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ENERGY SYSTEMS AND TECHNOLOGY DIVISION

ADVANCED REACTOR SYSTEMS DEPARTMENT

October 9, 1980

DOE/ST-171032--T25

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, July - September, 1980
~~DE-AT03-76SF71032~~

AT03-76SF71032

By copy of this letter we are forwarding advance copies of the subject report to Director of the FRS-TMC and the Chief, Safety Section, RRT. A copy is also being sent to the GE/California Patent Group for review and written patent release.

Very truly yours,

T. R. Sandke, Manager
Safety Projects

TRS:mlh
7-2/17A

Attachments

cc: Chief, Safety & Physics Branch, RRT(B-107)
Chief, Safety Section, RRT(B-107)
S. Berk, RRT(B-107)
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MASTER

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10/15/80

PROGRAM HIGHLIGHTS FOR THE QUARTER

A letter report titled "The Nature, Analysis, and Impact of Common Cause Failures on the Design and Licensing of Future LBR's" was completed and transmitted to DOE. The main conclusions included: Identifying the need for a methodology to predict a CCF rate, defining a way to justify the adequacy of CCF defenses, and defining correct uses for the mathematical reliability manual. An appendix presents good defenses which should substantially reduce the impact of CCF. (GE)

LOA
1.0b

The results of a preliminary reliability analysis of the Conceptual Design Study (CDS) Reactor Shutdown System (RSS) indicate that either the primary or secondary RSS can meet the goal for the total CDS-RSS unreliability of 10^{-5} per year. (GE)

LOA
1.2

A draft of the US contribution to the Internal Status Report on "Increasing the Performance and Reliability of Fast Reactor Protection and Shutdown Systems" was completed. (GE)

LOA
1.2

The test article for the SCRS Bellows No. 3 test was analyzed to determine why a leak developed late in the test. The most probable cause was attributed to an accumulation of concentric tolerances. (GE)

LOA
1.2.2.2

All objectives of the SCRS Latch Test No. 3 were met and no test rig or instrumentation failures or irregularities were experienced. Significant conclusions were documented including a reliability estimate for the latch release mechanism of ≥ 0.9991 at a 90% confidence level. A topical report is being prepared. (GE)

LOA
1.2.2.3

Two sensitivity studies were completed in support of the CDS Shutdown Heat Removal System (SHRS) reliability assessment. These studies led to the conclusion that the SHRS reliability still far exceeds the goal (3×10^{-5} per reactor year) given the imposition of either of two operating restrictions: 1) Stated conditions when one or more Primary Heat Transport System (PHTS) pony motors are operating, and 2) Stated conditions when all PHTS motors are in-operative. (GE)

LOA
1.3.2

The safety evaluation and design criteria for a self-actuated shutdown system (SASS) were presented within the frame work of a "system approach" by a draft report forwarded to DOE-SAN. The severity limits to be accommodated by SASS are established in accord with the LOA-2 success criteria and reliability goals. (GE)

LOA
2.1.1.2

The design specification for the Shutdown Core Coolability Model Test was completed and issued. Procurement and fabrication of the test loop components were initiated. The pretest analysis using the BIFR code has been performed to guide the model design and testing. (GE)

LOA
2.2.2

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4703-76SF71032

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As part of an evaluation of CDS in-vessel flow paths for natural convection, a parametric study of core responses to various conditions and operational modes of a Direct Auxiliary Cooling System (DRACS) was performed using the GE one-dimensional thermal/hydraulics computer code GENCON. (GE) LOA 2.2.2

Thermal response analysis of the W-1 SLSF test bundle has been performed with the LIFE-4, COBRA, and THT-D codes. Test data were used as input conditions for the codes and for evaluation of the code results. (GE) LOA 2.2.2

Analytical studies were initiated during this quarter to determine the effects of internal blanket pin geometry, assembly geometry, and steady-state operating conditions on the performance of internal blanket assemblies during various HCDA scenarios. (GE) LOA 3.1.1

Evaluations were made of the effects of initial sodium inventory on reactor cell integrity, post-boildry thermal response of the reactor cell system, identification of potential failure modes beyond sodium boildry, and the determination of cooling requirements to prevent sodium boiling. (GE) LOA 3.2.2

The LBR risk model has been modified to reflect current CDS design. Major model revisions include: Modifying the frequencies of initiating events, estimating failure probabilities for the shutdown systems and heat removal systems, updating core and containment response probabilities, redefining radiological material release categories. CDS core inventory was used as CRAC code input and Hanford site characteristics were employed in the consequence calculations. (GE) LOA 6.1.1

LOA-1: PREVENT ACCIDENTS

1.0 LOA-1 INTEGRATION

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

During this quarter, the effort on this task was directed at a study of the interrelationship between quantitative reliability analysis and an analysis of Common Cause Failure (CCF).

