas Functions in MATPRO, the Materials Properties Library for SCDAP/RELAP5/MOD2

	Specific Heat C_p	Thermal Conductivity k (W/m-K)	Effective Emissivity ε	Viscosity μ	Mean Free Path Length (m)
Helium	Constant	±(8.00e-7) (T^1.5)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Argon	Constant	±(4.96e-10) (T^2.25)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Krypton	Constant	±(1.45e-9) (T^2)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Xenon	Constant	±(2.77e-8) (T^1.5)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Hydrogen	2nd-order polynomial	±(2.10e-6) (T^1.5)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Nitrogen	2nd-order polynomial	±(2.64e-6) (T)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Oxygen	2nd-order polynomial	±(2.34e-9) (T^2)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Carbon Monoxide	2nd-order polynomial	±{[4/3(T-400)e-4] +0.002}%	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Carbon Dioxide	2nd-order polynomial	±(8.78e-12) (T^3)	No Estimate	No Estimate	$F(\mu, \rho, T^{\frac{1}{2}})$
Water Vapor	2nd-order polynomial	±6%	No Estimate	Linear data fit	$F(\mu, \rho, T^{\frac{1}{2}})$

E.21.2.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. This is a reasonable method of including the noncondensable gas effects into the code. The limited speciation implies that all cesium is either CsOH or CsI and that silver, tellurium, tin, and cadmium do not chemically react. These are poor assumptions.

E.21.2.3. Implementation Within the Code. The correlations are implemented correctly.

E.21.2.4. Results of Model Sensitivity Studies. The MATPRO correlations and documentation have been available for several years. The correlations are widely used, and there are no known published reviews.

E.21.2.5. Results of Benchmarking/Validation Studies. The correlations are widely used, **but** there are no known published reviews.

E.21.2.6. Identified Deficiencies and Options for Model Improvements. Higher-order correlations for heat capacity of the noncondensables are available, but data for viscosities and emissivities are less well known. Data for additional condensible species are available.

E.21.2.7. Importance of Model to Prediction of Dominant Phenomena. The noncondensable gases play only a small part in the prediction of dominant phenomena. The condensables are critical to the prediction of source-term phenomena.

E.21.2.8. Technical Adequacy of Model. The model is reasonable for noncondensables, but the lack of an adequate number of condensible species makes the model technically inadequate to use for source-term analysis.

REFERENCES

E.21-1. M. W. Zemansky, ed., *Heat and Thermodynamics* (McGraw-Hill Book Company, Inc., New York, 1957).

E.21-2. R. B. Bird, W. E. Stewart, and E. N. Lightfoot, *Transport Phenomena* (John Wiley and Sons, New York, 1960).

E.21-3. C. D. Hodgman, ed., *Handbook of Chemistry and Physics, Thirty-Eighth Edition* (Chemical Rubber Publishing Co., Boca Raton, Florida, 1956).

E.22. Fission-Product Decay Heat

E.22.1. Model Description and Pedigree

The reduction in fission-product decay heat resulting from the loss of volatile elements after a major disruptive event is accounted for by the model. The fractional release of volatile fission products from fuel is calculated by the PARAGRASS code, and the decay-power fraction is obtained from the tables calculated by the ORIGEN2 code. Fission-product decay heat for the intact fuel is calculated from the power history using the methods and data described in the ANS-5.1.-1979 Standard.

The decay-heat reduction fraction is determined by taking the ratio of decay power with volatile release to the decay power without volatile release using the method suggested by Schnitzler (Ref. E.22-1). The model distributes the decay heat into cesium, iodine, tellurium, and beta-gamma contributions. The decay-heat contributions from the remaining volatile fission elements are lumped into a single composite group. The method provides a means for treating decay-heat reduction in the disrupted fuel region but provides no information on the distribution of decay power among the released materials.

E.22.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics involved is well understood, and the treatment in the code is judged to be first order, but some shortcuts have been made to increase computational speed.

E.22.3. Implementation Within the Code

The FDECAY model is called by the nuclear-heat-generation model NHEAT routine to return the decay-heat fraction correction factor by which the decay-heat rate is multiplied. Fission-product, decay-power fractions are modeled in SCDAP for four representative fission-product inventories. Calculations were then performed in which the inventories were allowed to decay with and without the volatile element removal modeled.

The four fission-product inventories considered are representative of a wide range of possible fuel inventories, but they represent expert-opinion estimates of volatile release in quench scenarios only. The three volatile element release cases considered cover a wide range of conditions likely to be encountered by a SCDAP/RELAP5 user examining INEL experiments or TMI-2. However, if the considered scenarios are felt to be inadequate for the problem being analyzed, like station blackouts, the user may provide problem-specific data.

E.22.4. Results of Model Sensitivity Studies

None were found.

E.22.5. Results of Benchmarking/Validation Studies

The model has been validated indirectly by comparing the NHEAT predictions against those for the ANSI/ANS-5.1-1979 Standard problem and the TMI-2 decay problem; good agreement has been obtained.

E.22.6. Identified Deficiencies and Options for Model Improvements

The accuracy of the model predictions may largely depend on the applicability of the input data, and unless the reactor parameters such as the power density, burnup, core composition, and prior power history are close to one of the four representative inventories analyzed, it would be prudent to repeat the fission-product, decay-power fraction calculations for the particular reactor of interest. The available information cannot be applied with confidence to accident scenarios where the fuel failure mode, and as a consequence, the resulting volatile element releases, differ substantially from the fuel heatup and quench failure model assumed in the analysis. Some deficiencies and inconsistencies have been identified in the model. For example, SCDAP assumes that ^{235}U is the only nuclide present and the model is not appropriate for high burnup, whereas RELAP5 includes all three nuclides (^{235}U, ^{238}U, and ^{239}P) but does not apply the G factor. This is considered to be a major deficiency of the code.

