ACRS - 1741

CERTIFIED

6/27/80

DATE ISSUED: 6/27/80

MEETING MINUTES OF THE ACRS REACTOR FUEL SUBCOMMITTEE MEETING
APRIL 29, 1980
WASHINGTON, DC

On April 29, 1980 the ACRS Reactor Fuel Subcommittee met in Washington, DC, to begin discussion of the NRC Research Program in the area of fuel behavior for the Committee's annual reports to the Commission and Congress. The notice of the meeting appeared in the Federal Register on April 14, and April 25, 1980. There were no requests for oral or written statements from members of the the public and none were made at the meeting. Attachment A is a copy of the meeting agenda. The attendees' list is Attachment B. Attachment C is a tentative schedule of presentations for the meeting. Selected slides and handouts for the meeting are Attachment D to these minutes. A complete set of slides and handouts is attached to the office copy of these minutes.

OPEN SESSION - INTRODUCTION

Dr. Shewmon, Subcommittee Chairman, called the meeting to order at 8:30 a.m. The Chairman explained the purpose of the meeting and the procedures for conducting the meeting, pointing out that Mr. Paul Boehnert was the Designated Federal Employee in attendance. Dr. Shewmon introduced Dr. William Johnston, Chief, Fuel Behavior Research Branch (FBRB) to begin the day's presentations.

Dr. Johnston begin by discussing the priorities, the objectives, and summarized the FBRB programs. He noted that the overall objectives of the fuel program are to evaluate fission product and fuel behavior under normal and accident conditions, develop physical models through laboratory scale separate effects tests, verify fuel codes and models through integrated tests, and utilize models and codes to assess the consequences of severe reactor accidents including core melt events and to aid in the design and evaluation of mitigation features. Dr. Okrent asked a number of questions centering on the objectives of the FBRB programs. Following a lengthy discussion, Dr. Okrent asked that

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during the FBRB presentations, NRC discuss how the program results relate to the defined objectives of the given program. During a discussion of power coolant mismatch (PCM) tests run at PBF, Dr. Okrent said he believed these tests have not addressed the issues of concern for that type of accident.

Dr. Johnston noted that since the TMI-2 accident, the priorities of the FBRB Program have been rearranged to the following order: (1) core damage beyond LOCA, (2) cladding ballooning and blockage, (3) fission product release and migration, (4) operational transients (Classes I-III), and (5) fuel meltdown. Dr. Okrent observed that the Research Program does not factor in risk reduction and he felt that it should. Dr. Murley replied that the NRC does not license plants based on risk analysis, rather technical judgment is used.

CLOSED SESSION (9:30 - 10:20 am), FY-81, FY-82 FBRB BUDGET - W. JOHNSTON

In a closed executive session, the FY-82 FBRB budget was discussed. Dr. Murley observed that the budget situation is very uncertain since Congress has not acted on either the FY-80 Supplemental request nor the FY-81 budget. He also noted that a budget cut is expected, however the amount at this time is unknown. The FY-81 and proposed FY-82 budget for the various programs under the five priorities noted above were detailed (Figures D1-4). In general, it was noted that the budget is increasing in the new research areas, such as core damage beyond LOCA, and decreasing in older program areas (cladding, ballooning and blockage). In response to questions from Dr. Okrent, Dr. Murley noted that the FBRB programs are in a state of transition with emphasis shifting to the priority ranking noted above. Dr. Okrent agreed that this was the case, but questioned whether or not the pace of the shifting emphasis should be accelerated.

FUEL CODE DEVELOPMENT AND VERIFICATION - G. MARINO

Dr. G. Marino discussed the fuel code development and evaluation programs. He noted that the objectives of these programs are to predict transient and steady state fuel behavior under normal, off-normal, and accident conditions and to provide an integrated, easily accessible storage bank of fuel behavior information. The codes used include MATPRO - the material property correlations, FRAPCON - a steady-state code, used to develop fuel behavior under normal conditions, and FRAP-T - a transient code to simulate fuel rod behavior for a transient situation.

Dr. Shewmon asked when Research will feel that code development can end. He noted that he believes in many instances NRC codes are better than what the Industry has, and that the Industry is probably using NRC codes for their license evaluations. Dr. Marino acknowledged this may be the case, but noted that code development can never stop, and that the codes will be continuously updated as new information becomes available. He noted that FRAPCON will stop at Version II and that the FRAP-T code will stop at the T-6 version. From then on, the codes will be in a maintenance status to be updated as new information becomes available.

The FBRB Code Verification Program was described. This program involves both developmental verification, and independent verification. An example of the results of the assessment program was shown (Figures D5-6). In response to questions from Drs. Shewmon and Okrent on the Code Assessment Program, Dr. Marino noted that the assessment program is showing the limits of the code's predictability, given the uncertainties in data input.

Dr. Marino reviewed the expected fuel code accomplishments for FY-80/81 as well as developments expected beyond FY-81 (Figures D7-9). Work planned beyond FY-81 includes development of a small-break (slow transient) fuel rod damage code based on, and linkable to, FRAP-T and FRAPCON. FBRB requested Subcommittee opinions on how far into the core melting sequence the code should go. Dr. Shewmon expressed primary concern about coolability versus failure modes. Dr. Okrent expressed skeptism that the above LOCA code will be able to model core coolability for severe damage cases.

FUEL PELLET AND FUEL ROD PROPERTIES RESEARCH - G. MARINO

Dr Marino noted that the objectives of the Fuel Pellet and Fuel Rod Properties research programs are to provide information on changes to fuel pellets during steady-state and transient operation, improve models for calculating gap conductance in a fuel rod, and determine the extent to which fuel pellets affect the transient axial flow of gas within a fuel rod. The bulk of the program is centered on a series of instrumented fuel assembly (IFA) tests being conducted at the Halden reactor in Norway. Figures D10-14 detail the Halden tests and results to date.

Another section of the program deals with the development of a transient fission gas release code (GRASS, FASTGRASS). The GRASS and fast-running version (FASTGRASS) are under development at Argonne. Comparisons of the two codes predictions versus results of spent fuel pellet direct electical heating tests to determine fission gas release rates, show fairly good agreement (Figures D15-16). Dr. Marino noted that work on an even faster running GRASS code version (PARAGRASS) has been initiated. The FASTGRASS code, and when complete the PARAGRASS code, will be incorporated into the FRAPCON and FRAP-T fuel codes.

ZIRCALOY CLADDING RESEARCH - M. PICKLESIMER - NRC

Dr. Picklesimer discussed the zircaloy cladding research programs now underway. These programs include the Multi-Rod Burst Test (MRBT) program conducted at Oak Ridge, the study of the Mechanical Properties of Zircaloy at Argonne, a study of the strength and conductivity of the irradiated zircaloy using PWR spent fuel cladding at Battelle Columbus, and a study of zircaloy cladding creepdown in-pile via a joint program between ORNL and ECN PETON in the Netherlands. Highlights of the above programs include the following:

- The MRBT program is designed to characterize ballooning, burst, and loss of flow area in bundles of LWR fuel rod simulators during the refill-reflood stage of a LOCA. A number of single-rod and 4 x 4 bundle tests have been run; an 8 x 8 rod bundle test is scheduled for early June. Recent results (Figure D-17) indicate that single rod tests may be used to successfully model bundle behavior. It is hoped that if the results of the 8 x 8 rod bundle tests duplicate the 4 x 4 bundle results, a scaling factor will be able to be developed.

• The Mechanical Properties of Zircaloy program designed to quantitatively characterize zircaloy clad embrittlement by oxidation with steam, and relate this embrittlement to measureable mechanical properties of the embrittled material. The second phase of the program is designed to determine the stress-rupture properties of spent LWR cladding under simulated PCI conditions leading to clad rupture. The results of the recently completed first program phase are embodied in a set of failure limits for embrittled cladding for the cases of thermal-shock and impact. For the thermal-shock case, it is proposed that at least 0.1 millimeter of clad wall thickness remain with less than 0.9 weight percent oxygen. For the impact criterion, no less than 0.3 millimeter cladding wall thickness contain less than 0.7 weight percent oxygen.

• Dr. Picklesimer discussed the planned research program on PCI. Elements of the program will include the study of PCI failure by stress rupture, strain-rate ramping to PCI failure, both ex-pile and in-pile. The in-pile tests will be conducted at the Studsvik and PBF reactors (Figures D18-21). Dr. Okrent questioned whether or not a study of PCI was more appropriate for DOE or EPRI. Dr. R. Meyer (NRR) replied that industry does not believe that PCI is a safety concern, while the NRC does.

