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ENERGY SYSTEMS AND TECHNOLOGY DIVISION

ADVANCED REACTOR SYSTEMS DEPARTMENT

July 14, 1980

DOE/SF / 71032 -- T29

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

MASTER

Dear Mr. Harbison:

SUBJECT: Quarterly Technical Progress Report, April - June, 1980
DE-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)
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REC'D

LOA-1: PREVENT ACCIDENTS

1.0 LOA-1 INTEGRATION

ATO3-76SF 71032

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

This task, over the year, has been subdivided into three phases:

1. A review of the literature on Common Cause Failure (CCF) for the purpose of determining its characteristics and embodying those characteristics into a definition.
2. A review of the available reactor operating experience for the purpose of determining the defenses that can be applied, particularly during design, to prevent most causes from failing all redundant paths.
3. A review of the literature on CCF assessment for the purpose of recommending an approach to be used during LBR design.

The results of the Phase 1 study on the characteristics of CCF were reported in the October - December 1979 quarterly report. In the second phase of the study, the literature of experience data on operating Light Water Reactors (LWRs) was reviewed with the particular purpose of noting the actions that can be taken during design to provide defenses against CCF. The information was written into a document entitled "Guidelines for Defenses Against Common Cause Failure." The document is generic and can be used in any LBR design. The topic is timely for the present state of CDS design and the document was transmitted to the CDS Project with the suggestion that it be made an appendix to the CDS Overall Plant Design Specification. The principal recommendations made in this document are to implement specific tasks in both the Management-Administrative and the Design Engineering Areas of responsibility to assure that the Appendix A, 10CFR50 requirements are met.

1.0c Development of Methods for Reliability Assessment (GE: G. E. Ingram and O. Gokcek)

More efficient and accurate methods for the assessment of safety system reliability, where consideration is given to maintenance and repair, have been in development during this quarter. The Markovian technique is ideally suited for this type of analysis. However, it does frequently require solving a large set of linear differential equations which is a difficult and lengthy calculation. When the number of states used to formulate the problem is large, concern for costly computer time will often lead to the avoidance of a Markov Analysis. Fortunately, the computational time can be substantially reduced by merging all of the mergeable states of a Markov process.

A Markov process is called mergeable if the states of the original process can be grouped as superstates such that the result is also a Markov process. Thus, the necessary and sufficient condition for a Markov process to be mergeable is that the transition probability be the same from every mergeable state to the other states. In order to implement this mergeability feature in any Markov analysis, a computer code is being developed. This code will be used as a subroutine to merge all possible states before starting any numerical computation.

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

A draft of the report "Reliability Program Plan for Large Breeders" detailing the Tasks and Activities needed to fulfill the TMC LOA-1 plan was submitted to the TMC to meet the control milestone. The Tasks and Activities had the concurrence of WARD who has joint responsibility in preparing this program plan which will ultimately include priorities, cost and schedules, and status.

1.1 REACTOR SYSTEM RELIABILITY

1.1.4 Monitoring and Control System (GE: I. M. Jacobs)

1.1.4.2 Control System/Operator Interaction

The Control Room Design problem is an area that is ripe for the best advisory input that the industry has to offer. Accordingly, during the first quarter FY80 a literature search was conducted and information pertinent to human factors considerations in control room design was read, accessed, and summarized. From this study, it became clear that, there are many design features that are well recognized and that can be readily incorporated into the control room and panel board design and layout. These features, once incorporated, will undoubtedly improve the operator-plant interface capability even though the magnitude of the improvement may be difficult to quantify.

In order to make the most immediate and effective use of the information accumulated in this search, those well recognized design features were gathered into one document entitled "Guidelines for Human Engineering in the Design of Control Rooms." This document is generic and can be used in any LBR design. The topic is timely for the present state of CDS design and the document was transmitted to the CDS Project with the suggestion that it be made an appendix to the CDS Overall Plant Design Specification. The principal recommendations provided in this document are directed at achieving optimum integration of plant operators with mechanical or electronic components for the purpose of attaining safe, reliable, and economic operation of the power plant. This is accomplished by listing specific design guidelines for the control room, control room board, and control room panels as well as the plan for implementing new technologies in human engineering.

