January 15, 1980

Mr. Sidney G. Harbison, Director
Nuclear and Magnetic Fusion Division
Department of Energy
San Francisco Operations Office
1333 Broadway
Oakland, CA 94612

Dear Mr. Harbison:

SUBJECT: Quarterly Technical Progress Report, October - December 1979
DE-AT03-76SF71032

By copy of this letter we are forwarding an advance copy of the subject report to Dr. H. K. Fauske of the FRS-TMC. A copy is also being sent to the GE/California Patent Group for review and written patent release no later than January 29, 1980, in order to meet the FRS-TMC publication deadline.

Very truly yours,

T. R. Sandke, Manager
Safety Projects

ATTACHMENTS

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A study has been initiated to provide a basis for understanding some of the confusion surrounding Common Cause Failure. Hopefully this may lead to a more accurate definition and understanding of CCF phenomena. The work to date will be reported in a paper titled, "Common Cause Failures - A Dilemma in Perspective," A. M. Smith and I. A. Watson, to be presented at the 1980 Reliability Maintainability Symposium, January 22-24, 1980 in San Francisco, California. (LOA 1.0b, GE).

A topical report titled, "Reliability Evaluations of Candidate Shutdown Heat Removal Systems" was completed and presents the results of reliability studies of various Shutdown Heat Removal Configurations (LOA 1.3, GE).

A topical report titled "Identification of Reliability Test Needs for LBR-SHRS Subsystems," evaluates five of the seven-key subsystems of a typical LMFBR SHRS as one of the first steps in establishing an SHRS Integrated-Reliability Test Program (LOA 1.3.3, GE).

A two-dimensional diverter model and a modified center body for the existing axis symmetric diverter were completed. The latter will be tested in the SASS-FC water loop (LOA 2.1.1.2, GE).

Five inherent release tests were completed on SASS-ACA (LOA 2.1.1.4, GE).

Investigation of DCT-1 design parameters lead to the conclusion that design criteria for meeting the primary objectives (comparing neutron hodoscope and Pinex camera capabilities) can be met (LOA 2.1.2, GE).

Fuel rod mechanics and cladding failure analyses were performed and indicated that a transient ramp rate between 5%/sec and 10%/sec is slow enough to typify all 'slow' overpower transients to be conducted in the PFR-TREAT test program (LOA 2.1.2, GE).

The limited distribution report "SLSF W-1 LOPI Experiment - Preliminary Evaluation of Data"(P2777 ST-TN-80015) was issued. This report contains a discussion of the results of the LOPI Phase of testing, comparisons of data, and analysis and conclusions based on preliminary data (LOA 2.2.2, GE).

An expression was derived for determining the concrete porosity at any temperature (\( \geq 100^\circ C \)) as a function of temperature and initial porosity (LOA 3.2.2, GE).

A generic risk model for Breeder Risk Allocation was tested and modified. The model was qualified using GE assigned Large Breeder Reactor interim valves for the allocated variables. The base configuration model was subjected to sensitivity analysis to establish the relative effects on plant risk of changes in dominant accident sequences. Also, the most risk-sensitive functions were identified (LOA 6.1.1, GE).
LOA-1: PREVENT ACCIDENTS

1.0 LOA-1 INTEGRATION

1.0b Common Cause Failure -- Nature and Impact (GE: G. L. Crellin, I. M. Jacobs)

Three new studies of Common Cause (Mode) Failure have been published recently by authors outside ARSD. They are:


Hagan (Ref. 1) traces the history of the industry's awareness to the problem of Common Cause Failure and has compiled a comprehensive bibliography.

Edwards and Watson (Ref. 2) classify the causes of Common Mode Failure into the four broad categories of design, construction, procedural, and environmental, and discussed the defenses that can be mustered against the various root causes.

Rasmuson et al (Ref. 3) concentrates on Common Cause Failure analysis techniques published by others and provides a review and critique of the available methodology.

The recurring theme, running through these recent references as well as many earlier ones, is that Common Cause Failures have considerable potential impact on nuclear plant safety and public risk, but that the assessment of that impact is, at best, extremely difficult. In part, the difficulty may be characterized as a general lack of focus on what is meant by the term "Common Cause Failure".

Cueing on that observed lack of definition, ARSD-Reliability Engineering concentrated on evaluating several published definitions of Common Cause/Mode Failure. First of all, we determined the attributes required of a Common Cause Failure according to each definition. For example, most definitions require that there be multiple failures and that the failures be due to a common causal event. There is no consistency among the definitions in a requirement that the failed components be identical, that they be redundant, that the multiple failures must occur within a critical
time period, or that the multiple failures must lead to system failure. Most definitions do not distinguish between multiple failures as a result of a cascading effect and those that fail directly as a result of a common cause.

To judge the importance of a definition on an understanding of the common cause failures in the literature, twenty events originally reported as Common Cause Failures were examined and rejudged according to each of ten different author's definitions. In 12 of the 20 cases the definitions were unanimous; in nine cases all the definitions agreed that the reported event was a bona fide Common Cause Failure, in three cases all agreed it was not. The remaining eight resulted in split decisions with no particular pattern, and therein lies the problem.