FBRB FUEL MELT PROGRAMS - R. SHERRY - NRC

Mr. Sherry discussed the LWR fission product release and transport programs both underway and planned by FBRB. The overall objectives of this program are to develop fission product source terms for fuel rods under accident conditions including severe core damage and core melt, to develop models to predict the attenuation and transport behavior of fission products within the primary coolant system and containment, and to provide environmental release source terms for consequence analysis and evaluation of engineered safety features and mitigation feature design requirements.

There are five existing programs and two planned programs in the fission product release and transport research. The five ongoing programs are: (1) fission product transport analysis - TRAP code (Figure D-22), (2) separate effects tests for the TRAP code (Figure D-23), (3) fission product vapor deposition requirements (Figure D-24), (4) steam generator tube rupture iodine transport (Figure D-25), and (5) fission product release from LWR fuel (Figure D-26). The two planned programs include a study of fission product release from LWR fuel at high temperature (Figures D27-28), and charcoal filter iodine retention performance studies (Figure D-29). Commenting on the last program, Dr. Lawroski said he believes such a program should be conducted by DOE or the Nuclear Industry, not NRC.

Mr. Sherry also described four future programs that are under consideration. These programs are: (1) fission product release from melting fuel (Figure D-30), (2) fission product leaching (Figure D-31), (3) development of a fission product transport verification facility (Figure D-32), and (4) examination of TMI fission product release data (Figure D-33). Concerning the first program noted above, the study of fission product release from melting fuel, Dr. Okrent suggested FBRB try to determine if a "quick and dirty" experiment would be more productive in order to get a yard stick on the estimates for core melt releases described in WASH-1400.

SEVERE CORE DAMAGE STUDIES - M. PICKLESIMER - NRC

Dr. Picklesimer briefly reviewed the new proposed program to study severe core damage. This program would focus on the following areas: (1) development of core damage, (2) fission product distribution, (3) modeling of severe core damage, and (4) code development for prediction of core damage. Research areas in core damage would include integral bundle effects, and separate effects and basic studies, all both in- and out-of-pile. Also to be included would be a study of fission product release and distribution in the primary system, and the modeling for prediction of core damage. In-pile tests would be conducted in the PBF or ESSOR test reactors, and the planned TMI-2 core examinations would be relied upon to provide related information. Some of the above programs will begin in FY 81 with the majority of them scheduled to begin in FY 82.

Dr. Okrent suggested that the extensive debris bed experiments conducted in the fast reactor program may provide relevant information in lieu of tests in PBF. The NRC Research noted that the differences between the fast reactor and LWR systems preclude significant transfer of knowledge from one group of experiments to the other. In response to a question from Dr. Shewmon, Dr. Picklesimer noted that the tests planned for ESSOR would not begin until late 1982 or early 1983.

INTEGRATED CORE MELT RESEARCH PROGRAM - C. KELBER - NRC

Dr. Kelber briefly discussed the program logic for the Integrated Core Melt Program recently formed in NRC Research. The program objective is to determine the best-estimate of risk, analysis, and assessment of special features designed to assure containment integrity in the event of a core melt. To this end, Dr. Kelber said a number of significant questions must be answered (Figure D-34). ACRS review of the integrated core melt program will be conducted by the new Class 9 Subcommittee which has scheduled a meeting for May 9, 1980.

TMI-2 CORE STATUS STUDIES - M. PICKLESIMER

Dr. M. Picklesimer discussed the TMI-2 core status studies conducted as part of the Special Investigation Group for the Rogovin Report. He described the core status at 3 hours into the accident when significant core damage had taken place, and at 4 hours into the accident when additonal core damage occurred. The estimates noted below are based on computer code and system analyses.

At the 3 hour mark in the accident, it was estimated that the maximum core temperature reached was about 4400^{o}F in the upper 3 feet of more than two-thirds of the core, and a temperature of 3600^{o}F was reached for all of the core down 5 feet from the top of the fuel assemblies. A 2 foot thick debris bed was probably formed with a base at about 8 feet from the bottom of the core over the entire core, and its formation was aided by the thermal shock of embrittled cladding and "liquified fuel" at 2 hours and 54 minutes in the transient when the 2B reactor coolant pump was started.

At 4 hours into the transient, more than 60% of the zircaloy in the core had been embrittled or shattered, and the lower surface of the debris bed had dropped to about 5 feet from the bottom of the core, and liquified fuel had penetrated to within 1 foot of the bottom of the core in some areas. It was also estimated that a total of 700-820 pounds of hydrogen had been produced by oxidation of zircaloy at the 4 hour mark. Dr. Picklesimer also said that the stainless steel upper end fittings must have suffered additional damage, but the degree of damage cannot be estimated at this time. In response to a question from Dr. Shewmon, as to how hard it would be to pull the fuel from the core, Dr. Picklesimer replied that the current thinking it that the outer fuel bundles will be able to be removed, then it would be possible to work from the outside of the core in to the center, if the core barrel has not shifted its position.

HYDROGEN AND POST-ACCIDENT COOLANT CHEMISTRY RESEARCH PROGRAMS - D. HOATSON - NRC

Mr. Don Hoatson discussed the hydrogen and post-accident coolant chemistry research programs. These programs are new, and have grown out of the concerns raised by the TMI-2 accident. The programs include a study of combustible gas generation in containment, a study of hydrogen, and a post-accident coolant chemistry program. The Combustible Gas in Containment Program is primarily concerned with the formation of hydrogen from the large quantity of zinc found in containments and containment-related structures. This program grew out of the User's Need letter from NRR.

The Hydrogen Program will study such topics as: (1) radiolysis of reactor solutions, (2) sampling an analysis of hydrogen in reactor emergencies, (3) flammability and detonation limits under accident conditions, and (4) handling of post-accident hydrogen. FBRB is developing a compendium of information on hydrogen, focusing on the topics above and to be used in appropriate emergencies. For example, it was noted that to mitigate the effects of hydrogen in containment, a water fog system is under study. A draft of the compendium is scheduled to be issued in May 1980, with the final report issued in June 1980.

The post-accident coolant chemistry program will address the areas of fission product signatures from failed fuel, a study of iodine in containment, and a follow-on of the radiolysis work from the hydrogen program noted above. Mr. Hoatson noted that both the hydrogen and the post-accident coolant chemistry programs are scheduled to obtain money from the FY-80 supplement. In response to a question from Dr. Okrent, Dr. Johnston said that had he known how uncertain the supplemental appropriation would be, he would have requested funding of the hydrogen program be taken directly from his normal budget.

During a Subcommittee discussion, Dr. Okrent observed that he felt that experiments scheduled for facilities such as PBF must be judiciously planned and carefully thought out in order to obtain maximum benefit for the large amount of funds being expended. Dr. Mark observed that Research needs to sort experiments between what is useful to NRC, and what is of more use to the Nuclear Industry. He feels that such work as a study of degraded radioisotope filters may not be all that necessary a research topic. Dr. Johnston replied that he agrees with Dr. Okrent that any experiments run in PBF for example, encounter difficulties because of such constraints as length effects, and both NRC and the PBF program representatives are cognizant of these problems and take steps to minimize their impact. Dr. Johnston also said that the work on degraded filters was based on a request from NRR for information in this area.

NRC-NRR CORE PERFORMANCE BRANCH PRESENTATIONS - R. MEYER, D. HOUSTON

Dr. Ralph Meyer and Mr. Dean Houston from the Core Performance Branch of NRR discussed the CPBs' Technical Assistance Program and information pertaining to fuel failures in operating LWRs. Dr. Meyer began by noting that with the reorganization of NRR, the effort in the fuel area has been reduced in man-power from 11 to 4. He also noted that NRR is reconsidering its priority on the RIA, since recent information has indicated that the accident consequences may be much less severe than previously believed. Dr. Meyer then reviewed the CPB effort in technical assistance, noting that the total budget in technical assistance is about $380,000 for FY-80 (Figure D-35). During discussion of the technical assistance program that is addressing DNB-induced fuel failure propagation, Dr. Okrent questioned how NRR would make use of the data available to answer questions in this area. Dr. Meyer responded that he would prepare a memo to address Dr. Okrent's concern and forward it to the ACRS.

Mr. Dean Houston reviewed the fuel failure history for operating LWRs for the past year. He noted that, excluding the TMI-2 failures, there have been 116 assemblies that have experienced some degree of fuel failure (Figure D-36). Mr. Houston detailed the various fuel failure mechanisms (Figure D-37). Again, excluding the TMI-2 failures, failures were seen due to such mechanisms as water-side corrosion, stress corrosion cracking (in stainless steel fuel), PCI, handling accidents, and vibration and fretting.