1.2 REACTOR SYSTEM RELIABILITY

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

During this quarter and as part of the Safety Base Program, in support of the design and reliability analyses of the Reactor Shutdown System (RSS) for the Conceptual Design Study (CDS), ongoing analyses are being made to obtain estimates of the reliability of the RSS. Efforts to date indicate that the distribution of control rods between the primary and the secondary systems is the key factor in meeting the reliability goal of 10^{-5} for the RSS. Sensitivity analyses will be made to determine the effect of potential design changes on the reliability of the system.

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

Drafts of the U.S. contribution to the International Status report on "Increasing the Performance and Reliability of Fast Reactor Protection and Shut-

down Systems" to be provided by GE were made available to the FRSTMC on June 27. The sections on FFTF and CRBRP Reliability Study Results have been delayed until July 11 because of the difficulty in identifying referenceable study results consistent with the current system descriptions. Reporting in these sections will now be primarily on the types of studies performed and qualitative results.

Sections provided by GE were as follows:

- o I.2.11 Shutdown System Design Description for EBR-II
- 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 first draft of the topical report, "Driveline Lower Bellows Component Test Number 1 and Number 2", was prepared during June. Because the concept, criteria and design of the SCRS for both the Base Program and the CRBRP are very similar, 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 is 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: R. Yaspo)

This component test is the third of four tests of its kind. It is presently conducted at the GE-ARSD Sodium Test Facility in San Jose, California and this test is specified by Test Basis Specification 23A2547.

The Secondary Control Rod Unit utilizes 4 welded translation bellows to form a buffer chamber to prevent sodium liquid and/or vapor from contaminating the driveline mechanism.

The criterion for success is to demonstrate that the leak rate for the combined bellows arrangement is less than 18.0 SCCM argon under all testing conditions of pressure and temperature, and bellows length variations. With respect to Table 1, test article 1 & 2 (each consisting of two bellows) have already been tested successfully, (see Section 1.2.2.1).

Test Article No. 3 was being tested and phase 2 was started with a leak rate less than 0.003 SCCM argon when a failure occurred at 670 cycles on April 24, 1980.

During a normally scheduled leak rate measurement, it was found that the leak rate was 70 SCCM whereas the acceptable is 18 SCCM. During May 1980, subsequent measurements were made at successive 10 cycle increments. The leak rate decreased to about 17 SCCM and then increased to close to 100 SCCM. The test was formally terminated before total bellows failure to preserve the leak evidence and the bellows test articles were disassembled for inspection and metallographic examination. Three leak paths were observed as well as heavy gouging on the inside convolutions where the leak paths were observed. A complete failure report is presently being compiled (Failure Report; GE-ARSD 067).

1.2.2.3 Latch Scram Accelerated Test No. 3 (GE: R. Yaspo)

This component test, started on 1/21/80 is also the third test to demonstrate SCRS latch reliability. The tests were designed to investigate the variability of the coefficient of friction and to assure it is in the range of 0.2 to 2.0 and no self-welding occurs. Table 2 is the test performance requirements as specified in the test basis specification.

This test was successfully completed in June 1980. The coefficient of friction ranges from 0.252 for the few initial scrams to 0.692. Although only 2,338 cycles were planned for this test, 2,616 were actually performed because of an unplanned test interruption which required repeat of some cycles. This occurrence did not seem to effect the results. A formal test report is currently in preparation.

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

A test basis specification for a Latch Real Time Test was completed on 3/24/80 and is now being reviewed. 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. (See Section 1.2.2.3). 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

SHRS Integrated Test Plan (GE: S. B. Tulloch)

Two meetings have been held with GE-ARSD and GE-CR&D personnel to develop an outline for SHRS Integrated Test Plan (ITP). The tentative outline of the ITP has been prepared and rationale for selection of SHRS tests are being formulated. 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", GEF-00493, issued by GE-ARSD in September 1979. This report 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 licensing 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.

Identification of all seven key subsystem test needs has now been completed. The ITP draft outline was completed May 16, 1980. Currently completed are the following: 1) components list, 2) concerns list, 3) tests need to meet concerns list, 4) subsystem similarities of components list, 5) list of tests by primary organization, either Design or Safety/Reliability.

A four-phase test program for planning of tests is postulated, 1) research, materials, generic data, and proof of principal; 2) design scoping tests, subparts, design features, attribute tests; 3) design proof and validation prototype part and small subsystem performance and life; 4) subsystem, system and plant level tests.