E.22.7. Importance of Model to Prediction of Dominant Phenomena

A calculation of fission-product decay heat is required to calculate the fuel temperatures under all accident conditions; however, the reduction in fission-product decay heat resulting from the loss of volatile elements is significant only for a major disruptive event.

E.22.8. Technical Adequacy of Model

The overall approach is judged to be reasonable, and the model is considered to be zeroeth order. The model relies for the most part on user-specified inputs for determination of the fission-product decay fraction during a severe-accident condition. However, the inadequacies identified are not likely to be a limiting factor in most severe-accident calculations. If scenarios involving high-burnup fuel are to be analyzed, the model will have to be revised to deal with this inadequacy.

REFERENCES

E.22-1. B. G. Schnitzler, "Fission Product Decay Heat Modeling for Disrupted Fuel Regions (FDECAY)," EG&G report EGG-PHYS-5698 (December 1981).

E.23. Decay-Heat, Energy-Deposition Model

E.23.1. Model Description and Pedigree

The energy-deposition model in SCDAP/RELAP5 code assigns the released energy either to the vapor space or the solid that is first impacted by the decay particle or the solid upon which the fission product is deposited. The decay energy from the fission products carried in the vapor space or deposited on a structural surface is distributed according to the local group population in the cell. Both gamma and beta energy releases from airborne and deposited fission products are considered.

Gamma energy deposition from airborne release is apportioned according to the view factor of the surrounding heat structure, and the distribution within the solid structure is apportioned according to the attenuation model. Half of the energy released from the fission products on the surface of the heat structure is assumed to enter the structure, and half is assumed to be released toward the adjacent vapor space. By assumption, this half is not attenuated by the vapor space and is distributed among the heat structures bounding the control volume.

Beta energy deposition due to airborne release is according to the view factors, and the attenuation within vapor space is according to the Katz and Penfold model, which considers the presence of steam only. The energy absorbed by the steam in the vapor space is taken as a volumetric source for the control volume, and the remaining energy is apportioned among the various heat structures according to the view factor as an incident energy flux at the surface. Half of the beta energy release from deposited fission products is assumed to be incident on heat structure and the remaining half on the vapor space. Beta energy attenuation in vapor is calculated based on the equivalent distance to the bounding heat structure and on a characteristic endpoint energy for each group.

E.23.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

Gamma and beta energy attenuation in gases and solids are well established for well-defined geometries. Data on attenuation coefficients in steam and steel are also available. The overall treatment is reasonable for well-defined intact geometry, but the treatment for degraded core geometry is questionable. The document (Ref. E.23-1) is very terse, and little in the way of model description, quantitative information, or relevant literature sources is provided. Only a model summary is given, and no justification of the assumptions are included in the documentation. Possibly the code developers did not consider the gamma and beta energy deposition as contributing significantly to the overall heat deposition in a given cell and surrounding heat structure.

E.23.3. Implementation Within the Code

No details are provided (Ref. E.23-1) of how the model is implemented in the code. For example, no discussion is included in the document of how the "predetermined attenuation factor corresponding to the particular structure" is to be evaluated for calculating the gamma energy deposition from airborne fission products. There is also no discussion as to what criteria are to be used to select each fission-product group and how many such groups are used for beta energy-deposition calculations from the airborne fission products. In summary, the implementation of the model within the code is uncertain because information and data needed for exercising the code are not described in the manual (Ref. E.23-1).

E.23.4. Results of Model Sensitivity Studies

None were found.

E.23.5. Results of Benchmarking/Validation Studies

None were found.

E.23.6. Identified Deficiencies and Options for Model Improvements

Many of the assumptions made in constructing the model are ad hoc and have not been justified. The documentation in the SCDAP/RELAP5 code manual (Ref. E.23-1) is inadequate and needs attention.

E.23.7. Importance of Model to Prediction of Dominant Phenomena

A calculation of the decay-heat deposition is required to predict the structure temperature under all accident conditions. It is important to account for decay energy from fission products carried in the vapor space as an aerosol or deposited on the structure surface.

E.23.8. Technical Adequacy of Model

The model is considered to be zeroeth order. Our current state of knowledge with respect to energy deposition in the coolant and the structures connected to the volume because of fission-product transport and decay is much better than the simplified treatment used in the model would suggest. The approach may be adequate for the purposes intended in the code because the energy-deposition rate may be relatively small compared to the fission-product, decay-heat rate or the convective heat-transfer rate to or from a given cell. However, an assessment of the relative importance of the different contributions could not be identified in the documentation. The general approach taken in the energy-deposition model seems to be adequate, and improvements would be warranted only if more detailed computations had indicated otherwise.

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E.24. Decay-Heat, Gamma-Attenuation, Complete-Absorption Model

E.24.1. Model Description and Pedigree

Gamma energy attenuation in a given heat structure is approximated by assuming exponential penetration with a distance and assigning both a characteristic gamma energy for the radioactive decays and materials properties (mass attenuation coefficient, μ, and density, ρ), for each structure provided from input. Different geometry heat structures are treated as a slab, and the local volumetric energy absorption rate is approximated by an exponential function with incident gamma energy flux, attenuation coefficient $\rho\,\mu$, and slab thickness as the parameters. The total energy-deposition rate in a slab materials is obtained by integration of the local rate.