The meeting was adjourned at 5:05 p.m.

Note: Additional meeting details can be obtained from a transcript located in the NRC Public Document Room, at 1717 H St., N.W., Washington, D.C., or can be obtained from International Verbatim Reporters, Inc., 499 South Capitol Street, S.W., Suite 107, Washington, D.C. 20002.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Reactor Fus; Meeting

The ACRS Subcommittee on Reactor Fuel will hold a meeting on April 29, 1980 in Room 1046, 1717 H St., NW., Washington, DC 20555 to discuss the NRC research program on reactor fuels for the ACRS annual reports to NRC and Congress. Notice of this meeting was published March 19, 1980.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: *Tuesday, April 29, 1980—8:30 a.m. until the conclusion of business.*

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, their consultants, and other interesed persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92–463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Paul A. Boehnert (telephone 202/634–3267) between 8:15 a.m. and 5:00 p.m., EST.

Dated: April 9, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-11122 Filed 4-11-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Subcommittee on Reactor Fuel; Meeting Addition

It may be necessary for the ACRS Subcommittee on Reactor Fuel to close portions of its meeting on April 29, 1980 for the following reason:

The ACRS is required by Section 5 of the 1978 NRC Authorization Act to review the NRC research program and budget and to report the results of the review to Congress. In order to perform this review, the ACRS must be able to engage in frank discussions with members of the NRC Staff and such discussions wodld not be possible if held in public sessions. I have determined, therefore, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Public Law 92–463), that, should such sessions be required, it is necessary to close portions of this meeting to prevent frustration of the above stated aspect of the ACRS' statutory responsibilities. See 5 U.S.C. 552b(c)(9)(B).

All other items remain the same as announced in a notice published April 14, (45 FR 25195).

Dated: April 22, 1980.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 80-12[illegible] Filed 4-24-80; 8:45 am]

BILLING CODE 7590-01-M

Attachment A

ACRS
REACTOR FUEL SUBCOMMITTEE MEETING
APRIL 29, 1980
WASHINGTON, DC

ATTENDEES LIST

ACRS

P. Shewmon, Chairman
S. Lawroski, Member
J. C. Mark, Member
D. Okrent, Member
A. Bement, Consultant
F. Nichols, Consultant
P. Boehnert, Staff*

*Designated Federal Employee

EG&G

H. I. Zeile
J. E. Hanson
C. R. Toole
P. E. MacDough

NRC

W. V. Johnston, RES
M. L. Picklesimer, RES
G. P. Marino, RES
D. A. Hoatson, RES/FBRB
R. Meyer, NRR/CPB
C. Kelber, RES

EPRI

R. Leyse

Attachment B

ARPIL 29, 1980
WASHINGTON, DC

- Tentative Schedule of Presentations -

	Presentation* Time	Actual Time
I. Introduction	10 min	8:30 am
P. Shewmon, Chairman		
II. NRC Fuel Behavior Research Branch (FBRB) Presentations		
A. Priorities, Objectives, and Summary of FBRB Program	20 min	8:45 am
W. Johnston, Chief, FBRB		
B. FBRB Budget (Closed Session)	20 min	9:15 am
W. Johnston		
• FY 81-82 Budget		
• Status of Budget Supplement Request		
- Break -	10 min	9:45 am
C. Fuel Code Development and Verification	35 min	9:55 am
G. Marino		
D. Fuel and Cladding Programs		
• Fuel Programs - G. Marino	20 min	10:40 am
• Cladding Programs - M. Picklesimer	20 min	11:15 am
- Cladding Swelling and Rupture - PCI - Incipient Fuel Melt		
E. Fuel Melt Program	30 min	11:45 am
• TRAP Code and Related Studies - R. Sherry		
• Severe Core Damage Study -M. Picklesimer		
- Lunch -	60 min	12:30 pm

*Additional time has been allotted for Subcommittee Questions

ATTACHMENT C

- Tentative Schedule of Presentations -

	Presentation* Time	Actual Time
F. Comments on Integrated Core Melt Program and Related Studies	15 min	1:30 pm
C. Kelber		
G. TMI-2 Core Status	45 min	1:50 pm
W. Johnston M. Picklesimer		
H. TMI-2 Research Programs	30 min	2:50 pm
D. Hoatson		
• Hydrogen Programs		
• Coolant Chemistry Programs		
I. Summary	10 min	3:30 pm
W. Johnston		
- Break -	10 min	3:40 pm
III. NRR Core Performance Branch Presentations		
A. Technical Assistance Programs	15 min	3:50 pm
R. Meyer		
B. Discussion of Recent Fuel Failures	20 min	4:15 pm
D. Houston		
C. Regulatory Position on PCI	25 min	4:40 pm
M. Tokar		
IV. Discussion	15 min	5:15 pm
V. Adjourn		5:30 pm

*Additional time has been allotted for Subcommittee Questions

DETAIL BUDGET - FUEL BEHAVIOR

FIN #	CONTRACTOR	TITLE	FY 80	FY 81	FY 82
CORE DAMAGE BEYOND LOCA					
B7084	ANL	EXAMINATION OF TMI FUEL	(350)	500	750
B5702	PNL	CORE DEGRADATION IN ESSOR	---	1695	2250
B7100	SAN	HYDROGEN HANDBOOK AND DATA BASE	(500)	800	700
UND.	EG&G	SEVERE CORE DAMAGE - PBF	(1900)	2135	2810
B7281	---	INCIPIENT FUEL-CLAD MELTING	---	300	300
B2372	---	POST-ACCIDENT COOLANT CHEMISTRY	---	400	200
-----	---	DEBRIS COOLABILITY STUDIES	---	---	(600)
-----	---	MODELING OF SEVERE CORE DAMAGE	---	---	350
B7200	---	REACTOR CHEMISTRY	(400)	400	400
				6530	7960
	UND.	INLET FLOW BLOCKAGE TESTS	----	1000	500
CLADDING BALLOONING AND BLOCKAGE					
B2277	PNL	LOCA BUNDLE REFLOOD IN NRU	3015	1875	3050
B0120	ORNL	MULTIROD BURST TEST	960+(250)	900	505
-----	UND.	RESIDENT ENGINEER-CADARACH, FRANCE	100	155	160
A6041	EG&G	LOCA TEST IN PBF	1150	1175	937
				4105	4652

FIGURE D-1

(CONTINUED)

FIN #	CONTRACTOR	TITLE	FY 80	FY 81	FY 82
FISSION PRODUCT RELEASE AND MIGRATION					
*B6747	IN PROCUREMENT	FISSION PRODUCT TRANSPORT ANALYSIS	75	3-5MY	605*
-----	UND.	TMI FISSION PRODUCTION IN CONTAINMENT	(175)	85	200
B0127	ORNL	FISSION PRODUCT RELEASE AT HIGH TEMPERATURES	(365)	400	(575)
A2016	ANL	TRANSIENT FISSION GAS RELEASE AND MODELING	150	105	225
	NRL	IODINE FILTER AFFECTIVENESS TESTING	(110)	115	(122)
*-----	---	FISSION PRODUCT TRANSPORT VERIFICATION FACILITY	---	---	1000
*-----	---	FISSION PRODUCT RELEASE FROM MOLTEN FUEL	---	---	(700)
A1227	SAN	SEPARATE EFFECTS STUDIES FOR TRAP	150	210	---
-----	---	LEACHING OF FISSION PRODUCTS FROM FUEL	---	100	120
-----	---	MITIGATION OF LIQUID PATHWAYS RELEASES (CONTAINMENT BYPASS)	---	---	120)
				2280	347

D-2

DETAIL BUDGET - FUEL BEHAVIOR

(CONTINUED)

FIN #	CONTRACTOR	TITLE	FY 80	FY 81	FY 82
OPERATIONAL TRANSIENTS AND INITIAL CONDITIONS					
A6050	EG&G	FRAP AND FRAPCON CODE DEVELOPMENT	690	730	700
A6046	EG&G	CODE ANALYSIS AND ASSESSMENT	245	260	270
A6041	EG&G	OPERATIONAL TRANSIENTS - PBF	3000	3060	2810
B2043	PNL	EXPERIMENTAL SUPPORT AND DEVT. OF SINGLE ROD FUEL CODES	430	570	650
A2017	ANL	STRESS RUPTURE OF IRRAD. CLADDING	450	370	505
B5531	NRC HQ	HALDEN PROJECT MEMBERSHIP	477	490	535
A6041	EG&G	PCM, RIA TESTS IN PBF	2761	2880	1873
-----	UND.	MODELING OF OPERATIONAL DAMAGE TO ZIRCALOY	----	----	500
B7202	UND.	LONG BUNDLE TESTS	----	----	500
-----	UND.	RESIDENT ENGINEER - NSRR JAPAN	----	(150)	(150)
VARIOUS	EG&G	PBF OPERATION AND SUPPORT	6012	5721	5170
B6746	---	H_2 GENERATION IN CONTAINMENT	100	349	410
				15230	14723

0-3

DETAIL BUDGET - FUEL BEHAVIOR

(CONTINUED)

FIN #	CONTRACTOR	TITLE	FY 80	FY 81	FY 82
FUEL MELTDOWN*					
A1030	SAN	STEAM EXPLOSIONS	500+140	915	620
A1019	SAN	MOLTEN CORE/CONCRETE INTERACTIONS	194+(56)	210	155
-----	UND.	FUEL MELTDOWN SYSTEMS CODES	----	(200-500)	500+
-----	UND.	FUEL MELT MITIGATION FEATURES EVALUATION	----	(300)	(600)
-----	UND.	RESIDENT ENGINEER - KARLESRUHE	100	130	140

* INCLUDED IN INTEGRATED FUEL MELTDOWN PROGRAM.