1.3.1 Main Heat Transport System

1.3.1.2 Component and System Reliability

1.3.1.2.1 LMFBR Steam Generator Reliability - Single Wall vs. Double Wall (GE: A. P. Wood)

The analysis of steam generator reliability continued and included an extensive data search and sensitivity study of all parameters. These analyses have been summarized in a draft report on "LMFBR Steam Generator Reliability, Double-Wall versus Single-Wall." This report will be issued in July as a topical to meet Milestone 1.3-2.

The report analyzes the differences between double-wall tubes and single-wall tubes. Four concepts were selected and modelled using Markovian techniques. A data search provided failure and repair rates for each of four concepts. Extensive sensitivity studies considering the effects of variations in the failure rates, repair rates, and operating assumptions. These sensitivity studies help determine the important parameters in steam generator reliability, identify the critical regions of those parameters, and show which steam generator option is the most reliable.

The study shows that tube and weld failures are the dominant cause of steam generator unavailability. The reliability of the single-wall design is very dependent on, and very sensitive to changes in, tube and weld failure rates. Since double-wall tubes provide a second barrier against a sodium-water reaction, their reliability is dependent not only on tube and weld failure rates but also on the failure rates of the second barrier given that a tube or weld has been penetrated. When double tubesheets are employed to provide a capability to detect penetrations of the first barrier, steam generator availability is very dependent on the number of leaks for which the steam generator will cease operation. The report concludes on the basis of reliability issues that the recommended internal steam generator design for future LMFBR development is double-wall tubes with a single tubesheet. The concept has safety, reliability, and availability advan-

tages over other concepts and retains those advantages over a wide range of failure rates.

1.3.1.2.2 Influence of the Main Condenser on SHRS Reliability (GE: J. G. Elerath)

In support of the Safe Shutdown Reliability Program, quantitative reliability studies were performed to investigate the relationships among various auxiliary SHRS configurations and the main condenser. The results of these studies illustrate that the condenser's influence on overall SHRS reliability is highly dependent on the design selected for the auxiliary configuration. 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 28. This variability is primarily due to the different amounts of hardware that are shared between the dedicated auxiliary heat removal system and the condenser's systems for each SHRS configuration. The inverse of the factor is a measure of "reliability worth" of the condenser in terms of a conditional probability of failure, i.e., the probability that heat removal through the main condenser fails, given that a shutdown initiator has occurred and the Auxiliary Cooling System has failed. Thus, 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 auxiliary heat removal system increases.

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

GE-ARSD has been working with WARD in analyzing the reliability of the CDS (Phase II) Shutdown Heat Removal System (SHRS). The primary emphasis has been in the development of the models for the main heat transport system and the dedicated heat transport systems, as well as preparation of the input data for evaluating the models using the GE computer code PROBCALC. A preliminary reliability evaluation of the base case configuration has been completed. This assessment of the CDS designs for the SHRS, as well as that of the Plant Protection System, were discussed at a meeting held at AI on May 29, 1980. Sensitivity studies have been proposed by GE-ARSD and will comprise the next phase of the analysis. As more information becomes available, the reliability of the base case will be updated.

TABLE 1 - TEST ARTICLE TESTING REQUIREMENTS

PROJECTS		CRBRP (For Information Only)		SG033 (LBR)	
TEST PHASES	OPERATING PARAMETERS	TEST ARTICLE NUMBERS			
		1	2	3	4
I Refueling	Cumulative # Cycles Pressure, psi differential Sodium Temperature, °F	35 65 400	35 65 400	15 65-Scram* 95-Retract* *1050-Scram 400-Retract	15 65-Scram* 95-Retract *1050-Scram 400-Retract
II Full Power	Cumulative # Cycles Pressure, psi differential Sodium Temperature, °F	360 15-Scram 65-Retract 1000	360 15-Scram 95-Retract* 1050*	360 65-Scram* 95-Retract* 400/173** Cycles *1050/172 Cycles	360 65-Scram* 95-Retract* *1050/173*** Cycles 400/172 Cycles
III Life Cycles	Cumulative # Cycles Pressure, psi differential Sodium Temperature, °F	3600 15-Scram 65-Retract 1000	3600 15-Scram 95-Retract* 1050*	7200 65-Scram* 95-Retract* 400/3420** Cycles *1050/3420 Cycles	7200 65-Scram* 95-Retract* *1050/3420*** Cycles 400/3420 Cycles
I II, & III	Init. Assy. Length High Press. Side Press. Direction	Nominal Argon Normal	Maximum* Argon Normal	Maximum* Argon Normal	Maximum* Sodium Reversed*