A survey of the techniques used for modelling CCF and the application of this data into quantitative reliability analyses concluded:

1. The first attempts to integrate an analysis of CCF into a reliability assessment were reported in the Reactor Safety Study, WASH-1400. In reality the technique used did not actually provide a quantitative assessment of CCF; it simply made an allowance for them.
2. More recently, several techniques for modeling CCF have appeared in the literature. These were examined for their viability and the availability of data to support them and were found to be seriously lacking. The models are an important step toward an understanding of the large impact CCF might play in the overall reliability of a system. However, the models are much too simplistic to represent the subtle complexities of CCF. Also, the data on multiple failures and failures due to high environmental stresses needed for these model is almost nonexistent for high reliability systems.

As a direct result of this study, GE-ARSD has proposed an alternative to quantifying the impact of CCF. The proposed approach to be used for addressing LBR CCF problems is to proceed along the line of the following three part attack.

1. Make modifications during the early stages of design to include as many diverse defenses against the intrusion of Common Cause Failures as practicable.
2. Prepare a good mathematical model of the system including all identifiable causes of failure. Unidentified causes of failure are naturally absent from the model and no allowance is made for them. Exercise the model to perform analyses that aid in optimizing the design and meeting a reliability design goal, with margin.
3. Advance qualitative arguments, based on the inclusions of diverse defenses against Common Cause Failures, that the overall reliability is adequate or that the margin to the design goal will not be eroded.

A letter report documenting the results of this study entitled "The Nature, Analysis, and Impact of Common Cause Failures on the Design and Licensing of Future LBR's" was transmitted to DOE on August 29, 1980. This letter report also contains an appendix which gives guidance in providing defenses against CCF that should be implemented on all new designs.

The main conclusions to be drawn from the report are these:

1. The present state-of-the-art in reliability assessment does not include a methodology for predicting a CCF rate.
2. The actual impact of CCF on LBR's should be substantially reduced by providing good defenses as suggested in App. B of the paper.
3. The adequacy of the CCF defenses should be justified primarily by the use of substantive qualitative arguments.
4. The mathematical reliability model should be used primarily to provide guidance in optimizing the system configurations and in selection of reliable equipment.

1.0e Update Reliability Program Plan (GE: S. F. Armour)

FRS-TMC completed a review of the draft report, "Reliability Program Plan for Large Breeders, (Part 1, Tasks & Activities)." The principal result of the review is that FRS-TMC is generally in agreement with the overall organization and technical content. The comments will be resolved through a joint effort by WARD and GE-ARSD.

1.2 REACTOR SHUTDOWN SYSTEM RELIABILITY

Reactor System Shutdown Reliability Estimates (GE: G. E. Ingram)

The results of a preliminary reliability analysis of the Conceptual Design Study (CSD) Reactor Shutdown System (RSS) indicate that both the primary and secondary RSS can meet the goal for the total CDS-RSS unreliability of 10^{-5} per year. This was accomplished by two design changes. The first was to redistribute the total rods among the primary and secondary systems. The original distribution was twenty-four to the primary and six to the secondary. The redistribution was twenty-one to the primary and nine to the secondary. The second design change was to increase the number of logic trains in the primary RSS from three to four. The logic, therefore, to the scram breaker system changed from two out of three (2/3) to two out of four (2/4). These changes improved the reliability of both RSSs appreciably.

Reliability of Fast Reactor Protection and Shutdown System
(GE: G. L. Crellin)

A Draft of the U. S. contribution to the Internal Status report on "Increasing the Performance and Reliability of Fast Reactor Protection and Shutdown Systems" was completed.

Sections provided by GE were:

- o I.2.11 Shutdown System Design Description for EBR-II
- o III.3.2.5 Reliability Studies for FFTF
- o III.3.2.6 Reliability Studies for CRBRP
- o IV.2.1.6 Improvements in USA for Conventional Computer Data Processing
- o IV.2.2.6 Research and Development in USA for Computerized Early Detection and Failure Prediction
- o V.3.6 Descriptions of USA Complementary Shutdown System (that is, a self actuated shutdown system).