0-4

FRAP-T5 STANDARD MODEL ERRORS

Output parameter	Sample (rods/pts)		Standard error $\left[\sum_{i=1}^{n} (P_i - M_i)^2/n-1\right]^{0.5}$
CHF power at known flow	30/87		0.04 kW/CC channel
CHF flow at known power	30/87		390 $kg/s\text{-}m^2$
Initial fuel centerline temperature at scram	21/32		250 K
Fuel thermal decay constant during scram	21/32		5.7 s
Equilibrium fuel centerline temperature during scram	21/32		57 K
		MATPRO	FRAIL
Cladding burst temperature at known pressure	(155/155)	160 K	94 K
Cladding burst pressure at known temperature	(61/61)	16 MPa	23 MPa
Cladding permanent hoop strain	(327/327)	32% cladding OD	33% cladding OD

FRAP-T4 STANDARD MODEL ERRORS

Output parameter	Sample (rods/pts)		Standard error $\left[\sum_{i=1}^{n} (P_i - M_i)^2/n-1\right]^{0.5}$
CHF power at known flow	18/87		0.06 kW/CC channel
CHF flow at known power	18/87		400 $kg/s\text{-}m^2$
Initial fuel centerline temperature at scram	21/32		280 K
Fuel thermal decay constant during scram	21/32		5.4 s
Equilibrium fuel centerline temperature during scram	21/32		54 K
		MATPRO	FRAIL
Cladding burst temperature at known pressure	(158/158)	290 K	Not Analyzed
Cladding burst pressure at known temperature	(64/64)	34 MPa	Not Analyzed
Cladding permanent hoop strain	(370/370)	57% cladding OD	Not Analyzed

FRAPCON-1 MODEL ASSESSMENT — SUMMARY OF STANDARD DEVIATIONS BETWEEN MEASUREMENTS AND PREDICTIONS

Output Parameter	Sample Size (# of Rods/# of Points)	Standard Deviation FRAPCON-1
Fuel Centerline Temperature	32/274 (Pressurized Rods)	294K
	61/472 (Unpressurized Rods)	170K
Released Fission Gas	145/145	15.9%
Rod Internal Pressure	20/330 (Unpressurized Rods)	1.38 MPa
	28/285 (Pressurized Rods)	1.93 MPa
Gap Closure Heat Rating	88/88	11.4 KW/M
Axial Fuel Thermal Expansion	18/160	0.37%
Permanent Fuel Axial Deformation	97/354	0.45%
Permanent Cladding Hoop Strain	154/358	0.47%
Permanent Cladding Axial Strain	96/119	0.15%
Cladding Surface Corrosion Layer	40/69	5.8 micron
Cladding Hydrogen Concentration	33/46	37.2 ppm
Gap Conductance	17/112 (Unpressurized Rods)	10821 W/m^2K
	20/115 (Pressurized Rods)	21200 W/m^2K
Fuel Off-Centerline Temperature	20/111	208K

EXPECTED FUEL CODE ACCOMPLISHMENTS IN FY 80/81

A. ASSESSMENT OF FRAP-T5 COMPLETED

B. COMPLETION AND ASSESSMENT OF FRAPCON-2 - LAST VERSION OF CODE - MODEL UPDATING AS A RESULT OF ASSESSMENT AND NEW DATA WILL CONTINUE. HOWEVER, A NEW VERSION I.E., FRAPCON-2 MOD 1 WILL NOT BE MADE UNTIL SUFFICIENT CHANGES TO THE MODELS WARRENT IT.

C. COMPLETION AND ASSESSMENT OF FRAP-T6 - LAST VERSION OF CODE.

D. MATPRO-11 REVISION-1 COMPLETED

EXPECTED FUEL CODE ACCOMPLISHMENTS IN FY 80/81 (CONT.)

E. MAJOR IMPROVEMENTS EXPECTED:

FRAP-T6: LINK WITH FASTGRASS GAS RELEASE MODEL FROM ANL, A NEW BALLOONING MODEL BASED ON MRBT RESULTS, COMPLETE DYNAMIC STORAGE ALLOCATION, AN UPDATED FAILURE SUBCODE (FRAIL 6) COMPATIBLE WITH BALLOON-2, IMPROVED USER INPUT AND OUTPUT, θ-VARYING HTC MODEL, AND MANY OTHER SMALLER IMPROVEMENTS. COMPLETION DATE JANUARY 26, 1981.

FRAPCON-2: LINK WITH FASTGRASS, COMPLETE DYNAMIC STORAGE ALLOCATION, PELET MECHANICAL PACKAGE FROM GAPCON-3, IMPROVED INEL MECHANICAL PACKAGE, IMPROVED RELOCATION MODELS FOR BOTH MECHANICAL PACKAGES, ANS 5.4 GAS RELEASE OPTION, NRR-APPROVED EM MODEL OPTIONS, AND MANY OTHERS. COMPLETION DATE AUGUST 15, 1980.

MATPRO-11 REVISION-2: INC BCL ANNEALING PROPERTIES, TRUE STRESS/STRAIN U.F. DATA, REVISED CLAD CREEP AND THERMAL EXPANSION MODELS, UPDATED HOT PRESSING MODEL. COMPLETION MID 1981.

WORK PLANNED FOR FY 81 AND BEYOND

A. BEGIN DEVELOPMENT OF A SMALL BREAK (SLOW TRANSIENT) FUEL ROD DAMAGE CODE BASED ON AND LINKABLE TO FRAP-T AND FRAPCON.

B. CONTINUE TO IMPROVE THE MOST CRITICAL MODELS IN FRAP-T AND FRAPCON; NAMELY, FUEL RELOCATION AND CRACKED FUEL THERMAL AND MECHANICAL PROPERTIES, CLAD BALLOONING, PCI FAILURE ANALYSIS, AND LINKS WITH T/H CODES SUCH AS TRAC AND COBRA.

C. COORDINATE WITH NRR PERSONNEL TO PLAN AND ACHIEVE FUEL ROD BEHAVIOR STUDIES PERTINENT TO LICENSING STUDIES USING THE ABOVE CODES.

PROGRAMS TO STUDY FUEL ROD PROPERTIES

Halden Tests (EG&G)

IFA-429 — In-Reactor Measurement of Helium Absorption, Steady State and Transient Fission Gas Release, and Fuel Centerline Temperature as a Function of Burnup, Power, Gas Pressure, and Pellet Cladding Gap.

18 PWR — Type Rods-Pressurized to 375 psi — 25 cm Long.

IFA-430 — In-Reactor Measurement of Transient Axial Gas Flow and Centerline Temperature as a Function of Gap Size, Power, and Gas Flow Rates Plus Two Rods Unpressurized Instrumented for Fuel Temperature Measurements.

0-10

ACCOMPLISHMENTS TO DATE FOR IFA'S 429 AND 430

IFA-429 - HELIUM ABSORPTION REPORT ISSUED. RESULTS: AMOUNT OF HELIUM ABSORBED REDUCES PRESSURE BY AN INSIGNIFICANT AMOUNT (1.5%). PERIODIC POWER INCREASES (UP TO 50%) DID NOT DRIVE OUT THE ABSORBED HELIUM. BURNUP IS NOW AT 9000-24000 MWD/MTM. IRRADIATION WILL CONTINUE THROUGH 1980 AND INTO 1981. A PIE REPORT WILL BE ISSUED ON TWO RODS REMOVED AFTER 8000 MWD/MTM AND TWO RODS AFTER 30,000 MWD/MTM. IN 1981 OR 1982 THE TEST TRAIN WILL BE REMOVED AFTER 50,000 MWD/MTM.