*Over stressed conditions

**Run 400° cycles first

***Run 1050° cycles first

TABLE 2
TEST PERFORMANCE REQUIREMENTS

TEST ARTICLE NUMBER 3					
Sodium Temp. °F	Fast Release Cycles	Hold Period days	Slow Release Cycles	Number Sequences Performed	Test Identifi. Number
325 ⁺²⁵ ₋₀	0	0	5	1	1
	10	0	5	1	2
400	0	0	5	1	3
1050	10	0	1	4	4
		1	3	1	5
		0	1	4	6
		2	3	1	7
		0	1	4	8
		3	3	1	9
		0	1	5	10
		0	1	4	11
400	10	1	3	1	12
		0	1	4	13
		2	3	1	14
		0	1	4	15
		3	3	1	16
		0	1	5	17
		0	1	4	18
		1	3	1	19
600	10	0	1	4	20
		2	3	1	21
		0	1	4	22
		3	3	1	23
		0	1	5	24

TABLE 2
TEST PERFORMANCE REQUIREMENT (Continued)

TEST ARTICLE NUMBER 3					
Sodium Temp. °F	Fast Release Cycles	Hold Period Days	Slow Release Cycles	Number Sequences Performed	Test Identifi. Number
1050	10	0	1	4	25
		1	5	1	26
		0	1	4	27
		2	5	1	28
		0	1	4	29
		3	5	1	30
		0	1	4	31
		7	5	1	32
		0	1	4	33
		30	5	1	34
1050	360	0	10	1	35
400	0	0	10	1	36
600	0	0	10	1	37
1050	360	0	10	1	38
400	0	0	10	1	39
600	0	0	10	1	40
1050	360	0	10	1	41
600	10	0	1	5	42
		0	5	2	43
400	10	0	1	5	44
		0	5	2	45
325 ±25 -0	0	0	5	1	46
	10	0	5	1	47
TOTAL					
2090	Fast Releases				
248	Slow Releases				
61	Hold Period Days				

LOA-2: LIMIT CORE DAMAGE

2.1 REACTOR SHUTDOWN SYSTEM FAULT ACCOMMODATION

2.1.1.2 Functional Requirements (GE: E. Kujawski)

Analysis is in progress to determine the adequacy of the proposed SASS to meet LOA-2 success criteria for beyond design basis events. Since SASS is presently considered as a tertiary system, the proposed design for the CDS Phase II core would consist of three fully enriched control assemblies which replace three of the six primary control assemblies in the middle row. Calculations have been performed to determine the worth of the SASS for several configurations of interest. The worth of three fully inserted SASS rods is 3.15\$ at EOEC. The worth of two inserted SASS rods at EOEC is 2.02\$. The worth of two SASS rods at BOEC is 1.64\$ if all 12 outer primaries are inserted (to criticality), and 1.15\$ if the outer primary adjacent to the third (stuck) SASS rod is also stuck in the withdrawn position.

The bounding events for SASS have been identified, and the analysis of these transients has been initiated.

The SSE transient reactivity worths have been quantified for the CDS Phase II core. The SSE is to include the reactivity feedbacks of both stick-slip and relative core/control rod motions.

2.1.1.2 Thermionic Sensor (GE: C. Shires)

The adequacy of a thermionic sensor was evaluated and was deemed feasible. The physical parameters for a prototype have been determined. Future development and test requirements have been established.

2.1.1.2 Position Sensor (GE: C. Shires)

An active inductive position sensor network has been tested on the SASS Facility and determined to be feasible. Coil forms for future prototypic development have been procured and are being readied for winding. The bread-board inductive bridge circuits are being converted to the new coil parameters.

2.1.1.2 Alternate Concepts (GE: L. F. Peterson)

Several alternate SASS concepts have been created utilizing a thermionic switching element. Conceptual layouts for two of these, the Hydrostatic Supported Absorber (HSA) and the Thermionic Switched Electromagnetic Latch (TSEL), have been established. These are being given further consideration. In the HSA, the absorber element is retained in the normal position via a static pressure difference while in the TSEL the absorber element is up held by an electromagnet. In both concepts, the absorbers are released through thermionic switch actions. These concepts require through-the-head retrieval

of the absorber, and preliminary comparison of requirements for retrieval revealed no significant design problems.