1.2.2 Secondary Shutdown System

1.2.2.1 Driveline Lower Bellows Component Tests Number 1 and Number 2
(GE: R. Yaspo)

The SG033 portion of the Base Program contributed to the compilation and issue of the topical report CRBRP-GEFR-00533, "SCRS Driveline Lower Bellows Component Test Numbers One and Two,". This cooperative effort was made because the concept, criteria and design of the SCRS for both the Base Program and CRBRP are very similar. Therefore, the results of CRBRP testing have a direct bearing on fulfilling the reliability test and information needs of SG033. The objective of these SCRS Driveline Lower Bellows Tests was to demonstrate the ability of the bellows sealing system to meet and exceed its cyclic design goals and to evaluate various effects such as wear between bellows guiding surfaces and sodium deposits, as well as qualifying the particular design. This test was a success and indicates a significant positive degree of design margin since the design life is 500 cycle and this test article completed 3600 cycles without failure.

1.2.2.2 Bellows Test Number 3 (GE: T. M. Walsh)

Bellows Test No. 3 was run successfully through Phases I and II (Refueling and Full Power Phases) without evidence of a rejectable leak in

the test bellows. The range of vacuum leak rate values measured during these first 360 cycles was between $.26 \times 10^{-3}$ and 5.52×10^{-3} sccm. The leak rate continued in this same range for an additional 200 cycles into Phase III when somewhere between cycles number 570 and 670 the leak rate increased by several orders of magnitude to 69.5 sccm.

Testing continued in this manner for an additional 535 cycles with the leak rate showing no particular regular trend as it ranged between limits of 15 sccm to 186 sccm. Testing was discontinued after a total of 1205 cycles had been completed.

The failure of the lower test bellows was caused by a material build-up ($3/8" \times 1 3/4"$ length) on the sliding shaft (sensing tube) which then came into contact with the ID of the bellows causing a cutting action on the bellows material. The cutting action eventually penetrated through the edge of the bellows diaphragm creating leaks at the first, third, and fifth convolutions.

The cause of the material build-up was due to a rubbing action between the top bellows adapter piece and the sensing tube shaft. The adapter piece and the shaft are of the same material (type 316SS) and both showed signs of scoring at corresponding locations. The lower bellows adapter piece also showed signs of scoring but on the OD and 180° opposite to the upper adapter piece.

A review was made of previous test results which used essentially the same test rig. Bellows Test number 1 and 2 each completed a total of 3600 cycles with no evidence of failure of the test article and no rubbing or gouging of any component reported.

Consideration was given to the possibility of foreign material being lodged between the shaft and the bellows adapter piece, but no evidence of foreign material could be found.

In conclusion, the amount of wear and gouging indicates that rubbing occurred early in the test and consequently metal interference must have occurred at the bellows adapter piece and the sensing tube shaft during the earliest part of the test and most probably at assembly. This action eventually caused an asperity on the shaft which cut the bellows, causing a leak to occur. The most probable cause was an accumulation of concentric tolerances.

1.2.2.3 Latch Scram Accelerated Test No. 3 (GE: A. P. Wood)

The SCRS Latch Accelerated Life Test No. 3 began January 22, 1980, and was successfully completed on May 30, 1980. This period of testing includes 2626 scram releases, 67 days of hold periods, and a two week interruption of the test for vessel measurements. The test was conducted under "overstress" test conditions which included increased temperature, increased load, and a number of cycles greater than seven times the expected latch service life.

Latch Test No. 3 is the third test in a series, using the same latch assembly design, to successfully demonstrate latch assembly reliability. The two previous latch assembly tests also successfully demonstrated latch assembly reliability. This is important because the latch assembly is a critical component of the SCRS. The objectives of the Latch Test No. 3 were to demonstrate the ability of the latch assembly to meet and exceed its cyclic design goals, and to evaluate the effects of wear between the contacting surfaces on latch performance. In addition, this test was statistically designed to allow determination of the effects of cyclic wear, sodium temperature, and hold periods on the latch friction coefficient.