IFA-430 - BEGAN IRRADIATION 11/26/78. PRELIMINARY RESULTS INDICATE THAT AT POWERS WHERE REDUCED GAS FLOW WAS EXPECTED, AN EFFECTIVE GAS GAP OF GREATER THAN ONE-HALF THE INITIAL WAS PRESENT INDICATING THAT THE FUEL CRACKS ARE NOT TIGHTLY CLOSED. PRESENT BURNUP IS 3000 MWD/MTM.

DATA USING DIFFERING GAP GAS COMPOSITIONS OF He AND Xe (UP TO 10% Xe) HAVE VERIFIED THE MODELS IN FRAP-T FOR GAP CONDUCTANCE TO PRESSURES OF 1.0 MPA. AT PRESSURES > 1.0 MPA AND Xe CONCENTRATIONS OF 10%, THE CODE PREDICTED ABOUT 20% LOWER GAP CONDUCTANCE THAN OBSERVED.

IRRADIATION WILL CONTINUE THROUGH 1980 AND 1981. DATA WILL BE CONTINUALLY COLLECTED AND REPORTED.

PROGRAMS TO STUDY FUEL ROD PROPERTIES (CONT'D)

Halden Tests (PNL)

IFA-431/432/527 — In-Reactor Measurement of Centerline Temperatures (Both Ends) as a Function of Burnup, Power, Gap Width, and Gas Composition. Six 50 cm Unpressurized BWR Type Rods Each Assembly.

IFA-431: PIE Complete, Peak Burnup 5000 MWD/MTM. Reports Issued: NUREG/CR-0318, NUREG/CR-0332, NUREG/CR-0749, -0797.

IFA-432: Presently In-Reactor, Average Burnup 24,000 MWD/MTM. 16 of 26 Instruments Still Working. To be Discharged From Reactor in CY 1981. Reports Issued: NUREG/CR-0220, -0560, -1139

IFA-527: Xenon Filled Rods to Determine Pellet Relocation Effects. To go In-Reactor MAY 1980.

PROGRAMS TO STUDY FUEL ROD PROPERTIES (CONT'D)

Halden Project Sponsorship

IFA-513 — Same as IFA-431 Except: He-Xe Gas Mixtures; Longer Length; Continuously Recording Pressure Transducers; Intermediate Power; and One Rod Pressurized to 45 psi Helium. Began Irradiation 11/78. Rods Will be Used Later in PBF for RIA and LOCA Tests. Decision Regarding Removal/Continued Irradiation to be Made CY 1980.

Reports Issued: NUREG/CR-0862, NUREG/CR-1077.

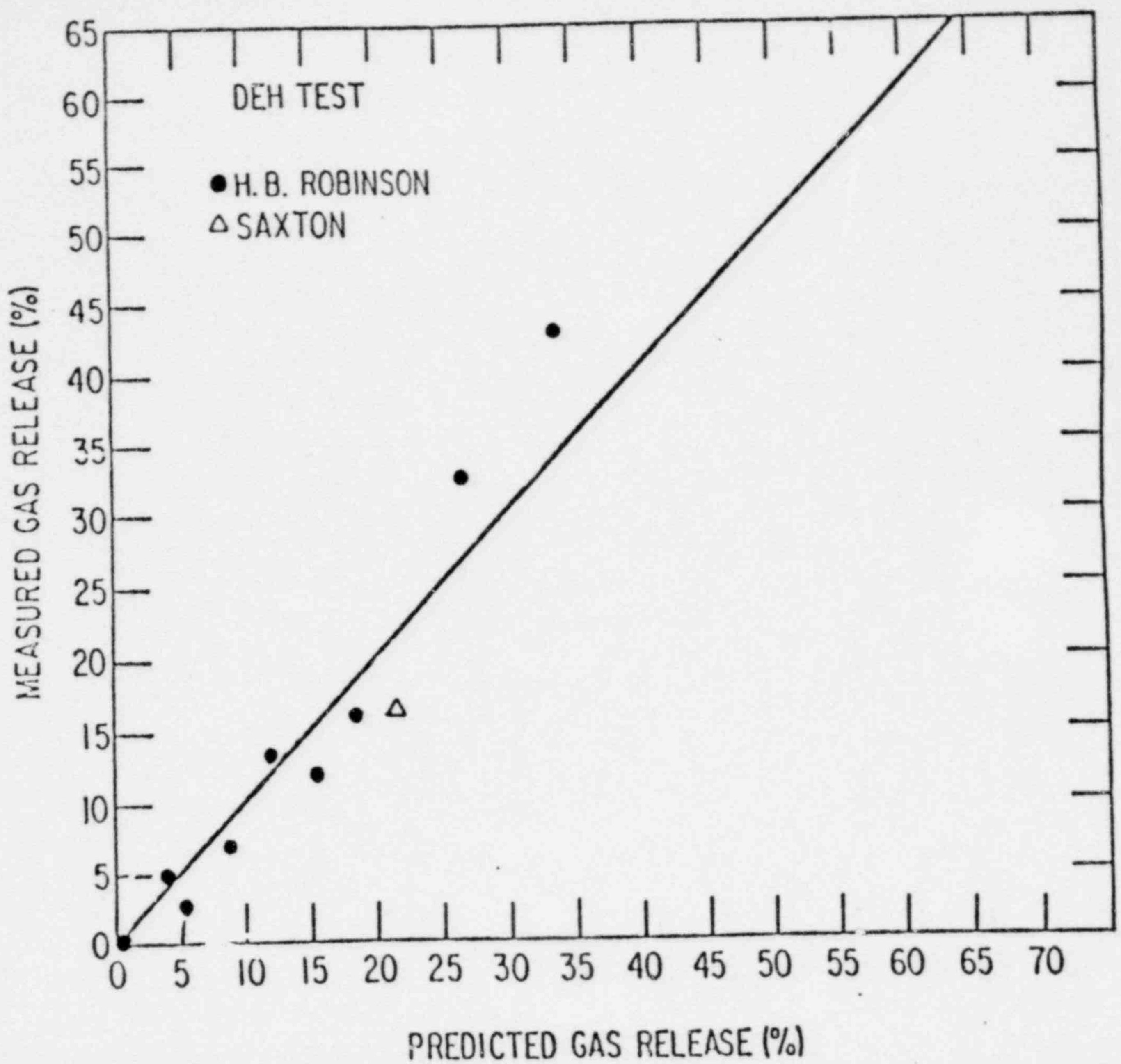
ACCOMPLISHMENTS TO DATE FOR IFA'S 431, 432 AND 513