2.1.1.2 Water Testing (GE: W. A. Brummond, D. M. Barrus)

Water testing of the axisymmetric (3D) and the flat (2D) cross-section diverter valves is continuing. The (3D) diverter testing revealed the need to reduce the required switching pressure. A new center body which has a pilot valve was constructed for the (3D) valve. Several pilot valve designs were tried, which successfully reduced the control pressure required to switch the diverter. This testing has led to the design of another diverter (3D-2) with adjustable internals. The (3D-2) diverter has not yet been tested. The flat (2D) diverter is a cross-section of the (3D) diverter. The (2D) has a transparent front to allow observation of the fluid flow paths. The (2D) diverter was successfully operated, demonstrating switching control and diverter switching stability. Still and moving, photos, were made of the (2D) diverter using a stream of gas bubbles to enhance the observation of the water streams. The results of the (2D) diverter tests lead to the design of a second plant diverter (2D-2) with a transparent cover and adjustable internals. The (2D-2) diverter has been installed in the test loop and has undergone some initial testing.

2.1.1.2 Thermal-Hydraulics Analysis (GE: J. P. Wei)

Conceptual thermal-hydraulics analysis on SASS absorber assembly was initiated. In order to meet the flow requirements of levitating the absorber assembly and cooling of the absorber pin bundle. Two design options were considered:

- a. absorber bundle with top plate and no lower seal ring,
- b. absorber bundle with both top plate and lower seal ring.

An analytical technique was developed to predict the levitating force on the absorber assembly and flow split between the pin bundle flow and the bypass flow in the annulus between the pin bundle and guide tube. The following conclusions were drawn from the analysis:

- With lower seal ring O.D. ranging from 5.0 to 5.15 inches, the pressure drop across the absorber assembly has an insignificant increase over that without the lower seal ring.
- For lower seal ring O.D. of 5.10", the bundle would receive near 50% of the total flow.

2.2 SHUTDOWN HEAT REMOVAL SYSTEM FAULT ACCOMMODATION

2.2.2 Coolability of Core

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

A simple case to test the capability of the THERMIT code for analyzing natural convection was conducted. The results appear reasonable. A request to MIT has been made for the latest version of the code. Work on the THERMIT model using the old version to determine in-vessel natural circulation flow patterns continued.

The conceptual design of a water test loop has been internally reviewed and approved. The test is to simulate the shutdown thermal-hydraulic condition of the LMFBR core and the associated plenum. With an emphasis on flow visualization, the test is aiming at providing information on hydraulic characteristics of buoyancy induced flow redistribution. A simpler lead test loop was built to study flow meter performance and flow visualization technique. The information is being used to guide the primary test loop design.

As a parallel effort to the experimental program, a simple computer program that adopts the flow redistribution model of the SSC code has been developed. Some preliminary calculations indicate that a significantly larger fraction of the flow is drawn in the higher power channels. The code will also be used to guide the final test loop design and operation.

2.2.2 Coolability of Core

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

The analysis of the anticipated loss of pump power accident on a representative CDS core has been completed. The analysis was done using the SAS3D computer code. The major conclusions are:

1. The core mixed mean outlet temperature reaches 1000°F 27 seconds after the accident initiation and the decays;
2. The core hot channel outlet temperature reaches 1092°F 23 seconds after accident initiation and the decays;
3. The core flow split remains relatively constant during this transient.

The analysis of the possible existence of natural circulation loops within a subassembly following a reactor scram and loss of pump power was completed. A model of a fuel subassembly was developed and analyzed with the THERMIT computer code. For the range of conditions studied no evidence of flow recirculation was found, either during the transition to natural circulation or immediately following. However, due to uncertainties in the study (such as the selection of the transient subassembly boundary conditions) it cannot be definitively stated that the study disproved the existence of flow recirculation loops within a subassembly.

2.2.2 Coolability of Core

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

Steady state heat balances have been used to define the fuel bundle power during the SLSF W-1 tests. These calculations show a variation in the power at pre-LOPI conditions from 651 KW for LOPI 3 to 715 KW for LOPI 4. The powers for LOPI 1 and LOPI 2A were very near the nominal value of 668 KW. The power history throughout the tests has been defined for LIFE-4 analysis of fuel condition.