All objectives of the Latch Test No. 3 were met and the following conclusions may be drawn:

1. The ability of the latch assembly to exceed its design goal was shown by completing a number of cycles greater than seven times the expected life of the latch assembly.
2. There was no evidence of erratic or off-design behavior of the SCRS Latch Test Assembly.
3. The friction coefficient ranged in value from .252 to .693, well within the specified limits of .2 to 2.0.
4. An increase in the friction coefficient corresponded to an increase in the accumulated number of cycles. However, after about 1000 cycles, the latch friction coefficient stabilized at a value of about .60.
5. There is an interaction between the sodium temperature and the number of cycles for predicting friction coefficient values.
6. The hold periods and test interruption seem to have had no effect on the value of the friction coefficient.
7. Considering the data to consist of 2626 trials with no failures yields a reliability estimate for the latch release mechanism of $\geq .9991$ at a 90% confidence level.
8. Considering the data to be normally distributed implies less than a .012 probability that the friction coefficient is below .20, and a probability much less than .0001 that the friction coefficient is greater than 2.0.

There were no test rig or instrumentation failures or irregularities. The main anomaly was the interruption of the test from March 10, 1980 to March 24, 1980 for vessel measurements in support of the SCRS Program. During this time, the vessel was drained and cooled. Following required work, the vessel was refilled and returned to the 1050°F test temperature. After again establishing test conditions of sodium temperature, flow, and purity, a series of releases were re-run, beginning March 25, 1980, to ascertain any effect of

the shutdown period. No evidence of any changes in release patterns or expected friction coefficients were found. Consequently, on April 1, 1980, approval was given to resume the test in accordance with the test procedure. Although 2338 cycles were originally planned for this test, 2626 cycles were actually performed because of the test interruption. The final releases at 3250F were performed May 29, 1980. Following final characterization runs and review of the data, the test was terminated and sodium was drained from test loop STL-1 on May 30, 1980. A topical report describing this test is currently in preparation.

1.2.2.4. Latch Real Time Test (GE: R. Yaspo)

The review of the test basis specification for a latch Real Time Test has been completed. This test basis specification will specify the third and fourth real time tests in which each latch will be put in the latching mode for up to one year without releases. These tests are to investigate the variability effects on reliable scram performance of long-term standby periods in the hot sodium environment.

1.3 SHUTDOWN HEAT REMOVAL SYSTEM RELIABILITY

1.3.2 Auxiliary Heat Removal Systems (GE: J. G. Elerath)

Two sensitivity studies were completed in support of the CDS Shutdown Heat Removal System reliability assessment. Both of the studies fit into the general category of design related changes. They investigate the impact that possible operational restrictions or future design requirements may have on the SHRS reliability. The selection of these is to a large extent based on the general philosophy that "the most reliable system is one which is least well defined". It seems that as the design of any complex system progresses, the conservatism that made it so attractive initially, begin to diminish. As one design problem is overcome its remedy may impact some other aspect of the design, with the net effect being a decrease in reliability. Coupling this philosophy with a detailed knowledge of progress of the CRBRP design permits extrapolation to these hypothesized design restrictions.

Based on these two sensitivity studies, it is concluded that the SHRS reliability still far exceeds the goal when either of the following two operational restrictions are imposed:

- 1) When any PHTS pony motor is in operation, only the DRHXs adjacent to the operating PHTS loop will be capable of heat removal. No restrictions are placed on DRHX capabilities when all pony motors are off, i.e., natural circulation with the reactor vessel has been established.

- 2) In addition to the restriction during pony motor operation, DRHXs on opposite sides of the reactor vessel must be operative when all PHTS pony motors are inoperative, i.e., natural circulation within the reactor vessel requires two functional DRACS loops; one on either side of the reactor.

The first sensitivity study addresses uncertainties in primary sodium circulation within the reactor vessel when one or more Primary Heat Transport System (PHTS) pony motors are operating. It assumes that for any operating PHTS pony motor only the two Direct Reactor Heat Exchangers (DRHXs) nearest to the PHTS outlets receive sufficient circulation to remove post shutdown heat. This assumption is based on the resulting complex flow paths, which have not yet been fully analyzed, and the potential to stall flow through the non-adjacent DRHXs. The net effect of this assumption is an increase in the dependence between the DRACS and IRACS under forced circulation. The natural convection capabilities of the DRACS alone, however, remain unchanged with respect to the Base Case. The probability of SHRS failure under the above assumptions increased by a factor of about 5 with respect to the Base Case but is still far below the reliability goal of 3×10^{-5} failures per reactor year.