If split decisions in 40 percent of the cases are typical, there is room for considerable confusion in the proper identification of Common Cause Failures. Of even more importance vague definitions and identifications may be coupled with totally inappropriate corrective actions.

This study has provided a basis for understanding some of the confusion surrounding Common Cause Failure and hopefully may lead to more accurate definitions. The work will be reported further in a paper entitled "Common Cause Failures--A dilemma in perspective," A. M. Smith and I. A. Watson, to be presented at the 1980 Reliability Maintainability Symposium, Jan. 22 - 24, 1980, San Francisco, California.

1.0c Development of Methods for Reliability Assessment
(GE: G. L. Crellin, G. E. Ingram)

The objective of this task is to develop reliability assessment methods that have the capability of including in the analysis such factors as dependencies among elements of the model, time phased missions, maintenance and repair, and the man/machine interface. Since the inclusion of these factors in the analysis increases the complexity of the model significantly, thus increasing computation times significantly, an attempt is being made to develop a combination of techniques that will improve the analysis and computational efficiency. This will allow the engineering analyst to include all the detail necessary to accurately depict the reliability model and still maintain a viable computational tool when it is required to exercise the model many times.

Safety related systems of nuclear power plants generally contain redundant components and subsystems such that only multiple failures of the redundant elements can affect the power plant safety. These systems generally include a capability for the restoration of failed elements either on a periodic basis, as demanded by inspection testing at specified intervals, or on a randomly demanded basis as announced by failure of the item. Some of these safety related systems are designed to perform their specified function during the normal plant operation as well as during the plant emergency operation (e.g., shutdown heat removal system). At the beginning of this emergency operation all of the redundant elements which developed during the normal plant operation or even one or more redundant subsystem may be non-operational or in repair. These types of initial conditions have to be taken into account for an accurate reliability assessment of safety systems. A Markov model will be used to compute the probabilities of different initial conditions and to predict the safety system reliability more accurately.

This approach, however, requires the solution of a set of linked linear differential equations, the number of which equals the number of states available to the system. For complex power plant systems, the number of states in a Markovian
model is very large which requires extensive computer time for numerical computation. The computational effort involved in a quantitative reliability evaluation can be substantially reduced if the Markov process describing the system behavior is mergeable. A technique for reducing the dimension of the transition probability matrix, with merging states, is being investigated.

A further effort during this quarter has been directed toward investigating the possibilities of combining a Markov analysis and a failure state modelling approach to reliability for improving the computational efficiency while maintaining flexibility for the analyst to develop an accurate reliability model that includes the various factors. ARSD has developed two efficient computer programs for performing these types of analyses: A program named "MARK" was developed for performing Markov analyses and one called "PROBCALC" for performing probability calculations associated with a failure state modelling approach. A technique for appropriately combining these programs to improve the accuracy and efficiency of the computations is currently being investigated.

1.0e Update Safe Shutdown Reliability Program Plan (GE: G. L. Crellin, S. F. Armour)

Discussion at ANL, Chicago on November 29, 1979 with FRS-TMC, GE-ARSD and W personnel resulted in determination of the outline and content for the report "Update Safe Shutdown Reliability Program Plan."

GE-ARSD provided initial input to W during December 1979 which addressed the Tasks and Activities to be discussed in Section 1.3, "Shutdown Heat Removal Reliability", of this report.

1.2 REACTOR SHUTDOWN SYSTEM RELIABILITY (GE: G. L. Crellin, H. Cohen)

The principal activity under this topic was directed at developing the SCRS Master Test Plan and evaluation of the CRBRP P-1 test results to support this task.

SCRS Reliability Testing Program. Development of the preliminary Integrated Test Plan (ITP) continued during the period covered by this report. The test plan outline, which had previously been conceived, was expanded to a greater level of detail and further refined. Several sections of the detailed outline were converted to narrative form, as actual sections to be included in the preliminary ITP during this reporting period. In order to assure an efficient, well-planned program with meaningful data output for reliability evaluation, the concepts of design of experiments are being applied in determining what tests are to be performed, the number of tests to be performed, the parameters to be tested, and the sequence of varying the tested parameters.
The Test Plan and Operating Procedures for Latch Scram Test #3 was completed and was distributed for internal review and approval. Final issue of the document is expected by January 8, 1980, and testing is expected to commence by February 1, 1980. In preparation for this event, the latch scram test rig upper and lower internals have been installed in STL 1. In addition, a Project Directive was issued which initiated planning for a Test Readiness Review which is to review and determine the status of the documentation, test facilities, test article, instrumentation, data acquisition system, etc., preparatory to beginning the Latch Scram #3 test.

Test procedures for Bellows Test #3 have been developed, and final issue is expected by January 18, 1980. In order to support a test date of February 15, 1980, the bellows test rig is in the process of being refurbished with a scheduled completion date of January 8, 1980.

1.3 SHUTDOWN HEAT REMOVAL SYSTEM RELIABILITY (GE: C. R. Herrmann, J. G. Elerath)

During this quarter, the Topical Report "Reliability Evaluations of Candidate Shutdown Heat Removal Systems" was completed and satisfied Milestone 1.3.1. This report presents the results of reliability studies of various Shutdown Heat Removal System configurations, thereby supporting the development program for Large Breeder Reactors. The two phases of analyses presented both employ similar modeling techniques but have significantly different purposes. Summaries of each phase are provided in Sections 1.3.1 and 1.3.2. Activity addressing the current efforts to develop the SHRS Integrated Test Plan is discussed in Section 1.3.3.