Fuel pin power distributions have been defined for thermal-hydraulic post-test analyses. These include axial distributions with peak-to-average ratios of approximately 1.3, radial variations with as much as a 20% power depression across an individual pin, and pin-to-pin variations of +3% to -2% about the average.

LOA-3: MAINTAIN CONTAINMENT INTEGRITY

3.1.1 Accommodation Within Primary System Boundary

In-Pile Experiments: TREAT

GE PFR/TREAT Test Program Planning (GE: D. B. Atcheson)

During this quarter, analysis and planning in support of the PFR/TREAT Test Program continued, with emphasis on developing an overall TREAT Test Plan that includes the PFR/TREAT Program as one element. The primary GE responsibility in this multi-contractor effort was to identify the key technical safety issues that must be resolved to assure containment integrity during hypothetical transient overpower (TOP) accident scenarios. Once the key technical issues were identified, the objective was to indicate how each issue could best be resolved. In general, each issue is to be resolved via a combination of TREAT experiments, other in-pile experiments, out-of-reactor experiments, detailed post-irradiation examination, and analysis. The relative contribution of each of these elements varies significantly between issues.

As a first step in developing an overall TREAT Test Plan, the top-level Work Breakdown Structure issues in the LMFBR Safety Program Plan were divided into sets of sub-issues (i.e., questions that need to be answered in order to resolve the top-level question). In most cases, two or three levels of sub-issues were needed in the TOP area to make the technical issues specific enough for experiment planning. This hierarchy of transient overpower issues was reviewed and approved by the TREAT Program Planning Task Force during a June 1980 meeting held at Argonne. Top-level issues were assigned priorities at this meeting, and the detailed issues most likely to be resolved by TREAT testing were tentatively identified.

One of the four top-level (technical issues affecting the transient overpower accident sequence is the axial location of initial molten fuel ejection from each fuel pin). In order to resolve this issue, one must demonstrate understanding of the axial location of initial (and subsequently) cladding breach. Analyses performed at GE-ARSD using the BEHAVE-SST and LIFE-4 codes, combined with analyses performed at HEDL and ANL, suggest that fuel and cladding behavior are significantly different in the low ramp rate regime (3¢/sec to 10¢/sec) than at the high ramp rates tested previously (50¢/sec and 3\$/sec). The analyses also suggest the possibility of earlier cladding failure at high burnup, as fuel-cladding gap volume is converted into less-available fuel crack volume, central void volume, and fuel porosity. Therefore, it seems clear that an important element of the test program should be experiments performed at high burnup and low ramp rate. However, the GE analyses suggest that fuel pin geometry (as-fabricated gap size, pellet density, and pellet diameter) may be equally important. This is especially true when advanced oxide fuel designs and heterogeneous cores are considered to be the "designs of the future."

The GE analyses indicate that fuel or blanket pins with small as-fabricated gap sizes or low initial pellet densities are susceptible to early cladding failures at relatively low axial locations. This tendency is most pronounced at high burnups at all transient ramp rates, and is predicted to be accentuated by low-swelling cladding materials. The mechanism causing early failure is the radial redistribution of void volume from the gap to the fuel. Some fraction of the gap volume is predicted to be trapped near the outer fuel surface in the form of porosity, and the absolute magnitude of this trapped void volume increases as initial gap size increases. Since void volume contained within cold outer fuel is relatively unavailable to accommodate fuel expansion, the likelihood (and magnitude) of cladding inelastic deformation increases as this type of void volume increases. This, in turn, is predicted to lead to early cladding breach, especially when fuel void volume is reduced by solid-fission-product fuel swelling at high burnup.

The analyses described above are supported by the HOPI-6A experiment in TREAT (3¢/sec, 110 MWD/kg), in which three pins with very different pellet geometries were tested. One pin of the three experienced cladding breach very early in the transient, while the other two pins did not fail and showed no measurable permanent cladding strain. Similar pins, tested at 3\$/sec in the HUT3-6A and HUT-6B TREAT experiments, also failed quite early in their transients. Therefore, it appears that fuel pellet geometry and burnup are of similar importance, and the effect of transient ramp rate is unclear.

