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MAY 1973

SODIUM-COOLED REACTORS SAFETY ENGINEERING PROGRAM

THIRD QUARTERLY REPORT
FEBRUARY-APRIL 1973

U. S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-893 TASK 2

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**SODIUM-COOLED REACTORS
SAFETY ENGINEERING PROGRAM
THIRD QUARTERLY REPORT
February—April 1973**

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Task 2

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ABSTRACT

The Demonstration Plant will demonstrate the safe, economical generation of electric power from a liquid-metal cooled fast breeder reactor. Many aspects of the safety of such a plant have been under study for several years.

The existing LMFBR activity concerning safety includes both analytical and experimental programs. A major requirement is the establishment of mechanisms for the guidance of the National Safety Program in the integration of these analytical and experimental programs. This guidance must establish specific objectives, priorities, and schedules for the programs, and it must provide a mechanism for identifying and correcting whatever problems arise before they interfere with the schedule for the design and construction of the Demonstration Plant.

The purpose of this project is: (1) to bring into sharp focus both the requirements or needs for specific technical information related to safety, and the results of previous studies, wherever performed; (2) to recommend to AEC-RDT a course of action which will provide the information still required.

This project began on July 1, 1972, as a separate activity. However, it is in part an extension of previous work under Tasks C-1, C-2, G-10, and G-12 of Contract AT(04-3)-189, Project Agreement 10. In addition to these areas which focused primarily on core safety, the current program is broadened to encompass all phases of reactor and plant safety.

This report covers the activity during February, March, and April 1973.

1. SUMMARY

1.1 PROGRAM INTEGRATION

The final version of the procedure "Proposed Mechanism for Integrating Reactor Manufacturer's Safety Needs into the National Safety Program" has been transmitted to DRDT. The essential features of this procedure are strong integration of design and development functions, independent "third party" evaluations, assessment of both demonstration and commercial LMFBR needs, and the application of decision theory to areas formerly treated by "engineering judgment."

The General Electric liaison engineer appointed to the Argonne National Laboratory reported on laboratory activities on a weekly basis and transmitted to GE-BRD design, test and code documents pertaining to the LMFBR safety development programs. At the conclusion of this reporting period the appointment of the resident liaison engineer at ANL was terminated.

FY-73 work scope changes, necessitated by funding reductions, were recommended to AEC-SAN. ANL liaison activity was terminated April 15, 1973. Major remaining work scope items are:

1. Safety test data review will continue.
2. Participation in round-robin analyses will continue.

1.2 PLANT SAFETY NEEDS

A listing of generic safety needs for LMFBR central power stations was completed; however, the assignment of priorities was deleted due to reduced work scope. Items required to support bases for plant technical specifications and reflect the near term development trends for commercial plants are included along with an indication of the FFTF design approach and regulatory experience.

The FFTF and Demonstration Plant designs were compared to identify additional safety development programs required for the Demonstration Plant and other LMFBR's as a result of the steam generation system and mode of plant operation.

A literature search was completed for identifying specific needs for various types of sodium-fire safety analyses.

1.3 EVALUATION OF ANALYTICAL AND EXPERIMENTAL PROGRAMS

Convergence problems associated with analysis of pipe break transients with SAS-2A have been resolved and a production version of the SAS-2A code has been installed on the Berkeley CDC system.

Code modifications were completed to permit studies of pipe break transients in a hot-leg pump plant design with FFTF outlet line configuration.

Computer models (using the THTE and SAS-2A codes) of the ANL R-4 experiment are being checked out using preliminary data on capsule design and experimental plans. These models will be finalized and an L-3 post-test analysis performed when the respective final data packages are received from ANL. These analyses in addition to an H-3 post-test analysis are in preparation for a GE/HEDL/ANL/RDT round-robin meeting scheduled for July 25 and 26, 1973.

1.4 REVIEW AND EVALUATION OF SAFETY-RELATED TEST DATA

Work has been initiated on the second summary report on safety development test data review. The following topics are being considered for inclusion in this report:

1. "S" Series test data review
2. ANL Homogeneous Nucleation Theory (relationship to "S" series tests)

3. Fuel motion during transient overpower
4. Sodium liquid-film thickness data
5. Sodium fires and aerosols

A review and technical updating of the post-test analysis and results for the existing GE transient overpower tests will be initiated next quarter with significant effort commencing with start of FY-74.

Work on evaluation of the REXCO-HEP structural response code continued during this reporting period with calculations using three different modeling variants of an LMFBR reactor geometry. Since anticipated continuation of this effort will not be possible due to budgetary restrictions, information from these runs, though incomplete, has been summarized. The study indicates that:

- The stiffness of the support structure has shown to have appreciable effect on distribution of core accident energy.
- Effects of modeling variables such as amount of artificial rotational viscosity in nodes, geometric modeling and material properties treatment, require further assessment.
- Improvements in the present REXCO program which would improve the accuracy and/or versatility of the program include:
 - Provision for "slip" between fluid nodes and elastic-plastic nodes, or between adjacent fluid nodes as specified in the input.
 - Improved and up-to-date documentation of code input requirements with particular attention called to specific restrictions on input.