1.3.1 Main Heat Transport System (GE: C. R. Herrmann, J. G. Elerath)

The objective of the first phase of study was to determine the amount of additional reliability which results when the dedicated Auxiliary Cooling Systems (ACSS) are coupled with the capability to remove heat through the steam generator to the condenser. This provides, among other things, some insight as to the importance of reliable steam generators.

The results show that in all cases, use of the condenser increases the SHRS reliability. However, the amount of increase is highly dependent upon the particular Auxiliary Cooling System configuration that is selected and the extent of the dependency among the various heat sinks. This is supported by the fact that the amount of decrease in unreliability due to the addition of the condenser is not a constant value. It varies from a factor of 1.2 to a factor of 28. This variability is primarily due to the different amounts of hardware which are shared between the dedicated heat removal system and the condenser's systems for each SHRS configuration.

Generally, the amount of decrease in unreliability resulting from the addition of the condenser into the analysis diminishes as the dependency between the condenser and the dedicated heat removal system increase. For example, a smaller decrease in unreliability is noted when the condenser is added to a Steam Generator Auxiliary System (SGACS) than when it is added to a Primary Reactor Auxiliary Cooling System (PRACS). This is because the SGACS and condenser systems have the entire main heat transport system operation as a commonality (dependency) whereas the PRACS and condenser systems have only the primary heat transport system in common. It should
be noted that attempts to determine the degree of dependency defy intuition and simplistic models. Detailed models showing dependencies (repeated events) and computer programs such as PROBCALC are required to determine this.

It is interesting to note that as the redundancy of the PRACS increases from three-out-of-four to two-out-of-four the relative influence of the condenser increases as well. These results support the conclusion that credit for the main condenser does increase the reliability of the overall SHRS mission and that systems, such as 1/4 DRACS or a 1/4 PRACS can meet specified reliability goals. The major advantages of plant designs which employ only one auxiliary SHRS configuration are simplicity and reduced cost, since the condenser is already a necessary part of the plant. Currently, conservative design considerations suggest two independent and diverse auxiliary systems, each totally dedicated to the shutdown heat removal task. The result of this study suggest that such conservatism may be overly complex and expensive, and that serious consideration and analysis should be given to a single auxiliary system coupled with capabilities of the main condenser.

1.3.2 Auxiliary Heat Transport Systems (GE: C. R. Herrmann, J. G. Elerath)

The second phase of analysis was far more detailed and was performed in support of the Conceptual Design Study (CDS) Task 7.3 SHRS Working Group. The purpose of the working group was to study and make a recommendation on the types of Auxiliary Cooling Systems that should comprise the dedicated portion of the CDS Shutdown Heat Removal System. Five of these dedicated heat removal system configurations were analyzed for their absolute and relative reliability rankings. From these five cases, which are based on a loop type plant, relative reliabilities for the pool plant and five other loop plant configurations were inferred. The results of the analyses indicate that a two-out-of-three DRACS with a three-out-of-four PRACS is the most reliable configuration for both the loop and pool type reactors, but all configurations analyzed meet the minimum reliability goal of $1 \times 10^{-6}$ failures per year.

Two sensitivity studies highlight potential risks associated with configurations involving the PRACS concept. The first shows that the reliability of the dedicated shutdown heat removal system which is comprised of a PRACS used in conjunction with either an Intermediate Reactor Auxiliary Cooling System (IRACS) or a SGACS is sensitive to the number of short term outages (less than 24 hours in duration). Furthermore, these configurations are very sensitive uncertainties in the rates of Primary Heat Transport System Failure, as is illustrated by the second sensitivity study. This implies a high risk of not meeting the reliability goal unless the actual failure rates can be shown to be the same as or lower than the estimates used. This sensitivity is attributed to the dependence which exists between a PRACS and IRACS (or SGACS) heat transfer path.

Combining these two sensitivity studies illustrates a very important and fundamental concept of reliability, that of independence. Use of two independent systems can provide a much higher reliability than two partially dependent system, even though the dependent systems have greater redundancy in the later mission time-phases. It is concluded, therefore, that independence of ACSs can not only provide a SHRS which can more easily meet the goal (or provide reliability margins to account for various uncertainties), but also one which is less sensitive to uncertainties in the estimated failure rates. It is recommended that the ACSs be as independent of each other as technologically possible when selecting a SHRS configuration.

A concern which arises in the DRACS concept is the necessity to control sodium flow within the reactor vessel by use of check valves. An additional study
examines the sensitivity of the probability of SHRS failure for a DRACS + PRACS configuration to the failure rates of the three check valves which isolate the hot and cold plena in the loop type reactor. All failure rates were increased simultaneously by a factor of 10, then by a factor of 100. The probability of SHRS failure is relatively insensitive to the check valve failure rates until they increase by more than a factor of 10. A one-hundred fold increase yields a probability of SHRS failure slightly in excess of the goal. Extrapolation in the other direction leads to the conclusion that reduction of the failure rates, below the base values, will not improve the reliability noticeable.