The second study incorporates the assumptions of the first study for the condition when one or more PHTS pony motors are operating, and also restricts the usage of DRHXs under the condition of natural convection within the reactor vessel. The additional restrictions arise out of the concern that removing heat from one or both DRHXs on one side of the reactor vessel may set up unacceptable temperature differentials across the reactor vessel, overstressing the upper internal structure. Thus, it is assumed that at least two DRHXs are required for all time periods, one on either side of the reactor vessel. These assumptions have an additional impact on SHRS reliability, but even under these pessimistic assumptions, there is still a great amount of margin between it and the specified reliability goal of 3×10^{-5} per reactor year.

Several other interesting results also came out of these studies. The first is that a major portion of the SHRS reliability is achieved through the Direct Reactor Auxiliary Cooling System (DRACS) alone in the Base Case and Sensitivity Study 1. The efforts to make them as inherent as possible is very effective in achieving high reliability. The second is that in the Base Case and both of these sensitivity studies, the long term heat removal missions (30 days) contributes the largest portion of the total unreliability. Furthermore, missions in which two main heat transport loops are unavailable at the inception of the SHRS demand contribute the largest percentage of unreliability in the long term outages. An example of such a case is a shutdown from three loop operation in which one Main Heat Transport System (MHTS) loop failure necessitates the three loops operation to begin with, and a second MHTS loop failure produces the SHRS demand. Additional sensitivity studies are being performed to investigate the reliability impact, if any, of other operational restrictions.

LOA-2: LIMIT CORE DAMAGE

2.1 REACTOR SHUTDOWN SYSTEM FAULT ACCOMMODATION

2.1.1.2 ALTERNATE CONCEPTS (GE: L. F. Peterson)

Further work on alternate concepts has resulted in a detailed concept description of the Hydrostatic Supported Absorber system. A pressure differential control device has been incorporated into this concept to improve flow anomaly detection. An upper end housing concept design (or design concept) has been made incorporating this feature and also a means for utilizing full bundle flow area during insertion while limiting coolant bypass of the absorber during the holding of the absorber out of core during normal reactor operation.

Further conceptual work on the Thermionic Switched Electromagnetic Latch has been accomplished with consideration of the application of various flow patterns. Of particular interest is the flow pattern utilizing hydraulic assist, although the assistance available may be limited.

2.1.1.2 ABSORBER DESIGN (GE: L. F. Peterson)

Conceptual absorber design was completed in this period. With the concurrent development of the hydrostatic supported absorber system, the conceptual upper end housing of that system has been incorporated into the absorber in place of the restrictive orifice being required by the levitated absorber concept used with the fluidic valve. The articulated absorber concept was reviewed during the design for applicability. Its use with the levitation system may be logical but for the hydrostatic design, the .093 annular gap width is highly restrictive and will tend to limit the absorber terminal velocity to low values. Figure 1 shows calculated terminal velocity for an absorber of the configuration but considered as a solid volume, for several annular widths. Significant improvement is evident with the larger annulus, which would also exist as an add on to the flow through the absorber - a feature which also must be included on the absorber to achieve rapid insertion. It appears that the logical design is one piece with generous annular clearance. Articulation causes substantial increase in complications of construction, multiplies the number of absorber pins by the number of articulated sections and causes added pressure drop to flow - of primary importance during insertion. It may be, however, that core design may mandate a maximum diameter absorber within the guide tube diameter restriction. The penalty is then within the possible minimum insertion time.

2.1.1.2 CRITERIA AND FUNCTIONAL REQUIREMENTS (GE: E. Kujawski)

Milestone 2.1-9, "SASS Criteria and Functional Requirements", has been completed, and a draft of the resulting report entitled, "Safety Evaluation and Design Criteria for a Self-Actuated Shutdown System", (XL-898-00567 SG076-AB, August 29, 1980) has been forwarded to DOE-SAN.

The major results are as follows:

The safety evaluation and design criteria for a self-actuated shutdown system (SASS) are presented within the framework of a "system approach". The severity limits to be accommodated by SASS are established in accordance with the LOA-2 success criteria and reliability goals. The transient and long term integrity of the reactor are addressed. The design basis transients have been explicitly quantified for a 1000 MWe mixed oxide heterogeneous LMFBR, the "CDS Phase II" core. A partial set of reactor responses for selected transients has been analyzed as a function of the SASS characteristics such as reactivity worth, trip points, and insertion times. It is explicitly shown that four SASS assemblies can successfully meet the desired design criteria.