Monte Carlo techniques have been combined with THTE code thermal analysis in a study of steady-state and transient (severe loss-of-flow) fuel rod failure/reliability. The technique examined the effects of intra-bundle thermal hydraulics and uncertainties in fuel damage limits on the expected number of fuel rod failures during the loss-of-flow condition. The study indicated areas where further effort is required to improve predictions of fuel rod transient reliability. These areas are:

- uncertainties in steady-state temperatures carry through to the transient calculations,
- acceptable statistical techniques for transient fuel rod reliability analysis,
- uncertainties of fuel-failure limits on both material property data and on fuel performance modeling.

Coordination of this work on a national level is needed to eliminate discrepancies that exist through the acceptance of statistical techniques for analyzing steady-state fuel rod performance and use of deterministic "worst case" or "hot spot" techniques for analyzing transient fuel rod performance.

2. SAFETY ENGINEERING

2.1 INTRODUCTION

A primary objective of Safety Engineering is to maintain a high level of expertise in and to remain cognizant of ongoing LMFBF safety development programs. From this basis, Safety Engineering will continue to make recommendations to RDT concerning activities within the safety development program. Safety related test data from all participating laboratories will be reviewed, evaluated and the results summarized in a series of Test Data Review documents. The objectives of these test data reviews are to establish current industry position and to recommend to RDT empirical/analytical requirements for developing information pertinent to a specific safety question (e.g., fission gas release, MFCI, Aerosols, etc.). In-house analytical tools such as THTE (Transient Heat Transfer Code-Version E) and safety analysis codes developed outside GE will be reviewed and selectively applied to the test data reviews, along with comparative (round-robin) analyses of safety tests, and to the analysis of nonprototypic effects on safety test results. The formulation of qualified models and detailed experiment plans at other laboratories will be monitored and predictive analyses performed when necessary to establish applicability of resulting models or data to demonstration plant needs.

The key areas of emphasis for such activities are:

- TREAT data analysis and interpretation
- Application of TREAT data to reactors
- Analysis on nonprototypic effects

Sufficient support-type analysis will be performed to provide a proper context for the above evaluations.

2.2 PROGRAM INTEGRATION

The technical activities described in this section include the preparation and implementation of procedures for integrating with national LMFBF programs as well as providing advanced planning and design information to relate safety development programs to the LMFBF. Additional activities include Quality Assurance, planning and schedule support, and on-site liaison at the national laboratories, initially limited to ANL.

2.2.1 Integration

The final version of BRD's "Proposed Mechanism for Integrating Reactor Manufacturer's Safety Needs into the National Safety Program" has been transmitted to DRDT.

This integration plan was evolved based upon a need for a more definitive effort to integrate the development efforts of national laboratories with design efforts of the reactor manufacturers. The essential features of this procedure are strong integration of design and development functions, independent "third party" evaluations, assessment of both demonstration and commercial LMFBF needs, and the application of decision theory to areas formerly treated by "engineering judgment." It is felt that implementation of this proposed procedure will result in improved efficiency in the national safety program as well as making more evident the decision processes which support the ultimate licensability of the LMFBF.

2.2.2 Liaison

The General Electric liaison engineer appointed to the Argonne National Laboratory reported on laboratory activities on a weekly basis and transmitted to GE-BRD specific design, test, or code data and ANL documents which pertain to the LMFBF safety development programs and test facilities. Control of the documents is maintained through the office of the Safety Engineering (893/2) Technical Project Engineer.

The liaison engineer's reports summarized recent ANL experimental data and analyses not yet published, but directly applicable to Safety Engineering work conducted at GE during this period. His involvement at ANL supported

the GE-BRD activity by his remaining cognizant of the latest laboratory-developed safety-related computer codes and by running sample problems of these codes on ANL computers and discussing and resolving potential operating difficulties with the code authors. On occasions, this free exchange of information and ideas has led to code improvements and adjustments in conformance with the preferences of users not associated with the code development.

The liaison engineer also provided the focal point for exchange of pertinent information from GE-BRD to ANL.

During his tour of duty at ANL, he spent a significant portion of his time participating in the R-Series project. His involvement consisted of performing engineering calculations on the R-Series apparatus and designing a thermal-hydrodynamics computer model representing the once-through TREAT loop with its flow control system and the test section thermal hydraulics. The model is especially tailored for the loop geometry and affords a more detailed representation of the system than would be possible with SAS. The code was completed and has been checked out. Loop operating characteristics have been analyzed with this tool and some operating specifications have been derived from it.

At the conclusion of this reporting period, the appointment of the resident liaison engineer at ANL was terminated as a result of budgetary constraints. It is expected that the personal contacts established by him will be maintained and will continue to prove highly beneficial to future work within the Safety Engineering program.