E.24.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

The physics of gamma energy attenuation are well established. The difficulties arise only in treating rigorously complex geometries. Data on mass attenuation coefficients for common structural materials are also available. The overall treatment is reasonable for intact slab geometry, although there may be error in using equations derived for other geometries (i.e., cylindrical). No foundation laid in the documentation (Ref. E.24-1, pp. 6-5 to 6-6) indicates that the incident gamma energy flux should be considered to be collimated (i.e., in the form of a beam). As a matter of fact, this idealization is inconsistent with gamma energy release from fission products on the surface of a heat structure (Ref. E.24-1, p. 6-2). The directional distribution of radioactive heat decay is expected to be much more isotropic than collimated. The error in predicting the local and total volumetric energy absorption rate by making this approximation has not been assessed.

E.24.3. Implementation Within the Code

There is no discussion in the documentation (Ref. E.24-1, p. 6-5) of how the "characteristic gamma energy" and "materials properties of the structure" are to be weighted to insure that the local generation rate computed on the basis of detailed integration over the energy spectrum is the same as one would obtain employing mean characteristics. This issue arises because the gamma-attenuation coefficient of a materials varies with energy. Without appropriate weighting over the energy spectrum, there is no assurance that Eq. (6-4) or (6-5) in Ref. E.24-1 would predict the same local or total volumetric absorption rate as the ones based on the integration over the energy spectrum. This is an important issue and appears to have been overlooked by the code developers.

E.24.4. Results of Model Sensitivity Studies

None were found.

E.24.5. Results of Benchmarking/Validation Studies

None were found.

E.24.6. Identified Deficiencies and Options for Model Improvements

It is well established in the literature that the local volumetric energy absorption rate due to an isotropic plane gamma-ray source obeys an integroexponential and not an exponential law. The approximation used will underpredict the gamma absorption rate, particularly for relatively thin heat structures. At least some mean direction should be estimated for the gamma beam to insure that the absorption rate predicted for the collimated gamma beam flux is the same as for an incident isotropic gamma energy flux.

Specific guidance to the user is needed of how the characteristic gamma energy and attenuation coefficient are to be prescribed to correctly calculate the absorption rate.

The model developers have recognized (Ref. E.24-1, p. 6-7) that if gamma energy is not fully attenuated in the cell containing the initial radioactive decay, only partial attenuation will occur. Even though an analysis that could handle such partial attenuation is feasible and could be developed, the model was judged by the code developers to be unwarranted at this time and was not implemented. Arguments against inclusion of a partial absorption model were not presented, but from the discussion, the main reason appears to be the difficulty in developing logic of how to track partial absorption, escape, and reabsorption of the gamma radiation by the heat structures in a cell.

E.24.7. Importance of Model to Prediction of Dominant Phenomena

A calculation of the volumetric heat absorption rate in heat structures is required to calculate the temperature under all accident conditions, particularly when the material relocates.

E.24.8. Technical Adequacy of Model

The state of knowledge of volumetric gamma energy absorption in heat structures is much better than suggested by the simplified treatment in the model. However, the approach may be completely adequate for SCDAP/RELAP5 purposes because of the large uncertainties in predicting fission-product deposition rates on the heat structures after a severe core-disruptive accident. The justification for the simplifications made are lacking, and the important model parameters are not described properly. The ultimate judgment on the technical adequacy of the model will have to await comparisons of model predictions with either more detailed phenomenological models and/or separate effect tests.

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E.25. Severe-Accident Thermal Hydraulics

E.25.1. Model Description and Pedigree

In general, most of the heat-transfer and friction models are from the standard RELAP5 package. The specific models in RELAP5 were not reviewed by the Peer Review Committee, but the models used in conjunction with SCDAP are largely based on pipe flow correlations. A key assumption is that by using a properly defined hydraulic diameter, the models will produce reasonable results. RELAP5 has undergone significant assessment and review, and the strengths and weaknesses of the modeling are generally known.

When the RELAP5 models are used, SCDAP passes hydraulic diameter information to RELAP5, and the standard RELAP5 correlations, which have not been modified specifically to address severe-accident conditions, are used. The correlations and the interphase exchange modeling are used in the solution of the mass, energy, and momentum equations. The RELAP5 models are used until a cell is blocked. From that point on, thermal-hydraulic calculations are not performed for that cell, even if the blockage melts and relocates.

In some cases, SCDAP has its own models for heat transfer.

The wall friction model is modified (Refs. E.25-1 and E.25-2) when a debris bed is present. Currently, the only modification to the wall friction model is to change the hydraulic diameter from a value corresponding to a rod-like diameter to a value corresponding to porous debris. The RELAP5 wall friction model is then used to determine the liquid and vapor wall friction coefficients needed in the momentum equations. The hydraulic diameter (d_h) corresponding to porous debris is calculated from

$$d_h = 4 \text{ (bed fluid volume)} / \text{(surface area of particles)} .$$

The porosity and particle diameter of the debris resulting from fragmentation is assumed to be the same as that formed by thermal shock during the TMI-2 accident. Analysis (Refs. E.25-3 and E.25-4) of this debris determined that it had an average porosity of 0.54 and an average particle diameter of 0.87 mm. The characteristics of the TMI-2 debris varied spatially, but these variations are not taken into account in the modeling.