It should be remembered that this analysis assumes that failure of two of the three check valves will prevent only the DRACS from operating adequately. At this time there are no identified failure modes which will prevent both the DRACS and the PRACS from functioning properly. In the future, additional reliability efforts should be devoted to the study of these check valves to determine how such valves could fail and how such failures affect the functional operation of both DRACS and PRACS.

1.3.3 SHRS Integrated Test Plan (GE: C. R. Herrmann, S. B. Tulloch)

Two meetings have been held with GE-ARSD and GE-CR&D personnel to develop an outline for a SHRS Integrated Test Plan (ITP). The tentative outline of the ITP has been prepared and rationale for selection of SHRS tests are in the process of formulation. The substantiation of Bayesian techniques for establishing reliability test sample sizes is in progress. Initial critical equipment test needs lists have been received from the subcontractor Energy Incorporated (EI) in the form of Topical Report "Identification of Reliability Test Needs for LBR - SHRS Subsystems", GEFR - 00493, issued by GE-ARSD in September 1979.

This report is a study which evaluates five of the seven key subsystem of a typical LMFBR SHRS as one of the first steps in establishing an Integrated Reliability Test Program. The report identifies reliability information necessary for licencing and design. Also, that information currently available from DOE programs or reactor operation is evaluated and existing data needs are identified to support attainment of LOA-1 criteria.
2.1 REACTOR SHUTDOWN SYSTEM FAULT ACCOMMODATION

2.1.1.2 Design (GE: E. R. McKeehan, W. A. Brummond)

A two dimensional model of the axissymetric fluidic diverter (Figure 2-1) was designed and built to illustrate the bi-stable switching phenomenon. The jet formation was so vivid that it was decided to vary the geometry (spacing) to obtain a visualization of switching under extreme dimensional changes. Recommendations from this experiment are:

- Continue use of two-dimensional models to gain understanding of the fluidic principles involved.
- Use two-dimensional models to explore more efficient (space allocation) designs.
- Use two-dimensional models to determine the physical limits for jet attachment.

Fig. 2-1 Two Dimensional Model
A new center body (Figure 2-2) has been completed for use in the existing axissymmetric diverter. It contains pilot valve which will permit the diverter to switch on an inverted signal (a pressure less than ambient). Tests will be conducted in the SASS water loop to determine the response time, sensitivity and reliability.

![Diagram of new center body](image)

Fig. 2-2 New Center Body

A conceptual arrangement drawing of the fluid controlled SASS for ADS application was completed. This drawing shows the integration of the fluidic diverter, sensors, guide tube assembly and driveline.

A one-third scale air demonstration model has been designed and built. This model is intended to provide a visualization of the functional components of the fluid controlled SASS concept (diverter, guide tube/absorber assembly, and sensors). Shake-down testing has begun.

2.1.1.4 ETEC Testing of SASS-ACA (GE: E. R. McKeenan, W. A. Brummond)

Five inherent release tests have been accomplished to date. An unusual occurrence in the SASS-ACA test loop at ETEC (October 17) resulted in sodium and/or sodium vapor contamination of the upper housing. During the clean-up operations a foreign particle was removed from the handling socket. It is postulated this is the particle that accounted for the electromagnet to handling socket misalignment, which resulted in reduced holding power. The loop was reassembled and refilled on December 4th. On December 12th an Unusual Occurrence Review was held at ETEC with AI, GE and W-ARD attending.
The initiating cause of the October 17th occurrence is still undefined; however, several possibilities were discussed. It was agreed that instrumentation should be installed to alarm the loop operator in the event of a reoccurrence. This should minimize damage to the upper housing and provide information on cause if the event occurs again.

Two lift withdrawal loads and drop current tests were done after the loop reassembly. Their data indicates a misalignment of the driveline. A procedure for readjusting the alignment was prepared by ETEC.

2.1.2 Inherent Shutdown by Fuel and Absorber Motion

Diagnostic Capsule Test No. 1 (GE: P. M. Tschamper, K. E. Gregoire, E. Kujawski)

The primary objective of this DCT-1 test is to compare the spatial resolution and abilities of the neutron hodoscope and PINEX camera under a set of conditions simulating in the present TREAT core a MK-III loop experiment, operating in the TREAT Upgrade core. These conditions include:

(i) calibration or power coupling factors which range from 0.8 to 1.8 and 2.0 to 4.0 W/g/MW TREAT, respectively (the axial power profile in the TREAT core provides a range of ~ 2 in specific heat generation),

(ii) a "reasonable" shielding effect.

The fuel will be held stationary in this test but a variety of pellet geometries will be represented in the fuel column.

Secondary objectives of the DCT-1 experiment are to measure the systematic errors for each diagnostic system due to self-shielding and axial power profile of the core and to detect and measure random errors.

Low energy pulses will be employed to energize the fuel and provide the signals which will comprise the test. The low energies will permit dry pin operation. Both single-pin and seven-pin configurations will be included in the test. Filter thickness and fuel enrichment will be varied to give the desired range of power coupling factors from ~ 0.8 to ~ 4. The test vehicle will be a static TREAT capsule which has different cross-sectional dimensions and is constructed of different materials than the MK-III loop. However, calculations performed for GE at Science Applications Inc., at ANL-W and General Electric indicated that the dimensions of the DCT-1 inner capsule can be chosen to provide adequate neutronic simulation of a MK-III loop experiment.