2.2.3 Quality Assurance

Quality Assurance activities associated with this project can be divided into two categories: 1) the formal Quality Assurance review of documents, procedures, etc., and 2) the informal assessment of quality obtained from measuring analytical techniques and results against standards, other analyses, and test data. This informal assessment is implemented through comparison of results in round-robin analyses with ANL and HEDL, comparison of results obtained from operating with different codes utilized at BRD, and an evaluation of analytical modeling of experiments carried out at the national laboratories.

2.2.4 Administrative

Recommended modifications in work scope for FY-73, necessitated by funding reductions, were summarized in a proposal to AEC-SAN. ANL liaison activity was terminated April 15, 1973. The major remaining work scope items are: 1) Safety test data review will continue; 2) Participation in round-robin analyses will continue.

The Planning and Scheduling Report (Document Schedule) has been issued completing initial planning for FY-73. Timely updates are being made as required.

2.3 SAFETY NEEDS

The technical activities in this section include the development and maintenance of a generic list of demonstration plant safety needs and the acquisition, adaptation and installation of safety analysis codes on local computation facilities. These activities are intended to provide proper focus and integration of fast reactor safety development programs as well as establishing the computational capability for performing the required supporting analyses.

2.3.1 Plant Safety Needs

Specific needs based on generic safety and licensing issues associated with demonstration and commercial LMFBR power plants were identified. However, the assignment of priorities to the identified needs has been deleted due to a reduction in the work scope.

Since the FFTF represents the state-of-the-art in fast reactor plant design and current regulatory requirements, a review of the FFTF design position and licensing experience (as of December 1972) was included for each of the generic areas. Although the different objectives of the FFTF and Demonstration Plants makes each unique, the facilities share many features related to LMFBR safety issues.

A generic format was chosen for general applicability to current large LMFBR central power stations. By necessity, the specific needs identified must be based on unique plant features and the design approach employed. A General Electric Demonstration Plant design was utilized to identify specific needs since the necessary detailed design information was readily available. This list reflects, as appropriate and possible, the safety needs related to establishing bases for the LMFBR Demonstration Plant Technical Specifications, near-term commercial plant development trends, and the FFTF experience in the identified generic areas. Application of this list to alternate plant designs would have to reflect the specific design features of those facilities. For example, The W Demonstration Plant design does not include a reactor auxiliary cooling system, per se, and the needs with respect to steam generator malfunctions and emergency core cooling would require modification.

The generic areas included in the safety needs list are as follows:

1. Fuel, blanket and control rod failure
2. Propagation of failures—fuel, blanket and control assemblies
3. Operation of fuel, blanket and control rods in failed condition
4. Reactivity control and protection system requirements
5. General instrumentation requirements
6. Shielding requirements
7. Coding of vessels for high temperature/nuclear operation
8. Core support structure and clamping mechanism
9. Steam generator malfunctions
10. Primary system seals
11. Sodium fires
12. Environmental and radiological effects of plant discharges
13. Emergency core cooling system (ECCS)
14. Containment and isolation effectiveness
15. Natural environmental hazards
16. Pipe break incidents
17. Materials properties data
18. Utilization of experimental data and analytic methods
19. Hypothetical core disruptive accidents (HCDA)

The Demonstration Plant and other LMFBR's require additional safety development work beyond that required for the FFTF since they will be operated in a different manner and because they include steam generating systems. A comparison has been made between the FFTF and a typical LMFBR. Some of the more significant safety development

areas identified are the effects of high fuel burnup and long term (2-year) fuel operation at high power density, systems to provide protection against sodium-water reactions (including means to detect such reactions before they become hazardous), and emergency cooling requirements and provisions for a reactor system in which the normal heat rejection path cannot be fully utilized for natural circulation cooling.

2.3.2 Sodium Fire Safety Analysis

A literature search was completed and a memo issued summarizing the results of this review. In addition, a study was initiated to identify the specific needs for various types of sodium fire safety analyses. The available information is being critically evaluated to identify additional information requirements based on Demonstration Plant and commercial plant requirements.

2.4 EVALUATION OF ANALYTICAL AND EXPERIMENTAL PROGRAMS

The objective of this subtask is to:

- Collect and utilize all of the analytical methods necessary to perform required safety analyses and evaluations;
- Maintain cognizance of on-going experimental programs,
- Critically evaluate both the analytical methods and the on-going experimental programs with regard to objectives, priorities, and correlations.

This activity provides the general and detailed requirements defining safety development test objectives and analytical-methods work based on plant safety needs. Safety related test data will be reviewed and incorporated into models in a cooperative effort with the national laboratories. Available analytical tools will be applied to comparative (round-robin) analysis of the safety tests and to the analysis of nonprototypic effects on safety test results. Planned safety development programs will be critically evaluated for applicability to plant safety needs. All evaluations will be forwarded to the cognizant program personnel.

2.4.1 Pipe Break Analyses

A convergence problem appeared in the application of the SAS-2A pipe break transients. This problem coincided with the time of initiation of sodium boiling. The problem has now been resolved with the aid of an ANL/RAS and has been verified by a successful run which went well into sodium boiling.