The wall heat-transfer model is modified (Refs. E.25-1 and E.25-2) when a porous debris bed is present. Three heat-transfer regimes are identified, and a different equation for the rate of heat removal is used for each regime. The regimes are dryout ($\alpha_g > 0.9999$), quenched ($\alpha_g < 0.9999$ and $T_{debris} < T_{sat}$), and transition between dryout and quenched ($\alpha_g < 0.9999$ and $T_{debris} > T_{sat}$). These regimes and equations are used instead of the RELAP5 wall heat-transfer model when a porous debris bed is present. The dryout regime assumes all heat is transferred to the gas and that the heat-transfer coefficient between debris and gas is infinite; the gas is instantly heated to the temperature of the debris. The quenched regime assumes all the heat is transferred to the liquid, and the equation has two parts. The first part assumes all the heat generated per unit volume (P) in the debris is immediately transferred to the liquid. The second part is applicable when the rate of heat transfer is increased by a decreasing coolant pressure, and thus, a decreasing saturation temperature. The variable h_s is assumed to have a constant value of 1000 W/m^2K. The transition regime assumes that all heat is transferred to the liquid; this assumption is based on the idea of a quench front passing through the debris. The model assumes that debris at the location of the quench front immediately transfers all of its stored energy to the liquid, and thus, the rate of heat removal from the debris is proportional to the velocity of the quench front.

Special treatment of melt quenching is also used. The quenching of melt as it relocates into the lower plenum is treated in a parametric manner using one of two options. In the first option, the user specifies a quenching time, i.e., the time period during which the melt will quench provided there is sufficient water. By using a thermal equilibrium assumption, the state at the end of the quench time can be determined. The integral heat transfer to the coolant is then passed to the RELAP5 thermal-hydraulic model. The second option uses the assumption of no interaction of the melt with the water, which results in a stratified configuration in the lower-plenum.

E.25.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena

In severe accidents, most of the flow regimes in the early phases of the accident are treated by RELAP5. This includes both single-phase and two-phase flow over a wide range of pressures. For intact or nearly intact geometry, it is expected that the RELAP5 models will yield reasonable results.

As discussed above, heat transfer and fluid flow in debris beds are not treated explicitly by RELAP5. While correlations for porous media flow do exist, these are not implemented in RELAP5. An attempt to incorporate such models led to numerical instabilities (Ref. E.25-5). The validity of using pipe flow correlations has not been demonstrated either analytically or through comparison to experimental data. However, the flow resistance that is calculated using the current approach is of the correct order of magnitude (Ref. E.25-5).

The modeling of counter-current flow in the hot legs requires that the standard RELAP5 models be supplemented by "fudge factors" and special nodalization to produce the desired results. The merits of this approach have been reviewed independent of the SCDAP/RELAP5 Peer Review (Ref. E.25-6). The modeling is not based on first principles and requires experimental data to obtain the correct fudge factor and geometrical modeling.

The modeling of multidimensional flow in the vessel uses a nonconservative simplification to the momentum equation that may cause the predicted flow patterns to be nodalization dependent. For most cases, this simplified treatment should be adequate.

The modeling of melt water interactions when melt pours into the lower plenum is not mechanistic. The modeling may produce bounding estimates of the expected behavior. However, the full range of potential behaviors cannot be calculated.

E.25.3. Implementation Within the Code

The implementation of the modeling for intact or nearly intact geometry is considered to be acceptable. Even though information transfer from SCDAP to RELAP5 is temporally explicit, instabilities are unlikely if the geometry changes slowly.

For degraded geometry, melt-water interactions, and quenching, the current implementation can lead to numerical difficulties. The results of the user survey indicate that such problems have appeared even in experimental assessment calculations (Refs. E.25-7 and E. 25-8).

E.25.4. Results of Model Sensitivity Studies

A sensitivity study was performed to exercise both options of the melt-water interactions model (Ref. E.25-9). The results of this study produced results consistent with the modeling. However, the use of either option only yields a bounding estimate of expected behavior.

For the case of superheated steam slowly flowing (< 0.1 m/s) through a region changing from intact rods to debris, the flow resistance in the region is calculated to increase by a factor of 100 as a result of the change in configuration of the region. This increase in flow resistance is consistent with measurements of flow resistance in debris beds (Ref. E.25-10). An attempt was made to use Ergun's wall friction model (Ref. E.25-10) instead of the RELAP5 model when a debris region was detected. The results were similar to the measurements and the RELAP5 model results for superheated steam slowly flowing (< 0.1 m/s) through a region changing from intact rods to debris. When two-phase calculations were made with Ergun's wall friction model, instabilities and code failures occurred. When Ergun's model was tested in the code, there was no time for a careful review of the implementation as well as a careful assessment of the model. Thus, it was decided to use the current RELAP5 model along with the hydraulic diameter changes. This approach did not result in instabilities and code failures.

Sensitivity studies have been performed to assess the modeling of multidimensional flow (Refs. E.25-11 and E.25-12). These studies indicate that the modeling is reasonable.

E.25.5. Results of Benchmarking/Validation Studies

An assessment of the multidimensional flow predictions for blockage conditions was performed (Ref. E.25-13). For a range of blockage conditions, the crossflow junction model produced reasonable results.

An assessment of natural circulation predictions was performed (Ref. E.25-14) through comparison to Westinghouse 1/7-scale PWR experiments. The predicted thermal-hydraulic behavior in the hot legs and steam generators compared reasonably well with the data. Vapor temperatures were overpredicted in the core and underpredicted in the upper plenum. The authors suggest that the reason for the discrepancies may be due to the simplifications associated with simulating three-dimensional flows with a one-dimensional code. The authors recommend that additional nodalization studies be performed to understand the sensitivity of the natural circulation flows to the nodalization scheme. A comparison of the results for the quenching of a debris bed by bottom flooding with the laboratory data shows poor agreement.