The use of enriched UO₂ fuel rather than mixed oxide fuel in all the test fuel for DCT-1 has been investigated at GE. Calculations have been carried out for a specific single pin configuration of the DCT-1 capsule. The design variations consisted of the fuel enrichment and thickness of the dysprosium flux filter. The axial midplane power couplings and radial power profiles were calculated using one-dimensional S₄ - transport theory with 49 - group P₆ - transport corrected cross-sections (MIC49 library). The results for the "reference" pin (solid, 0.195" diameter, 95% TD) are reported in Table 2-1.
Table 2-1 Power Couplings and Ratios of Peak to Minimum Fission Rates in UO₂ at the axial Midplane of the DCT-1 Experiment

<table>
<thead>
<tr>
<th>Enrichment, %U-235 in U</th>
<th>Filter thickness, mils</th>
<th>Power coupling W/g/MW · TREAT</th>
<th>Peak/min Fission Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>23.0</td>
<td>8.0</td>
<td>3.06</td>
<td>1.18</td>
</tr>
<tr>
<td>23.0</td>
<td>30.0</td>
<td>1.47</td>
<td>1.07</td>
</tr>
<tr>
<td>23.0</td>
<td>0.0</td>
<td>7.78</td>
<td>1.36</td>
</tr>
<tr>
<td>15.0</td>
<td>30.0</td>
<td>1.06</td>
<td>1.06</td>
</tr>
</tbody>
</table>

From the above results, taking calculational uncertainties into account, it is concluded that the design criteria could be achieved with the specified DCT-1 capsule for dysprosium flux filters of 25 and 5 mils using a single enrichment (23%) of UO₂ fuel.

The target fuel pins will be 48 inches long and thus will simultaneously utilize the entire TREAT core height and test the fuel height capability of the 1.2 meter hodoscope. The test fuel pins will be axially segmented into regions with different fuel geometries and masses which will be unknown to the diagnosticians. For example, annular fuel pellets with different OD's and ID's will be employed. A reference fuel segment with all fabrication parameters known to the diagnosticians will be included in each test fuel pin. Peripheral fuel pins in the seven-pin tests will be essentially uniform over their entire lengths.

Instrumentation for the test will be minimal. Flux wires will be provided to obtain a measure of the integrated exposure of the fuel. No thermocouples will be needed. A total of 16 transient pulses are planned. This embodies a matrix of two power levels and two fuel configuration with four pulses for each of the combinations to give statistical basis for evaluating random errors.

**Design Option Development (GE: P. M. Tschamper, T. A. Shih)**

Extensive analyses have been performed with the cooperation of ANL to compare the safety trade-offs of three core designs including homogeneous, loosely coupled heterogeneous, and tightly couple heterogeneous configurations. These core designs have been analyzed with the SAS-3D code to determine core response during hypothetical unprotected loss-of-flow transient events. The major conclusions are: 1) The heterogeneous core, with its smaller sodium void worth in the driver fuel region, consistently experiences milder transients and is significantly less sensitive to phenomenological uncertainties than the homogeneous core. 2) The loosely and tightly coupled heterogeneous cores are very similar in their transient response. 3) When design changes are introduced to enhance voiding incoherency, the heterogeneous core transient response can become even more benign.

In addition to the safety trade-off studies, a preliminary set of safety design criteria was developed. The purpose of these criteria is to provide designers with guidelines for core designs which would avoid severe consequences.
during CDAs. The current criteria include a formula that restrains three key safety parameters: sodium void worth, Doppler coefficient and sodium voiding incoherency. This study is being documented and will be issued to CDS Task 8.0 Working Group for use in the second phase of the CDS program.

PFR-TREAT Test Program (GE: P.M. Tschamper and D.B. Atcheson)

Fuel rod mechanics and cladding failure analyses have been performed using the BEHAVE-SST program in order to select a transient power vs time history for the test program that is slow enough to be representative of all "slow" overpower transients. In this analysis a PFR-TREAT Program fuel rod of US design was assumed to be irradiated to an axial peak burnup of 33 MWD/kg, to have an axial peak power at the end of steady state irradiation of 417 W/cm, and to have an axial peak cladding midwall temperature of 1123°F (606°C) at the end of steady state irradiation. This fuel rod was assumed to experience five different overpower transients, with ramp rates ranging from 50¢/sec to 2¢/sec, (i.e., power periods from 3 seconds to 75 seconds).

For the particular fuel rod design and operating conditions assumed, the BEHAVE-SST analysis indicates that the loading mechanism causing failure does not change as ramp rate varies within the limits investigated. Cladding failure was predicted in all cases to occur above the axial midplane with an axial midplane fuel melt fraction of 30% to 40%. In addition the analysis indicates that fast-transient characteristics (such as radial temperature distributions and fuel creep) begin to appear when the transient power period is reduced to approximately 10 to 15 seconds. As transient ramp rate reaches and exceeds this range predicted axial locations of cladding failure start to move closer to the axial midplane and more heat is retained within the fuel, so that fuel melt fractions at the time of failure begin to increase. As ramp rate increases, cladding failure becomes more likely prior to any fuel melting. These early failures were not predicted to occur, however, in the range of ramp rates considered.