Following resolution of the convergence problems, a production version of the SAS-2A code was stored on the Berkeley (Tape No. 01875, SAS2BNGE) system as of March 30, 1973. This approach will assure the availability of an operable version of SAS-2A for use in round-robin and other predictive analyses.

Analyses to quantify safety needs for an LMFBR which has a single line between the reactor vessel and the pump in the area of pipe break analysis, required some modifications to the hydraulics pipe code developed under PDP. The modifications were completed and verified to provide a proper representation of a Demonstration Plant hot-leg primary-pump design. Pipe break analyses will be performed with this code, as required to support the predictive analysis work.

2.4.2 Predictive Analyses

Information was received from HEDL defining the objectives and milestones for the upcoming R4 pretest as well as the H3 and L3 post-test round robins. The round-robin meeting, originally scheduled for April 26 and 27, has been tentatively rescheduled for July 25 and 26 due to changes in the ANL, GE and HEDL safety development program budgets.

ANL TREAT Test R4 is a loss-of-flow simulation designed to assess early movement of the stainless-steel cladding after fuel rod failure. The test section will utilize seven full length FFTF-type fuel rods containing unirradiated UO_2 fuel. The objectives of the R4 Round Robin pretest analyses are to predict the fuel, cladding and sodium response to the planned transient conditions and to optimize the power cutoff time in order to meet the test objectives.

Preliminary data packages, consisting of R Series information and fuel rod specifications, were received from ANL and HEDL, respectively. These data packages contained geometrical and material specifications for the fuel rods and test section. Based upon this information, a preliminary THTE (Transient Heat Transfer—Version E) and SAS-2A model for the R4 test are being developed. Each model simulates the central and one peripheral rod of the seven-rod bundle as well as the test section liner. The THTE model accounts for azimuthal heat transfer across the bundle; this model will provide the reference transient thermal response of the fuel rods and sodium up to incipient boiling. The SAS-2A model will provide a comparative transient thermal response prior to boiling, and will be relied upon to provide the thermal response during the two-phase heat transfer portion of the test. Preliminary results have been obtained for each model with hypothetical power and flow histories. Final model development and calculations will depend upon timely receipt of the R4 pre-test data package from ANL.

2.4.3 Review and Evaluation of Safety-Related Test Data

A significant part of the Safety Engineering Program charter involves reviewing and evaluating existing data from the LMFBR national safety development program. The review and evaluation involves critical examination of experimental and analytical data, both past and current, from selected areas of fast reactor safety studies. If indicated by the examination, recommendations are made for future testing/analytical requirements.

2.4.3.1 Current Review/Evaluation Studies

Work has been initiated on the second summary report on safety development test data review. The following topics are being considered for inclusion in this report:

1. "S" Series test data review,
2. ANL Homogeneous Nucleation Theory (relationship to "S" series tests),
3. Fuel motion during transient overpower,
4. Sodium liquid-film thickness data, and
5. Sodium fires and aerosols.

2.4.3.2 Future Review/Evaluation Studies

A review and technical updating of the post-test analysis and results for the existing GE transient overpower tests will be initiated next quarter with significant effort commencing with start of FY-74.

2.4.4 Rexco Structural Analysis

2.4.4.1 Introduction

Work on evaluation of the REXCO-HEP structural response code continued during this reporting period with calculations using three different modeling variants of an LMFBR reactor geometry. Since anticipated continuation of this effort will not be possible due to budgetary restrictions, information from these runs, though incomplete, is being summarized herein.

Much of this effort can be regarded as an extension of work with the REXCO-H code reported in GEAP-13921.⁽¹⁾ Like its predecessor, REXCO-HEP is a two-dimensional structural/hydrodynamics code which solves the energy and

continuity equations for compressible media in a Lagrangian coordinate system. The HEP code version utilized is similar to that installed on the CDC-7600 at the Lawrence-Berkeley Laboratory of the University of California. The basic equations and program are described in ANL 7499^(2, 3) with the current codes input requirements being described in informal ANL documentation (REXCO-HEP, Release 1, June 15, 1972).

Familiarization with the REXCO-HEP code required for this task included utilization of the CALCOMP plotting routine and the microfiche output used to obtain the prints showing the mesh distortions. One minor problem in the Berkeley program was detected by GE and corrected by ANL during the course of the work. Unfortunately, these efforts also coincided with major revisions to Berkeley's software packages and numerous hardware problems with the CDC-7600 stretching over a period of three months.

Some major advantages of REXCO-HEP over the earlier REXCO-H code are:

- REXCO-HEP permits "elastic-plastic" modeling of core zones (which permits modeling of regions capable of sustaining appreciable shear loads, such as the core support structure) rather than requiring pure "hydrodynamic" treatment of all regions.
- Unrealistic "hourglass" distortion of the mesh geometry can be suppressed with REXCO-HEP by utilization of an "artificial" viscosity which may be incorporated in the materials modeling at the user's option.