A. BOL MEASUREMENTS OF TEMPERATURE, POWER, AND CLADDING ELONGATIONS RESULTED IN:

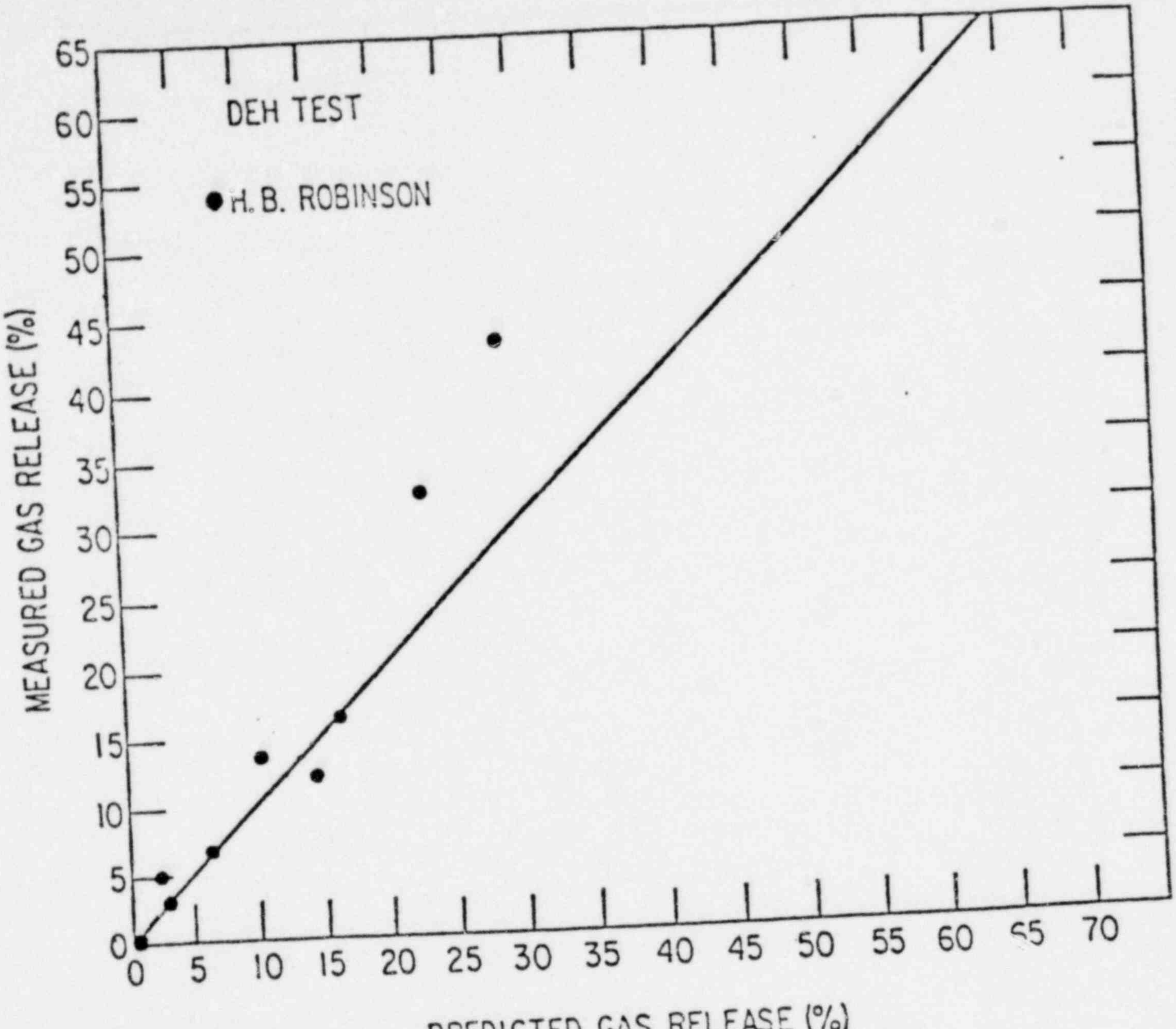
1. NO HIGH BURNUP ENHANCED FISSION GAS RELEASE NOTED TO DATE (24,500 MWD/MTM).
2. NO ADVERSE EFFECTS NOTED IN TWO RODS CONTAINING DENSIFYING FUEL.
3. THE DEVELOPMENT OF A NEW MODEL FOR FUEL RELOCATION, EFFECTIVE FUEL CONDUCTIVITY, AND CRACKED FUEL ELASTIC MODULI WHICH WILL BE TESTED IN FRAPCON-2.
4. CRACKED FUEL CONDUCTIVITY WAS REDUCED BY 20% AND THE ELASTIC MODULI TO ABOUT 1/40 OF SOLID UO_2 FOR 80% FUEL RELOCATION AT 30 KW/M.
5. THE RESULTING FUEL/CLAD GAPS HAVE REMAINED ESSENTIALLY CONSTANT SINCE.