It was concluded from this analysis that a transient ramp rate between 5¢/sec and 10¢/sec (power period = 10 to 15 seconds) is slow enough to typify all slow transients for the specific fuel rod design and steady state operating conditions of this study. However, other fuel rod designs and operating conditions would be expected to enter the "slow-ramp" regime at different ramp rates. Fuel pellet geometry, burnup, steady state cladding temperature, and steady state power are expected to affect these results most.

Planning for the execution of the joint PFR-TREAT Test Program continued with the participating US organizations.

GE participated in an ad-hoc committee of the FRS-TMC charged with the review of the TREAT Upgrade functional requirements and the needs for FBR fuels safety testing.
2.2.2 Coolability of Core

**Literature Search and Code Acquisition (GE: P. M. Tschamper, R. McGriff)**

An effort is being undertaken to characterize and quantify the effects of a protected loss-of-heat-sink accident in an LMFBR, and to model the establishment of natural circulation following pump coast-down. Special emphasis is placed on both the determination of the nature of resulting sodium flow patterns within the vessel, and of the appropriate, time-varying temperatures within the sodium. Based on the results from this study requirements for direct heat removal equipment design and positioning within the vessel will be specified.

A literature search regarding recently documented tools capable of representing the low power and low flow heat transport within the core assembly was conducted. It revealed that some more refined modeling of core internal flow paths and heat transport under natural circulation would be beneficial for meeting the objectives of this study. Of particular interest is the development of tools that allow integration of the heat transport in the shut-down core assembly with the heat transport in and through the core outlet plenum to the auxiliary heat sink. A modular analysis model was constructed and the proper functioning of the computer program logic was tested against SAS-3D results of an unprotected loss-of-flow accident analysis. Low flow, low power applications of the model will be evaluated next.

**SLSF W-1 Data Analysis (GE: P. M. Tschamper, D. D. Knight)**

The limited distribution report "SLSF W-1 LOPI EXPERIMENT - Preliminary Evaluation of Data" (P2777 ST-TN-80015) was issued. This report contains a discussion of the results of the LOPI Phase of testing, comparisons of data and analysis and conclusions based on preliminary data. The purpose of the LOPI tests was to determine the fuel pin and sodium coolant thermal responses during a loss-of-piping-integrity simulation with different heat transfer characteristics due to changes in fuel structure with irradiation.

During fuel conditioning, the effects of fuel restructuring and burnup were observed as a reduction of the fuel temperature measured with in-fuel thermocouples in seven of the fuel pins. A post-test analysis of the fuel condition throughout approximately 400 hours at full power in the test program was performed with the LIFE-4 code which includes fuel restructuring, thermal property changes and fuel/cladding gap parameters. The results show fair agreement of predicted temperatures with in-fuel thermocouple data after restructuring occurs. Calculated temperatures were 50 to 200°F lower than measured near the fuel pin midplane and 100° to 200°F higher near the top of the fuel pin. Predicted temperatures at the midplane were about 600°F low for fresh fuel. Additional calculations will be made to fully utilize the steady state data and to determine the fuel and fuel/cladding gap thermal properties at the pre-test LOPI conditions.
These results suggest an axial power profile which is more peaked than the 1.24 peak to average shape used in pre-test analyses. The in-fuel thermocouple data were used in a simple calculation to infer a peak to average of 1.33, close to the value of 1.3 subsequently obtained from ETRC testing.

Axial sodium temperature profiles measured at steady state with wire-wrap thermocouples match calculated profiles and provide a check of the steady state power to flow ratio. Calculations with assumed variations of axial power distribution show little effect on axial sodium temperature profiles.

Sodium temperature was uniform in the 10 inches above the fuel. In addition, radial gradients in the sodium at steady state were less than 20°F indicating little edge effect due to radial heat loss.

During LOPI transients, at nominal condition, no boiling occurred. The maximum sodium temperature as measured with wire wrap thermocouples was about 40°F below the saturation temperature. Pre-test analyses predicted boiling for about 0.6 seconds near the top of the fuel bundle. The last LOPI test was run with the test section power about 5% higher than nominal. Boiling was achieved with no superheat and without fuel pin failure.

Post-test CORBA-3M analyses were made with fuel thermal conductivity and fuel/cladding gap conductance values obtained from the post-test LIFE-4 calculations. The results show only slight sensitivity of sodium temperature to details of the fuel pin thermal properties in the range considered. Post-test calculations indicate boiling as did the pre-test calculations.