As a result of the analyses described above, a major GE recommendation for the TREAT Test Program Plan is that fuel fabrication tolerances and pellet design variables be considered as important independent variables in the program. It is believed that these data will directly support CDS fuel and inner blanket pin design and safety evaluations, as well as complementing the Reference Oxide and PFR/TREAT data bases.

3.2.2 Accommodation Within Inner Cells

Thermal Analysis of an Improved Reactor Cell System

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

An improved reactor cell system (GE: K. H. Chen and E. L. Gluekler, "An Improved Reactor Cell Concept and the Evaluation of its Long Term Retention Capability." US-DOE - Fast Reactor Safety Program Progress Report, January - March, 1980. ANL/TMC 80-2) was evaluated using the GROWS-2 code modified to account for the various thermal challenges to the reactor cell system. The response of the overall containment system with the improved reactor cell design was evaluated with the CACECO code.

The modified GROWS-2 model provides a time dependent heat sink temperature for use in determining the upward transport of the debris pool decay power into the reactor cell. The model considers thermal radiation heat transfer between the reactor cell walls after sodium pool boiling. This modified version of the GROWS-2 code can be considered as a reactor cell system thermal model. At present, up to five composite regions can be analyzed. This provides sufficient insight into the effectiveness and feasibility of applying various liner materials such as MgO or graphite, and insulating materials for concrete structure protection such as insulating firebrick or cement.

A parametric study was performed to determine the effects of the dimensions of MgO bed and the presence of an insulating firebrick layer below the MgO on the improved reactor cell system for a large LBR. The results of the study showed that the debris pool growth is sensitive to debris dispersion. For the 2500 MWe LBR considered, an initial debris pool diameter of about 13m would be permanently frozen on the MgO bed. For a deeper initial debris pool (smaller initial diameter), the growth characteristics depended on the cavity bed material and were not influenced by the sodium inventory prior to sodium boiling. Long term evaluations (up to 120 days) showed that it might not be feasible to contain penetration of core debris without using a cooling system. A layer of insulating firebrick between MgO and concrete base mat reduced downward penetration, but it might not contain the long term pool growth. The thermal analysis of the reactor cell system showed that the composite cell wall design provided sufficient protection for the concrete wall structure, but the reactor head ledge concrete would fail early under thermal load unless enough reinforcing steel was used to carry the load or a cooling system was applied. The overall containment response for the improved reactor cell design was evaluated using the CACECO code. The results indicated pressure and temperature increase in the reactor containment building (RCB) was very small within a 48 hour period. Water vapor release was small and all was collected in the water cell with no vapor transport to the RCB. The composite reactor cell design met a hold up period of over 48 hours.

Evaluation of Concrete Water Release Models (GE: K. H. Chen,
E. L. Gluekler, S. T. Lam)

The three mechanistic codes COWAR-2, WATRE and USINT were evaluated. The respective physical models and numerical methods were reviewed; the predictions were compared with the results of Water Release Demonstration tests No. 1 and No. 2 conducted by Hanford Engineering Development Laboratory for the full range of the experiment periods (30 hrs. and 50 hrs.); and the code handling case and costs were assessed. The USINT code was recommended as the reference code for its good agreement with test results, efficient computational scheme, and more complete stage of development. A detailed evaluation is being separately reported (K. H. Chen et al., "Comparison of Mechanistic Codes for Predicting Water Release from Heated Concrete," GEFR-000521 (DR), April 30, 1980).

LOA-4: ATTENUATE RADIOLOGICAL CONSEQUENCES

4.2.1A Radiological Consequences Assessment Methodology (GE: J. D. Teresi)

The specific measures of radiological consequences that may be used as criteria in reactor safety and design option evaluations have been identified. Although the quality and quantity of radionuclides released from containment have been used as measures of expected radiological consequences in some limited studies, the measures generally used are potential doses (whole body and organ), health effects (such as early fatalities, various organ cancers, thyroid nodules and genetic effects), and total potential risks (probability of health effects and property damage).

The objectives of the current study are to describe the various measures of radiological consequences and to present examples of their application. Preliminary results indicate that the potential dose to bone, lung, thyroid or whole body at distances of about 0.5 mile to 5 miles from the reactor is a sufficient measure for studies of 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.