E.25.6. Identified Deficiencies and Options for Model Improvements

Intact and Nearly Intact Core Modeling. The heat-transfer modeling of laminar forced convection flow for vapor, as well as for turbulent-free convection flow for vapor, are viewed to be deficient. This judgment is based on the assessments in the RELAP5 Models and Correlations document (Ref. E.25-15). The authors of this document recommend further study of this area. This recommendation should be followed because proper treatment of vapor flow is important for calculating natural circulation. The discrepancies noted in the Westinghouse 1/7-scale experiments (Ref. E.25-14) may be partly due to inaccuracies in the heat-transfer correlations for vapor flow. An assessment of the applicability of the single-phase and two-phase correlations to the geometries and range of conditions of interest needs to be made.

Core Debris Region Modeling. For dried-out porous debris beds, heat transfer is not modeled in a mechanistic manner; the modeling assumes instantaneous equilibrium. Thus, potentially important rate effects that may influence the flow behavior will not be calculated. Large debris regions that transverse multiple RELAP5 volumes may yield erroneous results. The modeling is deficient and should include rate effects in a mechanistic manner. Reference E.25-16 provides an approach to accomplish this recommendation.

For quenching cases, instantaneous equilibrium is assumed between the water and debris. After quenching, all heat is immediately transferred to the coolant. Depressurization effects are accounted for through the use of a constant heat-transfer coefficient. Again, important rate effects are neglected. The modeling is deficient and should include rate effects in a mechanistic manner. Reference E.25-16 provides an approach to accomplish this recommendation.

The correlations used for calculating degraded geometry and core debris region friction losses are deficient. Pipe flow correlations that provide a hydraulic diameter modified for the debris conditions are used with SCDAP. Correlations applicable to flow through porous media should be implemented.

Some of the heat-transfer coefficients [e.g., Eqs. (3-228) and (3-229) of Ref. E.25-17] are not defined. Also, Ref. E.25-17 uses inconsistent nomenclature at times (e.g., "P" for pressure at one point and "P" for power at another).

Lower-Plenum Region Modeling. The modeling of the interactions of molten material with water in the lower plenum is treated in a parametric manner. Only approximate or bounding calculations can be performed. Best-estimate calculations of the expected behavior cannot be performed within the scope of the current modeling because important rate effects are neglected. The modeling is deficient and should include rate effects in a mechanistic manner.

E.25.7. Importance of Model to Prediction of Dominant Phenomena

In the early phases, the thermal-hydraulic models are extremely important. As the core degrades and flow is reduced, the models become less important. However, in the latter stages ,when the melt relocates into the lower plenum, the modeling is again important.

E.25.8. Technical Adequacy of Model

This section has discussed the severe-accident, thermal-hydraulic models in SCDAP/RELAP5. The basis for judging technical adequacy is discussed in Sections 1 and 2. An important evaluation criteria used in this assessment is that the severe-accident, thermal-hydraulic models should yield best-estimate predictions.

The use of RELAP5 models and correlations in the code simulation of the first two intervals of the severe accident is acceptable. Calculations performed during these first two intervals would be expected to produce best-estimate results.

As core degradation progresses, the applicability of these models decreases. Also, SCDAP models, which are relatively simple, are used to predict heat transfer to the coolant. The models are judged to be largely parametric without sufficient validation to demonstrate applicability for all intended applications of the code. The models will not necessarily yield best-estimate predictions for the range of severe-accident conditions that might be anticipated. As such, the thermal-hydraulic modeling in Intervals 3 and 4 is judged not to be acceptable.

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E.25-4. P. Kuan, "TMI-2 Upper-Core Particle Bed Thermal Behavior," EG&G report EGG-TMI-7757 (August 1987).

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E.26. Options for Additional Models Currently being Developed or Upgraded

E.26.1. BWR Control-Blade and Channel-Box Component Modeling Options

E.26.1.1. Model Description and Pedigree. There are significant differences between the existing SCDAP component and BWR control-blade and channel-box components. Therefore, after carefully examining different options, a decision has been made to convert the CORA experiment-specific model into a new BWR control-blade and channel-box component for SCDAP as a preferred method (Ref. E.26.1-1). The reason for this decision is that overall, the CORA experiment-specific model provides a better physico-chemical description of the phenomena. The model accounts for the fact that the stainless-steel control blade must melt and relocate before it can interact with the zircaloy channel box. The model is based on slab geometry and includes the effect of B_4C/stainless-steel interactions to accurately predict control-blade relocation. Finally, the model accounts for the effect of stainless-steel/zircaloy interactions. The new BWR control-blade and channel-box component for SCDAP has five temperature nodes [i.e., two in the channel box (zircaloy), one in the blade sheath (stainless steel), one in the rodlets (stainless steel), and one in the absorber (B_4C)]. The control blade and channel box are modeled within a single SCDAP component so that these structures can interact. The new BWR control-blade and channel-box component has sufficient detail to capture physico-chemical phenomena but does not require excessive computer resources. The new component model represents all geometries of interest, including full-size BWR cores and experimental facilities.

E.26.1.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The approach appears to be reasonable because it contains sufficient detail to model important physico-chemical phenomena and B_4C/stainless-steel interactions.

E.26.1.3. Implementation Within the Code. The new component model has been implemented within SCDAP/RELAP5, Version 7Q (released September 1991), of the code. The subroutine is called once per timestep.

E.26.1.4. Results of Model Sensitivity Studies. None were found.

E.26.1.5. Results of Benchmarking/Validation Studies. Only very limited testing of the model has been done. Preliminary results show some inconsistencies. According to Handout #2-3 at SCDAP/RELAP5 Peer Review Committee Meeting #2 (Ref. 26.1-1), the modifications of the new control-blade and channel-box component model and the incorporation into SCDAP should have been completed on May 15, 1992; however, no reports have been submitted to the Committee. Therefore, it does not appear that an assessment will be completed in time for inclusion into a final summary report. Model details have also not been reviewed by the Committee.