2.1.1.2 ABSORBER THERMAL HYDRAULICS (GE: S. Wong)

A parametric thermal-hydraulic study on alternative SASS absorber assembly designs using electromagnetic control drive concept was initiated. In order to meet the flow requirements of cooling of the absorber pins and to create hydraulic scram assist force, a seal ring to restrict the bypass flow was considered.

An analytical technique was developed to predict the hydraulic scram assist force and absorber assembly insertion time. The following conclusions were drawn from the analysis:

- o Without the seal ring there is no hydraulic scram assist force at all.
- o Hydraulic scram assist force is a function of seal ring clearance and seal ring position.
- o For a given seal ring clearance, higher hydraulic scram assist force is resulted if the seal ring is attached to the bundle tube; lower hydraulic scram assist force is resulted if the seal ring is attached to the guide tube.

Results from a preliminary evaluation show that, for seal ring clearance = 0.1 in. and 40 in. stroke, the scram insertion time without dashpot is 0.6 and 0.82 seconds for the attached seal ring (to the bundle) case and detached case, respectively. It should be noted that dashpot is required to slow down the bundle near the end of the stroke to avoid impact.

2.2 SHUTDOWN HEAT REMOVAL SYSTEM FAULT ACCOMMODATION

2.2.2 Coolability of Core

SHRS Requirements and Experimental Investigations (GE: T. A. Shih/
S. Grewal)

The Design Specification for the Shutdown Core Coolability Model Test was completed and issued. Procurement and fabrication of the test loop components were subsequently initiated. The pretest analysis using the BIFR code has been performed to guide the model design and testing.

A mini test loop consisting of two channels with upper and lower plena was constructed to test the performance of the flow meter and flow visualization methods.

The hydrogen bubble flow visualization technique has been tried and proven feasible for the flow redistribution experiment. Photographs were taken and found acceptable for characterizing the fluid flow.

A conceptual design of the reactor system water model was completed. The model will be geometrically scaled as 1/10 of the CDS reactor. It will have 0.5 MW power and 250 gpm total flow. The key objective of the system model test is to investigate natural circulation flow pattern in the reactor vessel through direct visualization and other reliable measurement techniques.

The THERMIT model for in-vessel natural circulation analysis has been made operational for the development of SHRS criteria. Input data are being modified to properly represent the updated CDS design. Simulations of DRACS and check valves were accounted for.

2.2.2 Coolability of Core

Analysis and Computer Modeling (GE: D. D. Knight, D. W. Lunsford,
R. W. McGriff)

As part of an evaluation of CDS in-vessel flow paths for natural convection, a parametric study of core responses to various conditions and operational modes of a Direct Auxiliary Cooling System (DRACS) was performed using the G.E. one-dimensional thermal/hydraulics computer code GENCON. The code was extended to include a representation of the DRACS. Several assumptions with regard to the flow diode and check valve configuration and the sequence of events leading to activation of the DRACS were introduced because of the lack of detailed definition of CDS design. Flow resistance of the DRACS primary loop was calculated based on the original check valve configuration and then varied over the range $\pm 80\%$. Two starting conditions were used: first, with the DRACS initially at its design operating temperatures, and second, with the DRACS secondary NaK loop initially at a cold condition. For both cases, a low sodium level in the reactor and zero primary system flow were assumed. For all scenarios, reactor scram was assumed to occur concurrently with DRACS activation. The entire core decay heat load was assumed to be dissipated through three of the four DRACS loops. Plots of average core outlet temperature versus time show increasing peak temperature as flow resistance increases and also increased peak temperature when starting from the cold condition. Peak core outlet temperatures were in the range of 1350°F to 1550°F.

The results of this study indicate that the DRACS will be capable of dissipating the entire core decay heat load, even if flow diodes with conservatively-high head-loss characteristics are employed. Flow resistance, however, must be kept as low as possible in order to limit core temperatures. Work in the codes comparisons project was principally concerned with the COMMIX-1A code. A sample steady-state upper-plenum mixing problem supplied by ANL was successfully executed. The code limits were then tested by having the code evaluate a transient heat conduction problem. The code successfully solved the problem when only the energy equation was solved, but developed an instability when the mass and momentum equations were solved along with the energy equation. Numerical diffusion properties of the code were examined and at this time it does not appear that numerical diffusion will be a problem at low-flow conditions.