In-fuel thermocouple response during the test was also compared with pre-test and post-test calculations of fuel and thermocouple temperature transients. Calculations of thermocouple temperature made with revised fuel conductivity and fuel/gap conductance based on steady state conditions agree very well with the in-fuel thermocouple data. The major effect introduced by analysis with revised thermal properties is seen in the steady state temperature level. Transient response is similar within the range of thermal properties used.
3.2 DEBRIS ACCOMMODATION

3.2.2 Accommodation Within Inner Cells

Heat and Mass Transfer in Concrete (GE: J. W. McDonald and K. H. Chen)

Mechanistic modeling of heat and mass transfer in concrete under high temperature and heat flux conditions requires adequate characterization of concrete thermophysical properties such as permeability. The concrete permeability has been found to be primarily dependent upon the concrete porosity (K. H. Chen, et al., "GE LOA-3 Tasks Annual Report, Sept. 1978 - Aug. 1979," GEFR-00481, Aug. 1979). Based on the assumption that concrete permeability is predominantly a function of concrete porosity and that flow within the porous concrete is laminar, the Blake-Kozeny equation (Bird, Stewart, Lightfoot, "Transport Phenomena," John Wiley & Sons, Inc., pp. 196-199, N.Y., 1960), in combination with the Darcy's Law, gives the relation between permeability (K) and porosity (ε) as follows.

\[ K = C' \frac{\varepsilon^3}{(1 - \varepsilon)^2} \]  

(Eq. 1)

where \( C' \) is a constant. Based on experimental findings, it has been noted that at temperature ≤ 100°C, ε is primarily the porosity created by the capillary pores of the concrete and is essentially constant (K. H. Chen, et al., "GE LOA-3 Tasks Annual Report, September 1978 - August 1979," GEFR-00481, Aug. 1979). Following the procedure outlined by Neville (A. M. Neville, "Properties of Concrete," John Wiley and Sons, Inc., pp. 24-29, N.Y., 1973), the capillary pore fraction with respect to the cement volume can be estimated for a given water to cement ratio. The initial concrete porosity, \( \varepsilon_0 \), can be determined from a known amount of aggregate. The data based on the report, J. D. McCormack, A. K. Postma, J. A. Shur, "Water Evolution from Heat Concrete," HEDL-TME 78-87, Feb. 1979, give the following values for K and \( \varepsilon_0 \).

<table>
<thead>
<tr>
<th>Aggregate</th>
<th>Water/Cement</th>
<th>( \varepsilon_0 )</th>
<th>K (m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Basalt (B-1)</td>
<td>0.621</td>
<td>0.0600</td>
<td>1.21 \times 10^{-15}</td>
</tr>
<tr>
<td>Magnetite</td>
<td>0.496</td>
<td>0.0338</td>
<td>6.04 \times 10^{-17}</td>
</tr>
<tr>
<td>Limestone</td>
<td>0.548</td>
<td>0.0466</td>
<td>9.29 \times 10^{-17}</td>
</tr>
<tr>
<td>Steel Shot</td>
<td>0.496</td>
<td>0.0480</td>
<td>2.14 \times 10^{-16}</td>
</tr>
</tbody>
</table>

Table 3-1 Initial Porosity and Permeability of Concrete with Different Aggregates.
The report also gives $C' = 1.735 \times 10^{-12}$ (m$^2$) for all aggregates. The experimental results also give the correlation

$$k = a \ e^{bT}$$  \hspace{1cm} (Eq. 2)

Thus,

$$C' \frac{\varepsilon^3}{(1 - \varepsilon)^2} = a \ e^{bT}$$  \hspace{1cm} (Eq. 3)

From the test data, $b$ is approximately the same for different concretes. It is therefore considered to be independent of $\varepsilon_0$; thus, the following relation is obtained.

$$\frac{\varepsilon^3}{(1 - \varepsilon)^2} = \frac{\varepsilon_0^3}{(1 - \varepsilon_0)^2} \exp \left[ 0.011228(T-100) \right]$$  \hspace{1cm} (Eq. 4)

where $T$ is the concrete temperature and is in $^\circ$C. This expression determines the concrete porosity at any temperature ($^\circ$100°C) as a function of temperature and initial porosity. In conjunction with (Eq. 1), these expressions show that the set of conservation equations used to mechanistically treat heat and mass transfer in concrete are coupled through $K$, $\varepsilon$ and $T$. These expressions could be incorporated into the existing mechanistic models.
6. R&D INTEGRATION

6.1 INTEGRATED ANALYSIS


Risk Analysis Methods Development - Subtask B - Risk Analysis Application. During the reporting period, the generic risk model for Breeder Risk Allocation was tested and modified. The risk model was quantified employing GE assigned Large Breeder Reactor (LBR) interim values for the allocated variables.

The base configuration model was subjected to sensitivity analysis to establish the relative effects on plant operational risk of changes in dominant accident sequences and the most risk-sensitive safety functions were identified.

Risk Model. The preliminary risk model structure and information flow diagram were given in the Tenth Quarterly Report - Risk Analysis Methods Development (GEFR-14023-10). The current risk model includes the following features:

- 12 Initiating Event Categories (inclusive of initiators identified for earlier BR risk assessment studies).
- 5 System Response Functions (detection, shutdown, primary coolant flow, main heat sink, backup decay heat removal).
- 8 Generic Accident Types (TOP, LOF, LOHS, PLOF, PLOHS, TOP/LOHS, TOP/LOF, and qualified success). (P stands for "protected;" LOHS for "loss of heat sink;" TOP and LOF are familiar acronyms).
- 4 Core Response States (neutronic shutdown, core coolability, debris subcriticality, debris coolability).
- 7 Core Damage Categories (degree of core damage, energetics, vessel melt-through).
- 4 Containment Response States (extent of containment and cavity damage).
- 6 Material Release Categories (function of core damage categories and containment state).
- 3 Consequence Types (early fatalities, latent cancer fatalities and property damage).