E.26.1.6. Identified Deficiencies and Options for Model Improvements. No deficiencies have been identified because the model has not been extensively exercised. Some preliminary results of structural and interstitial temperatures vs RELAP5 volume number have been presented in Ref. E.26.1-2. A comparison of the calculated and measured axial temperature distribution obtained from the earlier model for CORA-18 BWR bundle heating and melting experiments shows that the calculated temperatures in the lower half of the bundle of the experiment are significantly higher than those measured (Ref. E.26.1-3). A comparison of the SCDAP/RELAP5 predictions (using BWR-specific models) with the DF-4 experimental data yielded good agreement between calculations and the data (Ref. E.26.1-4). This agreement was obtained only after a realistic temperature for the B_4C/stainless-steel eutectic was used in the control relocation model. Thus, the validation of the standalone control-blade and channel-box component model using small-scale integral test data does not appear to be conclusive.

E.26.1.7. Importance of Model to Prediction of Dominant Phenomena. Correct modeling of control-blade and channel-box modeling is important to capture structural liquefaction, material dissolution, and eutectic relocation phenomena that could influence accident progression by forming a blockage lower down in the core. Such a blockage could affect coolant flow, thermal behavior, and hydrogen production.

E.26.1.8. Technical Adequacy of Model. The new component model is a substantial improvement to the original treatment in the SCDAP/RELAP code. Model integration into this code is needed. Validation and testing of the model are also required.

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E.26.1-4. J. K. Hoh and C. M. Allison, "DF-4 Analysis Using SCDAP/RELAP5," *Nuclear Technology* **98**, 149–159 (May 1992).

E.26.2. Zirconium/Inconel Eutectic Model For Grid Spacers

E.26.2.1. Model Description and Pedigree. The model for zircaloy/Inconel interaction at grid spacers (Ref. E.26.2-1) is based on the separate effects experiments of Hofmann et al., at KfK (Ref. E.26.2-2). The rate of growth of the eutectic reaction zone is described by parabolic rate equations, and the effect of zirconium oxide layers in delaying the onset of the reaction is taken into account. Simplifying assumptions are made to map the growth of the reaction zones in the spacer grid and cladding onto a one-dimensional system so that the experimental correlations can be applied to the in-core geometry; effectively, the directions of growth of the reaction zone are prescribed. It is also assumed that the rate of growth of the reaction zone depends only on its volume, not on its configuration, provided that the area of contact of the Inconel and zircaloy is the same.

The model calculates the amount of Inconel and zircaloy dissolved and the time that the grid slumps due to liquefaction. It is the first known attempt to model the effect of the zircaloy/Inconel reaction in a large-scale computer code.

E.26.2.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The approach appears reasonable, and the rate equations are based on the latest available data.

E.26.2.3. Implementation Within the Code. No information is available in the documentation.

E.26.2.4. Results of Model Sensitivity Studies. None were found.

E.26.2.5. Results of Benchmarking/Validation Studies. The quoted reference describes the results of some demonstration calculations involving a comparison with CORA-2 data that appear to show that the model is behaving reasonably. It would be helpful to check the performance of the model in predicting other CORA tests, e.g., with slow heatup rates; internal pressure measurements for control rods are available, which indicate the time of breach. Validation against data from other facilities would give additional independent evidence on the performance of the model.

E.26.2.6. Identified Deficiencies and Options for Model Improvements. INEL noted uncertainties in (1) the area of contact between the grid and cladding, (2) the direction of spreading of the reaction zone, (3) the slump criterion for liquefied material, (4) the rate of spreading of the reaction zone after a substantial change from its initial configuration, and (5) the configuration of slumping material and how it interacts with a surviving grid spacer. A more detailed experimental analysis will probably be needed to evaluate the importance of these uncertainties.

E.26.2.7. Importance of Model to Prediction of Dominant Phenomena. Early relocation of zircaloy cladding, which has formed a low-melting-point eutectic with Inconel, has three main effects: (1) transferring thermal energy to a cooler part of the core, thereby affecting the axial temperature distribution; (2) forming a blockage lower down in the core, which may impede coolant flow; and (3) removing oxidizable material from a high-temperature to a low-temperature region where the rate of oxidation (and therefore of hydrogen production) would be lower, assuming sufficient steam availability. Thus, thermal behavior, melting, and hydrogen production can be affected.

E.26.2.8. Technical Adequacy of Model. This new model is a substantial improvement to the treatment of spacer grids in the code. The modeling may be considered first order. It is too early to give a formal categorization; more validation and testing are required.

REFERENCES

E.26.2-1. L. J. Siefken and M. Olsen, "Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," EG&G report EGG-SSRE-10209 (September 1992).

E.26.2-2. E. A. Garcia, P. Hofmann, and A. Denis, "Chemical Interaction Between Inconel Spacer Grids and Zircaloy Cladding: Formation of Liquid Phases Due to Chemical Interaction and Its Modeling," Kernforschungszentrum Karlsruhe report KfK 4921 (1992).

E.26.3. Lower-Head Failure

E.26.3.1. Model Description and Pedigree. Heat transfer from the core debris (solidified or liquefied) to the vessel wall and to the penetrations determine the modes in which the vessel lower head can fail (Ref. E.26.3-1). Examples of the modes in which the vessel can fail are: tube ejection, tube rupture, localized failure of the vessel, and global failure of the vessel. At present, only heat transfer from solidified core debris to the vessel wall is considered and that, too, in a parametric way. More recently, as part of the activity in support of the Savannah River reactors, a correlation for heat transfer from liquefied debris to the vessel wall has been included. However, this correlation is flawed because it is applicable to a circular trough and not to a spherical cavity.