2.2.2 Coolability of Core

SLSF W-1 Data Analysis (GE: D. D. Knight)

Thermal response analysis of the W-1 SLSF test bundle have been performed with the LIFE-4, COBRA and THT-D codes. Test data were used as input conditions for the codes and for evaluation of the code results. Steady state fuel condition during the irradiation was calculated with LIFE-4. During fuel restructuring and burn up, the fuel/cladding gap decreases and the gas conductivity decreases due to fission gas. These effects are compensating so the LIFE-4 results show little change in gap conductance with time.

Comparisons of calculated fuel temperatures with in-fuel thermocouple data were used to adjust the gap conductance values. The adjusted values range from 660 to 1070 BTU/hr ft²°F for the LOPI tests, increasing with time.

THT-D analyses at steady-state conditions were used to adjust the gap conductance values to match in-fuel thermocouple calculations and data. Transient in-fuel thermocouple temperature decrease in the LOPI experiment was predicted with THT-D. Comparisons with data indicate the presence of helium and fission gas inside the thermocouple sheath (replacing the original argon).

LOA-3: MAINTAIN CONTAINMENT INTEGRITY

3.1.1 Accommodation Within Primary System Boundary

Analytical Development

Behavior of Internal Blanket Fuel (GE: D. B. Atcheson, K. E. Gregoire)

Analytical studies were initiated during this quarter to determine the effects of internal blanket pin geometry, assembly geometry and steady-state operating conditions on the performance of internal blanket assemblies during various HCDA scenarios. In parallel with the analytical scoping studies, data from steady-state irradiations of blanket pins are being compiled.

Initial calculations were intended to provide typical inner blanket pin thermal conditions during both steady-state and transient operation, so that the blanket pin fuel and cladding phenomena of greatest interest in safety evaluations could be identified. The results of analyses of the Loss-of Flow (LOF) scenario were of particular interest because of the marked difference between blanket pin and fuel pin post-irradiation fuel-cladding gap size. CDS peak-power fuel and internal blanket pin performance during an unprotected flow coastdown accident was analyzed with the LIFE-4 code. The fuel pin was assumed to be at beginning-of-life thermal conditions and the internal blanket pin at end-of-life (2.3% burnup) conditions for the analysis. The respective axial peak powers were 12.4 kw/ft (407 w/cm) and 10.4 kw/ft (341 w/cm) for the fuel and internal blanket pins, respectively.

At the end of its steady-state irradiation (i.e., after two years), the inner blanket pin was predicted to have an open fuel-cladding gap. Conversely, the fuel pin gap closed after 200 to 500 hours of irradiation, depending upon the power vs. time history assumed. During the subsequent loss of flow transient, the fuel-cladding gap of the inner blanket pin remained open according to the analysis. This suggests that the full reactivity effect of blanket fuel axial expansion may affect the LOF scenario, independent of the burnup of the blanket pins. These LIFE-4 predictions of blanket pin gap sizes have yet to be tested by comparison with data; this comparison is planned. In addition, the effect of very low steady-state cladding temperatures on the transient overpower behavior of inner blanket pins is to be estimated based on existing FCTT cladding data and theoretical models describing the fuel adjacency effect on cladding.

3.2.2 Accommodation Within Inner Cells

Evaluation of Core Debris Accommodation Within Inner Containment Cell

(GE: K. H. Chen, E. L. Gluekler, C. M. Hu, S. T. Lam)

Evaluations were made of the effects of initial sodium inventory on reactor cell integrity, post-boildry thermal response of the reactor cell system, identification of potential failure modes beyond sodium boildry, and the determination of cooling requirements to prevent sodium boiling. This work concluded the evaluation of long-term/permanent core debris retention capability of an MgO delay bed; composite-cylindrical-wall reactor cell design with no cooling.

The use of a large sodium inventory can reduce the temperature level inside the reactor cell after sodium boildry and prevent the melting of the reactor vessel. For the large breeder reactor (LBR) plant investigated, using a 1.9×10^6 kg sodium inventory, the reactor cell temperature would rise to a maximum of 1160°C after sodium boildry; no melting of the reactor vessel would occur. For a 1×10^6 kg inventory, reactor vessel melting would occur 16 days after sodium boildry.

Evaluations of the reactor cell system response showed that besides the high temperature environment inside the reactor cell after sodium boildry, rapid radial penetration of the core debris into the reactor cell side wall would be the major concern at the early stage of the accident. The reactor cell structure foundation could fail before the MgO bed is penetrated in the downward direction. For the LBR investigated, the MgO bed would require a radius of 10 m to prevent side wall penetration.