For the current allocation of goals, the calculated risk relative to the assigned risk limit is shown on Figure 6-1. While this allocation meets the overall risk constraint, it is not necessarily cost-optimized and may be modified when optimization studies are completed. Further analysis of the model output, based on intermediate probability matrices built into the model computer code, established the dominant accident sequences contributing to each material release category.
LBR BASE CASE

Fig. 6-1  Calculated Risk - Risk Allocation Model  RISGE19
12/4/79
Sensitivity Analysis. Based on the dominant accident path analysis and judgement as to the ability of design improvement to affect allocated variable probability values, twenty-two allocated variables were selected for sensitivity analysis.

Risk Model outputs were obtained for five values of each allocated variable. The values were judged to cover the anticipated range that might be obtained through design variations on related safety functions.

The selected allocated variables, their frequency or probability values and the top ten variables based on sensitivity ranking are presented in Table 6-1.
**Table 6-1. CDS Allocated Variable Values**

<table>
<thead>
<tr>
<th>Allocated Variable</th>
<th>Goal Frequency or Probability Values</th>
<th>Sensitivity Ranking&lt;sup&gt;a&lt;/sup&gt; (1 = Maximum Sensitivity)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2-50¢/sec. Reactivity Ramp</td>
<td>.33/yr</td>
<td>10.</td>
</tr>
<tr>
<td>Core Support Failure</td>
<td>10&lt;sup&gt;-6&lt;/sup&gt;/yr</td>
<td></td>
</tr>
<tr>
<td>Local Core Blockage</td>
<td>1.1 x 10&lt;sup&gt;-4&lt;/sup&gt;/yr</td>
<td></td>
</tr>
<tr>
<td>Large Leak in Primary Piping</td>
<td>10&lt;sup&gt;-5&lt;/sup&gt;/yr</td>
<td></td>
</tr>
<tr>
<td>Loss of Substantial Primary Pumping Capability</td>
<td>10&lt;sup&gt;-2&lt;/sup&gt;/yr</td>
<td></td>
</tr>
<tr>
<td>Loss of Main Heat Sink (Operating)</td>
<td>2.3/yr</td>
<td>6.</td>
</tr>
<tr>
<td>Loss of Off-Site Power</td>
<td>.1/yr</td>
<td></td>
</tr>
<tr>
<td>Failure to Detect Coolant Blockage</td>
<td>10&lt;sup&gt;-3&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>Reactor Shutdown System Failure</td>
<td>5 x 10&lt;sup&gt;-5&lt;/sup&gt;/demand</td>
<td>1.</td>
</tr>
<tr>
<td>Pump Trip Given &quot;Mechanical&quot; Shutdown Failure</td>
<td>3 x 10&lt;sup&gt;-3&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>Loss of Primary Coolant Flow (Shutdown)</td>
<td>2 x 10&lt;sup&gt;-2&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>Loss of Main Heat Sink (Shutdown)</td>
<td>3 x 10&lt;sup&gt;-3&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>Loss of Dedicated Heat Removal Systems</td>
<td>5 x 10&lt;sup&gt;-6&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>No Neutronic Shutdown Given Failure to Prevent CDA due to Loss of Heat Sink</td>
<td>.15</td>
<td>3.</td>
</tr>
<tr>
<td>No Neutronic Shutdown Given Failure to Prevent CDA due to Loss of Flow</td>
<td>.2</td>
<td>8.</td>
</tr>
<tr>
<td>No Neutronic Shutdown Given Failure to Prevent CDA due to Transient Overpower</td>
<td>.15</td>
<td>9.</td>
</tr>
<tr>
<td>Core Not Coolable Given Neutronic Shutdown</td>
<td>.2/demand</td>
<td></td>
</tr>
<tr>
<td>Debris Not Coolable Given Neutronic Shutdown</td>
<td>.9/demand</td>
<td>5.</td>
</tr>
<tr>
<td>Containment Rupture Given Non-Energetic Whole Core Meltdown</td>
<td>.05/demand</td>
<td>2.</td>
</tr>
<tr>
<td>Continuous Containment Release (No Containment Rupture)</td>
<td>2 x 10&lt;sup&gt;-3&lt;/sup&gt;/demand</td>
<td></td>
</tr>
<tr>
<td>Cavity Failure Given Melt-Through</td>
<td>.8/demand</td>
<td>4.</td>
</tr>
<tr>
<td>Containment Catastrophic Failure Given High Energetics</td>
<td>.2/demand</td>
<td>7.</td>
</tr>
</tbody>
</table>

<sup>a</sup>Ranking index is the ratio of the fractional change of the expected risk (area under complementary cumulative risk curve) to the fractional change in the allocated variable. The sensitivity ranking does not show the magnitudes of the ranking indices of individual variables nor the extent to which high or low consequences are affected by a particular variable change. A more complete set of sensitivity indices and functions reflecting the degree of variable sensitivity and the effect on high or low consequences will be presented in a topical report covering the risk model development and sensitivity study.