B. PIE COMPLETE ON 431. IFA-432 WILL BE REMOVED IN SUMMER OF 1981.

O-14

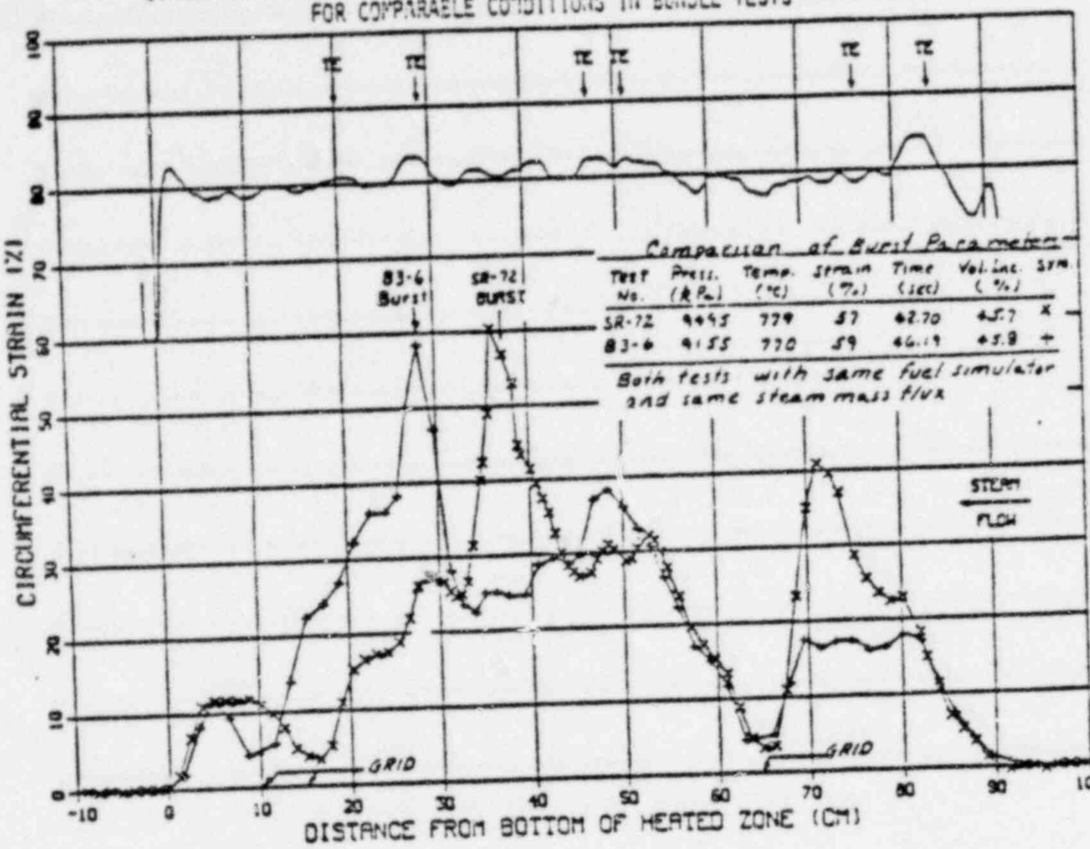


GRASS-SST-PREDICTED TRANSIENT GAS RELEASE



FASTGRASS-PREDICTED TRANSIENT GAS RELEASE

STRAIN IN SINGLE ROD HEATED SHROUD TESTS TYPICAL OF STRAIN OBSERVED
FOR COMPARABLE CONDITIONS IN BUNDLE TESTS

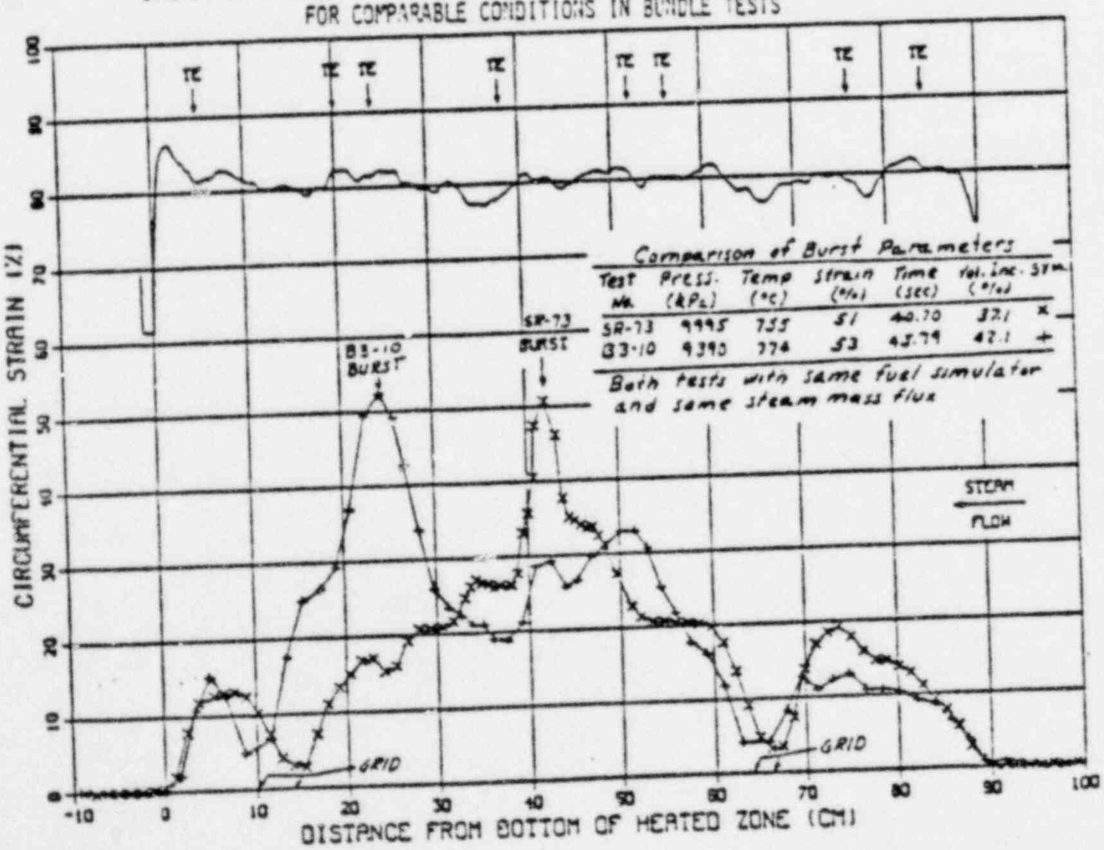


Comparison of Burst Parameters

Test No.	Press. (kPa)	Temp. (°C)	Strain (%)	Time (sec)	Vol. Inc. (%)	Sym.
SR-72	9495	779	57	42.70	45.7	x
B3-6	9155	770	59	46.19	45.8	+

Both tests with same fuel simulator and same steam mass flux

STRAIN IN SINGLE ROD HEATED SHROUD TESTS TYPICAL OF STRAIN OBSERVED
FOR COMPARABLE CONDITIONS IN BUNDLE TESTS



Comparison of Burst Parameters

Test No.	Press. (kPa)	Temp (°C)	Strain (%)	Time (sec)	Vol. Inc. (%)	Sym.
SR-73	9995	755	51	40.70	37.1	x
B3-10	9390	774	53	45.79	47.1	+

Both tests with same fuel simulator and same steam mass flux

PCI FAILURE BY STRESS-RUPTURE

- o STUDY BY KASSNER, ANL, BEGUN IN FY 80
- o EXAMINATION OF PCI FAILURE IN SPENT FUEL CLADDING BY LOADING SPECIMEN WITH EXTERNAL PRESSURE AND INTERNAL EXPANDING MANDREL IN AUTOCLAVES
- o USING HIGH-TEMPERATURE STRAIN GAGES ON EXTERIOR SURFACE OVER "CRACK" OF EXPANDING MANDREL TO SENSE HOOP STRESS AND INITIATION AND GROWTH RATE OF GROWING PCI CRACK
- o TESTS IN FY 80 AND FY 81 SHOULD HAVE ESTABLISHED STRESS-RUPTURE FAILURE CURVES WITHOUT STRESS CORRODANT
- o TESTS IN FY 82 TO EXAMINE FAILURE WITH STRESS CORRODANTS PRESENT
- o BWR SPENT FUEL CLADDING WILL BE EXAMINED AS WELL AS AVAILABLE PWR MATERIAL (H. B. ROBINSON, MAINE YANKEE, OCONEE)

STRAIN-RATE RAMPING TO PCI FAILURE EX-PILE

- STUDY BEGUN IN FY 81 BY P. PANKASKIE, BNWL.
- OBJECTIVE IS TO OBTAIN DATA FOR ESTABLISHING PARAMETERS IN PROFIT MODEL OF PCI FAILURE, DEVELOPED FOR CPB/NRR IN FY 80.
- SPECIMEN CONSISTS OF TUNGSTEN WIRE CENTERLINE HEATER, UO_2 ANNULAR PELLETS, ZIRCALOY FUEL CLADDING, EXTERNALLY PRESSURIZED IN A LOOP. HEATER WILL BE RAMPED IN POWER TO LOAD CLADDING AT VARIOUS RATES, POWER INCREMENT BETWEEN HARD CONTACT AND RAMP FAILURE DETERMINED. CAN ALSO BE USED FOR TIME TO FAILURE AT PRESELECTED LOADING PAST HARD CONTACT.
- INITIAL STUDIES TO BE WITH UNIRRADIATED CLADDING, THEN IRRADIATED CLADDING, AND WITH AND WITHOUT STRESS-CORRODANT.

STRAIN-RATE RAMPING TO PCI FAILURE IN-PILE: DEMO-RAMP PROGRAM

- NRC PARTICIPATING IN DEMO-RAMP PROGRAM AT STUDSVIK ON HIGHER BURNUP FUEL.
- SELECTED PRE-IRRADIATED FUEL RODS TO BE POWER-RAMPED IN THE R2 REACTOR AT STUDSVIK TO DETERMINE POWER INCREMENT OR TIME TO FAILURE IN FUEL RODS AT ABOUT 25 MWD/T U BURNUP.
- DATA WILL BE COMPARED WITH THAT FOR LOWER BURNUP RODS OF SUPERRAMP AND INTERRAMP PROGRAMS.
- THIS PHASE OF STUDY COMPLETED IN JUNE 1981.

STRAIN-RATE RAMPING TO PCI FAILURE IN-PILE: PBF-OPTRAN TESTS

- PBF-OPTRAN TESTS DESIGNED TO EXAMINE PELLET-CLADDING INTERACTION DURING POWER RAMPING CAUSED BY VARIOUS SCENARIOS OF OPERATIONAL TRANSIENTS LIKELY IN COMMERCIAL LWR POWER PLANTS

- DATA DETERMINED WILL INCLUDE ONE OR MORE OF THE FOLLOWING:
  - INCREMENT OF POWER FROM BASE TO HARD CONTACT BETWEEN PELLET AND CLADDING
  - INCREMENT OF POWER FROM HARD CONTACT TO CLADDING FAILURE
  - CLADDING FAILURE AS FUNCTION OF RATE OF RAMPING
  - TIME TO FAILURE AS FUNCTION OF POWER INCREMENT AFTER HARD CONTACT
  - EFFECT OF BURNUP
  - EFFECT OF REPEATED CYCLING BELOW RAMP FAILURE LIMIT

- SCHEDULE OF TESTS:

  FIRST OPTRAN TEST PLANNED FOR 1980

  FOUR OPTRAN TESTS PLANNED FOR FY 1981

  OPTRAN TEST MATRIX COMPLETED IN FY 1982

  TOTAL OF SEVEN OPTRAN TESTS PLANNED, SIX 4X, ONE 9-ROD BUNDLE

FISSION PRODUCT TRANSPORT ANALYSIS - TRAP CODE - BCL

OBJECTIVE: TO DEVELOP A MECHANISTIC COMPUTER CODE TO MODEL FISSION PRODUCT TRANSPORT BEHAVIOR WITHIN THE PRIMARY COOLANT SYSTEM AND CONTAINMENT.

STATUS: PRIMARY SYSTEM MODEL ESSENTIALLY COMPLETE.
RFP ISSUED FOR ADVANCED CODE.

ACCOMPLISHMENTS: DEPOSITION OF FISSION PRODUCTS WITHIN REACTOR COOLANT SYSTEM UNDER CORE MELT ACCIDENT CONDITIONS IS <u>RELATIVELY</u> UNIMPORTANT.

GROWTH OF AEROSOLS WITHIN RCS IS IMPORTANT.

FUTURE PLANS: IMPROVE TRAP CODE MODELS,
EXTEND TRAP CODE TO MODEL CONTAINMENT FISSION PRODUCT BEHAVIOR,
SOURCE TERM MODELLING,
SENSITIVITY ANALYSIS,
DEFINE VERIFICATION TEST FACILITY FUNCTIONAL DESIGN REQUIREMENTS.

FUNDING: FY 80-83 -- 10-12 MAN-YEARS

SEPARATE EFFECTS TESTS FOR TRAP CODE - SANDIA

OBJECTIVE: TO PROVIDE BASIC DATA ON FISSION PRODUCT COMPOUND VAPOR PRESSURES AND CHEMICAL INTERACTIONS IN A HIGH TEMPERATURE STEAM ENVIRONMENT TO SUPPORT DEVELOPMENT OF THE TRAP CODE.

STATUS: VAPOR PRESSURE EXPERIMENTS IN PROGRESS AT SANDIA LABORATORIES AND AT THE NEW MEXICO INSTITUTE FOR MINING AND TECHNOLOGY - COMPOUNDS OF CESIUM AND IODINE BEING INVESTIGATED.

FISSION PRODUCT REACTION SYSTEM (FPRS) APPROXIMATELY 60% COMPLETE.