The reactor cell concept with MgO bed and a composite wall design but without a cooling system is capable of preventing the reactor vessel from melting, stabilizing the debris pool growth and maintaining a low side wall concrete structure temperature (173°C), but it does not provide enough thermal protection of the reactor head ledge concrete structure (936°C at sodium boildry, 1105°C at 75 days after sodium boildry).

The temperature and pressure rises of the RCB with the improved reactor cell design are insignificant during the sodium boiling period.

Long-term integrity of the reactor cell system is uncertain because the long-term structural response of the system at elevated temperatures is uncertain and the properties of the core debris mixture are not well defined. In order to maintain a low temperature level in the reactor cell, and hence the structural integrity, the sodium coolant must be prevented from boiling.

The CACECO code was used to establish the extent of cooling required. Two cases were studied. In one case an active cooling system was added to the reactor cell. The cooling system was initially kept at a constant heat removal rate until it equalized the core debris decay power.

After that the heat sink capacity was reduced to maintain a zero net heat input to the reactor cell. The results showed that the liquid sodium in the reactor cell could be kept from boiling with a cooling capacity of ≥ 5.2 MW(t). In the other case a fictitious heat structure with large heat capacity was introduced in the containment building to simulate a passive heat sink of 5 MW(t) capacity. It was found that the heat sink had negligible effect on the response of the reactor cell, and there was only a slight reduction in the temperature level of the RCB. Hence, a cooling system incorporated in the reactor cell region appears to be the more efficient approach. Although a minimum cooling rate of 5.2 MW(t) has been determined to be capable of preventing sodium boiling, the minimum cooling capacity required to maintain the reactor cell structural integrity is uncertain because of the uncertainty in the maximum temperature level at which long-term structural integrity could be maintained.

LOA-4: ATTENUATE RADIOLOGICAL CONSEQUENCES

4.2.1 Radiological Consequences Assessment Methodology (GE: E. L. Gluekler)

The identification of measures (in quantified units) of radiological consequences suitable for use as criteria in reactor safety and design-option evaluation was completed. Results indicate that measurement units which directly convey information on potential dose to bone marrow, lung, thyroid or whole body at distances of about 0.5 to 5 miles from the reactor are sufficient measures for use in evaluating: 1) the sensitivity of the parameters for in-containment aerosol behavior and transport, 2) the effect of hold-up time in containment, 3) the effect of warning time prior to radionuclide release, 4) the effect of height of release, 5) the effect of natural shielding, and 6) the effect of deposition velocity.

Estimates of health effects evaluations are also of importance in developing therapeutic approaches for mitigation of harmful effects. This is particularly pertinent in such cases as latent fatalities involving delay in the onset of various cancers that may result in death.

In using total potential risks as criteria for reactor safety and design specifications, it is necessary to estimate doses, population exposed, incidence of health effects as a function of dose, and levels of property damage.

A detailed description of this work was separately reported (J. D. Teresi, "Measures of Radiological Consequences in Nuclear Reactor Safety Assessments", September 1980).

6. R&D INTEGRATION

6.1 INTEGRATED ANALYSIS

6.1.1 Methodology and Data Base Development (GE: M. I. Temme, D. E. Hurd)

Risk Analysis Methods Development - Subtask 6.1-3, Complete Development of LBR Risk Assessment Model and Report

The risk model was modified to reflect the current CDS design. Major model revisions were made in the following areas:

1. The initiating events were reviewed and their frequencies were modified to be consistent with CDS design. Seismic related event frequencies were calculated based on Western U.S. earthquake data.
2. Failure probabilities for the shutdown systems and the heat removal systems were estimated based on current CDS reliability evaluations and engineering judgment of system reliability degradation due to stressful accident initiation and accident progression conditions.
3. Core and containment response probabilities were reviewed and updated to reflect current CDS design definition.
4. Radioactive material release categories were redefined to provide magnitudes and release parameters consistent with CDS cavity and containment design features.
5. CDS core inventory was used as CRAC code input and Hanford site characteristics were employed in the consequence calculations.

A preliminary risk evaluation was performed on the CDS. Sensitivity analysis was performed to determine the effect of input data uncertainties on the risk analysis results.

The risk analysis results are presently being documented for inclusion in the CDS Phase II Report.