Three calculational models were chosen which utilized each of the above code features. These improved upon the previous REXCO-H model by using an elastic-plastic representation of the core support structure. The material properties of the structure was stainless steel. Initial plans called for additional calculations using a more flexible support structure in order to evaluate the sensitivity of the calculational results to this parameter; however, these could not be carried out due to the early termination of further work. The source term used (in all cases) for the core pressure-volume relation was the same as that used for HCDA No. 1 in Reference 1. The initial pressure of 456 bars would produce 300 MW-sec of core work if allowed to expand to one atmosphere. The selection was arbitrarily done based on expected structural deformations which would permit code evaluation to be based on a relatively severe and possibly limiting type of accident.

2.4.4.2 Modeling

Figures 1 and 2 show the mesh structures used for the two different geometric reactor models used. The difference is only in the lower head region, where the representation shown in Figure 1 approximates the curved lower head of a typical reactor vessel. The major differences from prior REXCO-H representation⁽¹⁾ are:

- The horizontal core support structure is modeled as an elastic plastic material, where previously it was treated as a vessel horizontal plate.
- The sodium below the core support structure is modeled as sodium region whereas previously only its mass was modeled by lumping it with that of the support plate.
- The length of the two-to-one thickness taper of the upper vessel region has been increased to avoid numerical problems previously found with strain calculations at a sudden cross section change.

The core was represented as nine zones, with the seven highest power density regions being used to input the high-pressure source term. Two vessels (the reactor vessel and the vessel closure) and one structural cylinder (the lateral core support cylinder or core barrel) are included in the model. Vessel material properties and elastic-plastic region modeling assume material yield strength of 30,000 psi and ultimate strength of 50,000 psi, corresponding to estimated properties at 1000°F for Type-304 stainless steel.

Figure 1. Reactor Model With Curved Lower Head

Figure 2. Reactor Model With Flat Lower Head

2.4.4.3 Results and Discussion

Two calculations, identified as Model 10 and Model 10A, were carried out using the curved vessel head model of Figure 1. Model 10 did not attempt to utilize the code option which incorporates an artificial rotational viscosity in the fluid nodes to reduce distortion of the mesh lines. The calculation indicated a failure of the outermost element of the vessel lower plate or head at 12.05 msec (calculation cycle 759) into the transient. The geometry of the mesh system shortly before this failure, at 11.1 msec, is shown in Figure 3 and indicates considerable distortion of the grids adjacent to the rapidly expanding core region with a definite flow pattern radially outward just below the reactor vessel closure. The considerable mass of the shielding in the lower portion of the closure has diverted the upward flow from within the core barrel to a radial direction. A noticeable outward radial flow pattern has also been established just above the lower head of the reactor vessel. At this time, however, the strain of the outermost head element is less than 1%. Failure at 15% strain occurs less than 1 msec later. Since the support structure would be expected to transmit little of the pressure loadings in the region above it to the lower plenum, this failure was unexpected. Figure 4, which is a CALCOMP plot of the lower plenum pressure along the shell about midway between the bottom of the support structure and the shell-lower head juncture, indicates peak pressures of under 70 atmosphere and an average pressure at the time of failure of ~20 atmosphere.

Utilizing the same initial problem conditions and geometry, a REXCO-HEP calculation was performed with an artificial rotational viscosity of 0.1 included in all nodes outside the core. Surprisingly, this resulted in an indicated failure of the vessel about 2 msec earlier—at 9.93 msec into the transient. The mesh geometry just 0.4 msec prior to failure is shown in Figure 5. Comparing this with Figure 6 indicates the stabilizing influence of the rotational viscosities on the grid system, particularly in the high distortion regions adjacent to the core. Again the loading of the vessel in the lower plenum has been quite rapid as the strain of 0.4 msec prior to failure was only ~1%. The calculation was terminated due to rapid increase in the total system energy to 200% of the initial value. Figure 7 indicates that the lower plenum pressure, with the artificial viscosity term in the calculation, has less high frequency components but increases at a faster rate than without the artificial viscosity.

To determine whether the observed vessel failures might, in some manner, be related to the geometric representation of the curved head, it was decided to perform these calculations with a flat-plate representation of the lower head, identified here as Model 9. The problem was run to 1000 time steps (15.9 msec), restarted and successfully run to 2000 time steps (23.9 msec) without any structural failure occurring. The total system energy had increased by ~20% at that time, though at 1000 cycles it had changed only 7%. (Determination of the effects of artificial rotational viscosity on ability to calculate this transient for longer times using this "rectangular" head geometry could not be accomplished due to early termination of this work.) The geometric distortions of the mesh structure at 1000 cycles are shown in Figure 8. Vessel strains are shown in Figure 9. These are qualitatively the same as seen in Figure 3 for the curved vessel representation at 11.11 msec. Concurrently, Figure 10 indicates the two models give virtually identical grid distortions above the lower plenum region, at times less than the time to fail the vessel.