Examples of the use of potential doses as measures of radiological consequences are shown in Figures 1 and 2 (T. C. Huang, "Review and Development of Aerosol Analytical Methods," GEAP-14147, September 1976). Figure 1 shows part of the results of a sensitivity study of two aerosol parameters. These parameters are the Stokes correction factor (α) and the gravitational collision efficiency (ϵ). Figure 2 shows the effect of warning period before venting for various melt through scenarios.

FIGURE 1

2-Hour Site Boundary Doses Versus ϵ

FIGURE 2

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 6.1-2, LBR Risk Allocation Model Development.

The development of a Risk Allocation Model for LBR design application was completed. A demonstration risk allocation analysis was performed using cost and system reliability estimates based on CDS design information. The model development and the demonstration analysis have been documented in the topical report, "Thirteenth Quarterly Report - Risk Analysis Methods Development, April-June 1980" (GEFR-14023-13), currently being issued.

The safety related systems design options considered in the Risk Allocation Demonstration are shown in Table 6-1. The design options selected by the Risk Allocation to satisfy LWR comparable risk limits at minimum cost are also indicated in Table 6-1.

The demonstration analysis performed using the Allocation Model, quantified with judgemental estimates on system reliabilities and costs, yielded the following results:

1. The designated risk limits, based on comparable risks for LWR plants, are satisfied at minimum cost by employing two of the identified higher cost options (vented core coolant ducts and pump trip inhibit circuit).
2. The coolable free-standing steel containment option provides the most risk reduction per dollar spent, considering capital costs, reliability improvement costs and R&D costs. The SASS (Self-Actuated Shutdown System) with the vented coolant duct design and improved failed fuel detection system reliability are the next most cost-effective design options.
3. Sensitivity analysis was performed by individually varying cost and reliability estimates associated with the allocated variables over approximately a $\pm 2\sigma$ uncertainty range. Optimal design vectors were calculated for 155 uncertainty bound conditions. Only eleven of these optimal vectors differed from the nominal value results.

Work now in progress will review, expand and update the risk model element of the Risk Allocation Model to provide a current LBR risk assessment tool.

TABLE 6-1

RISK ALLOCATION DESIGN OPTION SELECTION (OPTIMUM CASE)

<u>Safety Function</u>	<u>Design Variable</u>	<u>Design Option</u>	<u>Design Option Selected</u>	<u>Incremental Cost, \$</u>
Initiating Event Prevention	Duct Design	1. Closed 2. Vented	X	4.1×10^6
	Fuel Failure Detection	1. Current Reliability 2. Improved Reliability	X	
Accident Prevention	Primary & Secondary Shutdown System	1. Current Reliability Level 2. Improved Re- liability	X	
	Pump Trip Inhibit Circuit	1. Without Circuit 2. With Circuit	X	3.7×10^6
Accident Termination	Self Actuated Shutdown System	1. Without SASS 2. With SASS	X	
Hazardous Material Containment	In-Vessel Debris Retention	1. Without Debris Retention Device 2. With Debris Retention Device	X	
	Cell Liners	1. Without MgO Cavity Liner 2. With MgO Cavity Liner	X	
	Containment Structure	1. Reinforced Concrete 2. Free Standing Steel with Cooling	X	

DISTRIBUTION OF FUNDING AMONG GE SAFETY BASE PROGRAM TASKS

(CONTINUED)

<u>LOA-2</u>	<u>% OF FY-80 FUNDING</u>	<u>% OF FY-81 FUNDING (PREL.)</u>
● SELF-ACTUATED SHUTDOWN SYSTEM (SASS)	16	14
● SHRS-REQUMT'S., IN-VESSEL FLOW PATHS EVAL., NAT. CIRC. TESTS AND CODES EVAL.	9	15
● CORE ACCIDENT ANALYSES/INHERENT SHUTDOWN	2	2
	<u>27</u>	<u>31</u>
<u>LOA-3</u>		
● FUEL SAFETY PROGRAM - PFR/TREAT, FUEL PIN FAILURE MODELING, INTERNAL BLANKETS, W1 TEST ANALYSIS	11	8
● IN-VESSEL DEBRIS RETENTION	2	4
● EX-VESSEL DEBRIS RETENTION AND CONTAINMENT MODEL DEVELOPMENT	5	4
	<u>18</u>	<u>16</u>
<u>LOA-4</u>		
● SPARGING EXPERIMENT DESIGN	≤ 1	1