For heat transfer from solidified or liquefied debris to a structure, gap conductance is defined. Gap conductance between solidified debris and structure is specified by the user. For liquefied debris, a natural convection heat-transfer type of correlation is used. More recently under another research program sponsored by the NRC, models for investigation of modes and timing of vessel failure have been developed. If these models are implemented in SCDAP/RELAP5, they should enhance the capacity of the code in analyzing the modes and timing of vessel failure.

E.26.3.2. Applicability/Physical Reasonableness of Model to Prediction of Dominant Phenomena. The present model is parametric and is of very limited value. At present, no models are available in the code with respect to localized failure of the vessel at penetrations. The inclusion of the models developed for Savannah River reactors should mitigate the situation somewhat with respect to heat transfer to the lower head.

E.26.3.3. Implementation Within the Code. The temperatures in the vessel wall are calculated with COUPLE, a two-dimensional conduction code that is used in conjunction with SCDAP/RELAP5.

E.26.3.4. Results of Model Sensitivity Studies. None were found.

E.26.3.5. Results of Benchmarking/Validation Studies. None were found.

E.26.3.6. Identified Deficiencies and Options for Model Improvements. The current model for heat transfer from solidified core debris to the vessel is at best parametric.

The correlations used for heat transfer from liquefied debris to the vessel lower head (developed as part of the Savannah River Reactor Project) are for a cylindrical trough and not for a spherical cavity. No models exist in the code to predict failure of the penetrations. Furthermore, the code does not provide output on probable modes of vessel failure.

E.26.3.7. Importance of Model to Prediction of Dominant Phenomena. The model will provide an important input to Accident Management studies. The mode and timing of vessel failure will determine the manner and rate at which core material is released from the vessel into the containment.

E.26.3.8. Technical Adequacy of Model. Currently, the model for vessel lower-head failure is severely deficient with respect to prediction of the mode and the timing of vessel failure.

REFERENCES

E.26.3-1. L. J. Siefken and R. L. Moore, "Extensions to SCDAP/RELAP5/MOD2 Debris Analysis Models for the Severe Accident Analysis of SRS Reactors Final Design Report," EG&G report EGG-EAST-8508 (September 1990).

APPENDIX F

SCDAP/RELAP5 User Survey

SCDAP/RELAP5 users are asked to provide important information for an on-going independent peer review of the code. This survey asks users to answer questions related to the technical adequacy of the code. There are four sections. The first section explores code technical adequacy issues related to "design objectives" that have been set forth by the US Nuclear Regulatory Commission (USNRC). The second section investigates code technical adequacy in relation to the USNRC-specified "targeted applications." The third section asks for a ranking of "level-of-completeness" of the code with respect to design objectives and targeted applications and for input on current code-development milestones. The fourth section is a single sheet asking for user-related information.

Please do not be discouraged by the length of this survey. Try to concentrate on and give detailed answers to those areas you have the most experience with. Be as specific as possible. We want you to provide as many answers as you can, even though there are many questions you may not be able to address. Also, please state to which code version each of your answers applies (many users have experience with several versions of the code).

Responses to surveys often give an unwarranted negative impression of a product because users tend to concentrate on deficiencies rather than praiseworthy aspects. We encourage you to provide positive, as well as negative, comments in this survey. Tell us what you found good about the code, as well as what you found were deficiencies.

This is a limited-time opportunity for code users to provide input to a comprehensive review of the SCDAP/RELAP5 code. Thank you for your participation.

I. SCDAP/RELAP5 Code-Design Objectives

1. Modeling detail shall be capable of representing key and important phenomena of severe-accident experiments, the TMI-2 accident, and anticipated plant accidents and transients.

   Question: What experience can you relate that shows that the user can adequately model PWR and BWR reactor coolant systems, operator actions, and experimental facilities with the code?

   Your reply (state code version): ______________________________

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a. Expected modeling uncertainties should be comparable to uncertainties in integral severe-accident experiments and TMI-2 accident conditions and results.

Question: Are the uncertainties in important parameters calculated by the code less than or equal to measured values? (For example, if the uncertainties in the measured bundle temperatures and associated boundary conditions are +/-20%, would you expect to find that the associated code-computed values are also within +/-20%?) Please provide examples of your findings.

Your reply (state code version): ______________________________

b. User-defined parameters, other than those needed to define experiment or plant-unique features, should be eliminated where experimental or other credible bases exist to define those parameters.

Question: What user-defined parameters, other than those noted, have you encountered that should be eliminated from the code input?

Your reply (state code version): ______________________________

2. The code should provide reasonable predictions of the in-vessel melt progression phenomena during the course of a severe accident. It should permit estimates of the uncertainties of severe core-damage predictions without requiring modifications to the code.

Question: What do your analyses show regarding the code's capability to predict major trends for dominant phenomena based on assessment against integral facility data? How well does the code predict values of important parameters associated with dominant phenomena within measurement uncertainty when assessed against integral facility data? Did you employ a "frozen" released version of the code without any code modifications made during the period of assessment?

Your reply (state code version): ______________________

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3. The code should be applicable for severe core-damaged studies under various accident sequences for both pressurized water reactors (PWRs) and boiling water reactors (BWRs).

Question: What experience can you relate that demonstrates that the code can predict core damage resulting from risk-dominant accident sequences identified by probabilistic risk assessment studies for both PWRs and BWRs? Are physical models, as well as component models, adequate to accurately predict dominant phenomena?

Your reply (state code version): ______________________

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4. The code should be robust, portable, and fast running.