Distribution of accident energy is shown in Figure 11 for this flat plate vessel model. Compared to results reported⁽¹⁾ for the somewhat simpler model of REXCO-H, there is more upward and radial kinetic energy for this model, and less downward (negative) kinetic energy—a probable consequence of a stiffer core support structure; the strains of the core barrel and vessel are also slightly higher. Figure 12 indicates little difference in energy distribution between the curved and flat vessel models. Figure 13, however, shows the energy at a given time to be noticeably less for the model which includes artificial rotational viscosity. This is also evident in the core (expansion work being about 10% lower for this case than for calculations without this viscosity term) as evidenced by the curves on Figure 14.

The REXCO calculation also provides useful information on support structure loading and pressures induced in upper and lower plenum regions in the vicinity of the reactor outlet and inlet nozzles. This information, reproduced from CALCOMP plots obtained from Model 9, is presented in Figures 15, 16, and 17.

2.4.4.4 Conclusions

1. The stiffness of the support structure has been shown to have appreciable effect on distribution of core accident energy. Additional calculation using REXCO with a reference Demonstration Plant geometry should be made to bracket a range of design value loads and strains to be expected on vessel structural components.

2. Modeling variables such as amount of artificial rotational viscosity in nodes, geometric modeling, and material properties treatment require further assessment for their effects on the calculated structural response to accident loads and the accuracy of the calculated results.
3. Improvements in the present REXCO program which would improve the accuracy and/or versatility of the program include:
 - a. Provision for "slip" between fluid nodes and elastic-plastic nodes, or between adjacent fluid nodes as specified in the input.
 - b. Improved and up-to-date documentation of code input requirements with particular attention called to specific restrictions on input.
 - c. Improvements in programming to provide more consistent input requirements (e.g., if an option is not utilized, sometimes the input card must be included and sometimes it must be omitted).

2.4.5 Fuel Rod Reliability

Monte Carlo techniques have been applied within the LMFBR industry to determine steady-state fuel rod reliability. Other less costly reliability techniques are being developed but without uniform guidance or integration into a national program. The high cost of Monte Carlo techniques preclude their direct application to determine fuel rod reliability during severe transients. However, the results of an example which illustrates the implication of statistical transient analysis are presented here. The example uses an existing computer model of a portion of an LMFBR fuel assembly to examine the effects of intrabundle thermal hydraulics and uncertainties in fuel damage limits on the expected number of fuel rod failures during a severe loss-of-cooling transient.

2.4.5.1 Steady-State Fuel Rod Reliability Analysis

Performance predictions for fuel in LMFBR cores involve use of complex models requiring input data having significant uncertainties. Monte Carlo statistical techniques are available to the industry in standardized texts and have been applied to the determination of steady-state fuel rod reliability^(4, 5, 6) of LMFBR cores. Other steady-state fuel rod reliability techniques⁽⁷⁾ which avoid the large number of computer calculations of Monte Carlo methods are being developed within the industry by individual companies. An integrated national effort to develop standard compilations of materials property data in a form suitable for these analyses and standardization of methods for performing these analyses would assure optimum development in this area.

2.4.5.2 Transient Fuel Rod Reliability Analysis

An evaluation of the applicability of statistical techniques to determine fuel cladding integrity during low probability loss-of-flow or overpower transients (events in the extremely unlikely category) has revealed a scarcity of published data in contrast to a rather extensive amount of data on statistical analytical techniques for steady-state fuel cladding integrity and core performance. An unsophisticated first approach to the problem is possible by combining Monte Carlo methods with existing transient analysis codes such as FORE II, FREADM, or SAS-2A. The estimated minimum cost for statistically analyzing a reactor transient with FORE II is $\approx$ \$10 per run with 5000 required Monte Carlo analyses, or \$50,000 per transient studied. This cost could be reduced through the introduction of biased sampling techniques if only a tail of the resulting probability distribution (cladding temperature, cladding strain, or cladding stress) need be determined. It is evident that the large computer costs involved would eliminate the use of the technique for parametric studies. An acceptable statistical transient analysis program awaits the development of techniques which will reduce computation costs to a reasonable level.

CYCLE = 700 TIME = 0.0111100

Figure 3. Model 10 Mesh Geometry at 11.11 msec

Figure 4. Model 10—Lower Plenum Pressure (Node 11, 9)

CYCLE = 600 TIME = 0.0095100

Figure 5. Model 10A Mesh Geometry at 9.51 msec

CYCLE = 600 TIME = 0.0095100

Figure 6. Model 10 Mesh Geometry at 9.51 msec

Figure 7. Model 10A—Lower Plenum Pressure (Node 11, 9)

CYCLE = 1000 TIME = 0.0159100

Figure 8. Model 9 Mesh Geometry at 15.91 msec

Figure 9. Model 9—Circumferential Strain of Lateral Core Support and Reactor Vessel Cylinder

CYCLE 700 TIME = 0.0111100

Figure 10. Model 9 Mesh Geometry at 11.11 msec

Figure 11. Model 9—Accident Energy Distribution

Figure 12. Model 10—Accident Energy Distribution

Figure 13. Model 10A—Accident Energy Distribution

Figure 14. Model 10—Core Work vs Time

Figure 15. Model 9—Net Downward Force on Core Support Structure

Figure 16. Model 9—Pressure at Outlet Nozzle (Node 11, 34)

Figure 17. Model 9—Pressure at Inlet Nozzle (Node 11, 9)

2.4.5.3 Examples of Simplified Probabilistic Transient Analysis

Model Description

A simplified approach was utilized to compare deterministic transient calculations with statistical transient analysis. The transient chosen to examine the implications of probabilistic transient calculations and probabilistic failure limits was a severe loss-of-flow transient initiated by a large pipe break at the reactor inlet nozzle for a typical Demonstration Plant reactor. This transient has been previously evaluated by deterministic methods for the GE Demonstration Plant, hot-leg pump reference design by use of the FREADM⁽⁶⁾ computer code. The results of this previous analysis were combined with a probabilistic failure distribution and the results of an intrabundle thermal-hydraulics model to determine the expected number of fuel rod failures.

A cross-section of the intrabundle model is pictured in Figure 18. The fuel rods and channel wall were divided into ten axial layers. Each layer was divided into eight coolant subchannels and ten channel wall nodes. Each of the three outer fuel rods was divided into four sectors with each sector having four fuel nodes and two cladding nodes while the inner fuel rod had only two sectors. This geometrical model was used in conjunction with the THTE computer code to determine the effect of overcooling of the edge fuel rods and the channel wall heat capacity in lowering sodium and cladding temperatures and reducing the expected number of failed fuel rods. The fuel assembly consisted of 0.250-inch diameter fuel rods with a pitch to diameter ratio of 1.28 and a channel wall thickness of 0.100 inch. The flow and power traces resulting from an inlet nozzle break and used in the THTE model are shown in Figure 19. The fuel rod power was that of the center core fuel assembly (peak power of 14.85 kW/ft) and was assumed to be uniform across the assembly.

Determination of Cladding Failure Temperature Density Function

For this severe loss-of-flow transient, cladding failure was assumed to occur by overpressurization resulting from the cladding temperature rise which increased the plenum temperature and gas pressure as the cladding strength was decreased. The cladding failure limits were determined from the intersection of the cladding ultimate tensile strength versus temperature curve with the cladding stress versus plenum temperature curve as shown in Figure 20. The cladding ultimate tensile strength for Type-316 stainless steel, 20% cold worked is given in Reference 9. The ultimate tensile strength shown is post-irradiation, but the 95% confidence levels shown are from the pre-irradiation data since confidence levels were not available for the post-irradiation data. This confidence interval is optimistic because the scatter in irradiated data is typically greater than unirradiated data. For this example the plenum pressure at the start of the transient is assumed to be 1000 psia and the cladding thickness is assumed to be 0.010 inch. Figure 20 shows that the median failure temperature is 1640°F with a 95% confidence range extending from 1620 to 1660°F employing a standard deviation of 10.2°F.

A more sophisticated approach to the determination of cladding failure limits for the purpose of a core safety analysis would consider other factors such as the uncertainty in cladding stress from uncertainty in plenum pressure. The cladding stress curve, as shown in Figure 20, should also be represented by a median value and a probability density function. The uncertainty in cladding stress results from uncertainties in gas release fraction from the fuel, neutron flux, fuel-cladding attack, gas atoms produced per fission, sodium corrosion, steel swelling and fuel rod dimensions.

Cladding Temperature Distribution Function

The calculated cladding temperatures also have an uncertainty. It is expected that the peak temperatures calculated during the transient will have a probability distribution at least as broad as the initial steady-state distribution. The estimated standard deviation in the steady-state temperature distribution was 40°F assuming a normal distribution. The cladding temperature probability distribution is assumed (for this example calculation) to broaden to a standard deviation of 50°F as a result of additional uncertainties introduced by the transient calculations.

Figure 18. Cross-section of THTE Model

Figure 19. Core Flow and Power

Figure 20, Cladding Failure Temperature

Expected Number of Fuel Failures

The probability of failing a fuel rod during the loss-of-flow transient is given by the area overlap of the cladding temperature distribution and the cladding failure temperature distribution as illustrated in Figure 21. The failure probability is given as (Reference 7).

$$P = \int dP = \int_{-\infty}^{\infty} f(T_c) \int_{T_c}^{\infty} F(T_f) dT_f dT_c$$

where $f(T_c)$ is the probability density function for the calculated cladding temperature and $f(T_f)$ is the probability distribution function for the cladding failure temperature. If both distributions are normal, the failure probability can be determined from normal probability distribution tables.⁽¹⁰⁾ If the cladding temperature is assumed to be deterministic rather than probabilistic, then the failure probability is determined by the location of the clad temperature on the failure temperature distribution using standard normal probability techniques.