Question: While runtime is certainly machine dependent, what have you found regarding the "reasonableness" of the code runtime? How has this affected your ability to perform sensitivity and/or uncertainty analyses for the phenomena/conditions the code is designed to model? What is the fraction of the code runtime compared to the time required to perform an entire analysis? Please give as many examples as possible.

Your reply (state code version): ______________________________

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Question: How effective are the code-user guidelines and other lessons-learned information in the code manual? How useful is documentation in setting up a plant model (input deck) to truly represent a full-scale LWR plant and successfully perform plant calculations?

Your reply (state code version): ______________________________

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a. The code should not abort prematurely because of user-input errors or numerical nonconvergence but should exit with sufficient diagnostic messages for users.

Question: What has your experience been in this area? Did you encounter problems where the code did not converge? If so, please describe. Does the code output give you sufficient detail to determine where most of the computation time was spent (what model, subroutine, limiting parameter, or control volume, for example)? Overall, how would you characterize the code's "user friendliness"? Give examples, if possible.

Your reply (state code version): ______________________

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b. Numerical precision should be compatible with modeling precision. Spatial convergence should be compatible with the modeling scale. Timestep control should be automatic.

Question: What has your experience been with the SCDAP/RELAP5 numerical precision, spatial convergence (noding sensitivities), and timestep control (timestep sensitivities)?

Your reply (state code version): ______________________

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c. The code should be transportable for mainframe and workstation computing machines.

Question: How transportable is the code? Please give examples from your implementation experience, if possible.

Your reply (state code version): ______________________

5. The maintenance of the code should follow accepted quality assurance standards for configuration control, testing, and documentation.

a. All code changes should be controlled and verified by redundant means.

Question: Do you have any comments on this design objective? Please comment on whether you believe the standards currently being applied are sufficient.

Your reply (state code version): ______________________

b. Testing standards and benchmarks should be defined for all versions released for production applications.

Question: Do you have any comments on this design objective? Please describe further the level of testing you believe to be appropriate and whether the standards currently being applied are acceptable.

Your reply (state code version): ______________________

c. Documentation should define the theoretical bases, limits of applicability, and testing or assessment results of the code.

Question: What strengths and weaknesses do the code documents have in these areas?

Your reply (state code version): ______________________

II. SCDAP/RELAP5 Targeted Applications

1. Experimental analysis and support for in-vessel, severe-accident experimental programs such as CORA, PBF, LOFT, and NRU.

   Question: What has your experience been in applying the code for these facilities? Do you believe the code provides reasonable predictions of dominant in-vessel, severe-accident phenomena ("reasonable" here means that compared to experimental data, the calculated results will be within the experimental uncertainty bands)?

   Your reply (state code version): ______________________________

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2. LWR plant analysis with and without water addition.

   Question: What has your experience been in applying the code to analyze LWRs, and do you believe the code can provide reasonable predictions of associated dominant phenomena with and without water addition? Also, have you performed full-plant analyses that included the code's fission-product-release and transport models? If so, did you find the code performance acceptable?

   Your reply (state code version): ______________________________

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3. Selected detailed analysis for specific technical issues—lower-head-failure analysis, influence of water addition, natural circulation, hydrogen generation upon reflood, and accident management evaluations.

Question: What has been your experience in applying the code to detailed analyses of these specific technical issues and predicting the associated dominant phenomena? What related assessments have you performed against experimental results in these areas?

Your reply (state code version): ______________________________

4. MELCOR benchmarking and assessment.

Question: What benchmarks and assessments of the MELCOR code in-vessel behavior have you performed with SCDAP/RELAP5 (at least for integral experiments)? What were your results? How would you characterize the use of SCDAP/RELAP5 to benchmark the MELCOR code?

Your reply (state code version): ______________________________

5. TMI-2 accident evaluation.

Question: What evaluations of the TMI-2 accident have you performed with the code? Did you obtain a reasonable prediction of the dominant phenomena? (Again, "reasonable" means that compared to the TMI-2 data, the calculated results will be within the measured uncertainty bands.) What were your results, in summary?

Your reply (state code version): ______________________________

III. Ranking of Code "Level-of-Completeness"

1. You have made separate replies to questions related to whether or not the code satisfies its design objectives and targeted applications. A ranking (list) of code objectives and targeted applications would provide your overall appraisal of the completeness of the code.

   Question: What is your ranking of the code "Level-of-Completeness" with regard to satisfying code objectives and code-targeted applications? Please list the items that you addressed in I and II with "most complete" first and "least complete" last. For example:

   Example Ranking :

II-4.	MELCOR Benchmarking	*(most complete)*
I-3.	Applicability for both PWRs and BWRs	
	(all the rest that one responds to...)	
I-5b.	Numerical precision, ...	*(least complete)*

   (Note: this is a fictitious example to show format for ranking.)

   Your reply (state code version): ______________________________

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2. Current code-development plans call for the code early-phase models to be completed within the next year and a half and late-phase models to be completed in the 1995-96 timeframe.

Question: How does this schedule match with your expectations and the requirements for your planned workscope?

Your reply (state code version): ______________________________

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IV. USER INFORMATION REQUEST

DATE SURVEY COMPLETED:______________________________

NAME:______________________________

ORGANIZATION:______________________________

ADDRESS:______________________________

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TELEPHONE:______________________________

FAX:______________________________

E-MAIL:______________________________

COMPUTER(S) USED TO RUN CODE:______________________________

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COMPUTER OPERATING SYSTEM(S):______________________________

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VERSION OF CODE USED ______________________________

LATEST VERSION OF CODE IMPLEMENTED AT YOUR SITE:______________________________

ADDITIONAL COMMENTS:______________________________

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Please return completed survey to:

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