A failure probability was calculated for each fuel rod in the THTE model for the hot bundle of the core. The failure probability was weighted by the number of fuel rods in the row to obtain the expected number of failures in the row. This was added for each row to give the expected number of failures in the hot channel. An expected number of failures was also calculated assuming the channel wall had no effect on neighboring cladding temperatures and that all cladding temperatures in the channel were equal to the cladding temperature of the interior rods. The calculation was also repeated assuming deterministic cladding temperatures and probabilistic failure temperatures.

Figure 21. Schematic Showing Probability Distribution Interference

Results of Example

The results of this example calculation are presented in Table 1. The results show: 1) that for this severe loss-of-flow transient, consideration of the intrabundle effects can significantly reduce the expected number of failures (by approximately a factor of two) from the expected number of failures determined when this effect is ignored; 2) by using deterministic techniques, the expected number of failures is zero since the peak cladding temperature lies about 180°F below the failure temperature; and 3) the results of this example also indicate that this apparent 180°F margin might not be sufficient to prevent a significant overlap of the cladding temperature distribution and the cladding failure temperature distribution as shown schematically in Figure 21. The actual numbers presented are for the purpose of demonstrating the statistical techniques. Although realistic estimates were made for many assumptions required, the results should not be considered a firm estimate of the expected number of failed rods in the hot bundle for this event.

This example has served to illuminate areas where further work is needed for the determination of transient fuel rod reliability. These are itemized below.

1. *Uncertainties in Steady-State Temperatures.* Since the transient is initiated from steady state, these uncertainties carry through the transient.
2. *Techniques for Statistical Transient Fuel Rod Reliability Analysis.* An acceptable statistical transient analysis program awaits the development of techniques which will reduce computation costs to a reasonable level.
3. *Fuel Failure Limits.* Failure limits depend on material property data which is just beginning to come from test programs and it also depends on fuel performance modeling. Both still have many uncertainties.

Coordination of this work on national level is needed to eliminate the discrepancy that now exists through the acceptance of statistical techniques for analyzing steady-state fuel rod performance, but the use of deterministic "worst case" or "hot spot" techniques for analyzing transient fuel rod performance.

Table 1
EXPECTED FUEL ROD FAILURES

Row	Number of Rods	Failure Temperature (°F) $\sigma T = 10.2^\circ\text{F}$	Cladding Midwall Temperature at Core Exit (°F) $\sigma T_c = 50.0^\circ\text{F}$	Probability of Failing One Rod	Expected Number of Failures
1	48	1640	1427	9.36E-5	4.49E-3
2	42	1640	1457	7.90E-4	3.32E-2
3	36	1640	1461	1.02E-3	3.67E-2
4	30	1640	1461	1.02E-3	3.06E-2
Interior	61	1640	1461	1.02E-3	3.06E-2
Expected Number of Failures					$\Sigma = 0.131$
Expected Number Assuming all Rods at 1461°F					$= 0.221$

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APPENDIX A**ABBREVIATIONS AND ACRONYMS**

AEC-RDT	Atomic Energy Commission—Reactor Development and Technology
ANL/RAS	Argonne National Laboratory/Reactor Analysis and Safety Division
DEFORM	Fuel Element Deformation Model (Subroutine in SAS-2A)
FCI	Fuel-coolant interaction
FEFPL	Fuel Element Failure Propagation Loop
FFTF	Fast Flux Test Facility
FGR	Fission gas release
FREADM	Fast Reactor Excursion Accident Dynamics Model, a computer code
FTR	Fast Test Reactor
GE/ANL	General Electric/Argonne National Laboratory
GE-BRD	General Electric—Breeder Reactor Department
GETHERM	A multigroup, one-dimensional integral transport code for computation of thermal neutron fluxes
GFY	Government fiscal year
HCD	Hypothetical core disassembly
HCDA	Hypothetical core disassembly accident
HEDL	Hanford Engineering Development Laboratory
LMFBR	Liquid Metal Fast Breeder Reactor
LOC	Loss of coolant
LOD	Line of defense
LOF	Loss of flow
MARS	Computer code for core disassembly
MFCI	Molten Fuel-Coolant Interaction, a computer code
PAHR	Post-Accident Heat Removal
PRETAB	Table of normalized driving pressure of incoming coolant
PTS	Plant Technical Specifications
QA	Quality assurance
REXCO-H	Two-dimensional hydrodynamics code for primary containment
REXCO-HEP	Two-dimensional hydrodynamics elastic-plastic code for primary containment
REZONE	Reformulates Lagrangian mesh for use with REXCO
SAD	Safety assurance diagram
SAP	Safety assurance program
SAS-2A	A pre-disassembly computer code
SN1D	A multigroup, one-dimensional transport theory/diffusion theory code for the computation of neutron fluxes and reaction rates
THTE	Transient Heat Transfer Code, Version E
TOFF	Transient overpower fuel failure
TPD	Technical Program Director
TREAT	Transient Reactor Test Facility
UC-LBL	University of California—Lawrence Berkeley Laboratory
VENUS	Computer code for core disassembly
WARD	Westinghouse Advanced Reactor Division

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