FUTURE PLANS: VAPOR PRESSURE TESTS ON OTHER FISSION PRODUCT COMPOUNDS WILL BE CONDUCTED AS NECESSARY.

FPRS WILL BE COMPLETED AND TESTING INITIATED.

NON-INTRUSIVE REAL TIME FISSION PRODUCT COMPOUND IDENTIFICATION BY LASER RAMAN SPECTROSCOPY WILL BEGIN IN THE FPRS APPARATUS.

FUNDING: FY 80 - 150K -- FY 81 - 210K

FISSION PRODUCT VAPOR DEPOSITION EXPERIMENTS - BCL

OBJECTIVE: TO PROVIDE EXPERIMENTALLY DERIVED FISSION PRODUCT DEPOSITION RATES AT HIGH TEMPERATURE ON PRIMARY SYSTEM SURFACES TO AID IN DEVELOPING THE TRAP CODE. TO DETERMINE THE NATURE OF THE INTERACTION BETWEEN VARIOUS FISSION PRODUCT COMPOUNDS AND PROTOTYPIC SURFACES.

STATUS: CONSTRUCTION OF FISSION PRODUCT VAPOR DEPOSITION APPARATUS IS COMPLETE.

STAINLESS STEEL AND INCONEL DEPOSITION COUPONS HAVE BEEN SUBJECTED TO SIMULATED PRIMARY SYSTEM AGING.

IODINE VAPOR DEPOSITION EXPERIMENTS HAVE BEEN INITIATED AND WILL BE COMPLETE IN APPROXIMATELY 4 MONTHS.

FUTURE PLANS: PROGRAM TO BE COMPLETED FY 80.

CESIUM AND TELLURIUM VAPOR DEPOSITION EXPERIMENTS WILL BEGIN IN APPROXIMATELY 4 MONTHS.

DATA WILL BE ANALYZED AND MODELS DEVELOPED FOR INCORPORATION INTO TRAP CODE.

FUNDING: FY 79 (PART OF TRAP DEVELOPMENT PROGRAM) FY 80 - 95K FY 81 - 0

STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT - BCL

OBJECTIVE: TO DEVELOP MECHANISTIC COMPUTER MODELS FOR IODINE TRANSPORT WITHIN THE STEAM GENERATOR AND SECONDARY SYSTEM UNDER SGTR ACCIDENT CONDITIONS. TO EXPERIMENTALLY DETERMINE THE AMOUNT OF ATOMIZATION OF THE PRIMARY COOLANT DURING BLOWDOWN INTO THE SECONDARY SYSTEM.

PROJECT STATUS: DESIGN OF THE EXPERIMENTAL FACILITY TO MEASURE PRIMARY COOLANT ATOMIZATION IS COMPLETE AND CONSTRUCTION IS UNDERWAY.

THE IODINE TRANSPORT MODELS HAVE BEEN DEVELOPED AND ARE BEING ASSEMBLED INTO A COMPUTER CODE.

FUTURE WORK: PROJECT WILL BE COMPLETED IN FY 80.

THE AMOUNT OF ATOMIZATION AND DROP SIZE DISTRIBUTION WILL BE MEASURED AS A FUNCTION OF PRESSURE DIFFERENTIAL (100 - 1300 PSI).

THE SGTR IODINE TRANSPORT COMPUTER CODE WILL BE COMPLETED AND DELIVERED TO NRC/NRR.

FUNDING: FY 79 - 70K FY 80 - 63K

FISSION PRODUCT RELEASE FROM LWR FUEL - ORNL

OBJECTIVE: TO DETERMINE THE QUANTITY, SPECIES AND CHEMICAL FORM OF FISSION PRODUCTS RELEASED FROM DEFECTED FUEL RODS UNDER ACCIDENT CONDITIONS.

STATUS: PROGRAM COMPLETE.

FISSION PRODUCT RELEASE FROM LWR FUEL - HIGH TEMPERATURE

OBJECTIVE: TO EXPAND THE INVESTIGATION OF FISSION PRODUCT RELEASE FROM DEFECTED LWR RODS WITHIN THE TEMPERATURE RANGE OF 1000°C TO 1750°C.

STATUS: 189 RECEIVED - PROGRAM START AWAITING SUPPLEMENTAL FUNDING AUTHORIZATION.

FUNDING: FY 80S - 365K -- FY 81 - 400K -- FY 82 - (575K)

FISSION PRODUCT RELEASE FROM LWR FUEL - HIGH TEMPERATURE

SCOPE: MEASURE RELEASE FROM 12 BWR AND PWR RODS UP TO ~1750°C.

DETERMINE Cs, Kr, Ru, Ag, Sb, AND Eu BY GAMMA-SPECTROSCOPY.

DETERMINE I BY NAA.

RATIONALE: CAN USE EXISTING APPARATUS WITH MINOR CHANGES TO REACH ~1750°C.

LIMITATION: MAXIMUM TEMPERATURE IS ~1750°C.

22-0

CHARCOAL FILTER IODINE RETENTION PERFORMANCE - NRL

OBJECTIVE: TO INVESTIGATE THE PERFORMANCE OF ACTIVATED CHARCOALS IN REMOVING AIRBORNE RADIOIODINE UNDER LWR ACCIDENT CONDITIONS. TO ASSESS THE EFFECTS OF IN-SERVICE WEATHERING AND EXPOSURE TO CONTAMINANTS ON THE REMOVAL AND RETENTION OF RADIOIODINE.

STATUS: 189 RECEIVED - PROGRAM INITIATION AWAITING SUPPLEMENTAL FUNDING AUTHORIZATION.

PROGRAM ELEMENTS: EXPOSE SAMPLES OF COMMERCIALLY AVAILABLE ACTIVATED CHARCOALS (IMPREGNATED WITH TEDA, KI_x AND OTHER WIDELY USED IMPREGNANTS) TO WEATHERING AND TO KNOWN ATMOSPHERIC CONTAMINANTS.

TEST THESE CHARCOALS FOR RADIOIODINE RETENTION UNDER THE RANGE OF SEVERE ACCIDENT CONDITIONS INCLUDING:

A. EXPECTED RADIOIODINE LOADINGS,

B. TOTAL RADIATION LOADING, AND

C. EXPECTED TEMPERATURE AND HUMIDITY ENVIRONMENT.

FUNDING: FY 80S - 110K -- FY 81 - 115K -- FY 82 - 120K

FISSION PRODUCT RELEASE - MELTING FUEL

OBJECTIVE: TO EXPERIMENTALLY DETERMINE THE RELEASE OF FISSION PRODUCTS FROM IRRADIATED LWR FUEL IN THE TEMPERATURE RANGE ~1800°C TO 2800°C.

STATUS: PROGRAM UNDER EVALUATION.

PROGRAM ELEMENTS: CONSTRUCT A FACILITY CAPABLE OF TRANSIENT HEATING OF COMMERCIALLY IRRADIATED FUEL ROD SEGMENTS TO MELTING IN A STEAM OR STEAM/H_2 ENVIRONMENT.

CONDUCT EXPERIMENTS TO MEASURE THE RATE, QUANTITY, SPECIES AND CHEMICAL FORM OF RELEASED FISSION PRODUCTS UNDER HIGH TEMPERATURE INCIPIENT FUEL MELT CONDITIONS.

FUNDING: FUNDING IDENTIFIED FOR FY 83 AND BEYOND IF NEED FOR PROGRAM IS ESTABLISHED.

FISSION PRODUCT LEACHING

OBJECTIVE: TO EXPERIMENTALLY INVESTIGATE THE LONG TERM RELEASE OF FISSION PRODUCTS FROM SEVERELY DAMAGED FUEL RODS UNDER THE PHYSICAL AND CHEMICAL CONDITIONS EXPECTED WITHIN THE REACTOR VESSEL FOLLOWING A SEVERE ACCIDENT.

STATUS: PROPOSED PROGRAM UNDER REVIEW.

PRELIMINARY JUDGEMENT IS TO NONSUPPORT.

FISSION PRODUCT TRANSPORT VERIFICATION FACILITY

OBJECTIVE: TO CONSTRUCT OR MODIFY AN EXISTING FACILITY FOR THE PURPOSE OF TESTING THE VALIDITY OF CURRENT LWR FISSION PRODUCT TRANSPORT CODES SUCH AS TRAP-MELT, CORRAL, NAUA, ETC. EMPHASIS WILL BE ON CONTAINMENT FISSION PRODUCT BEHAVIOR. HOWEVER PRIMARY SYSTEMS EFFECTS WILL ALSO BE INCLUDED.

STATUS: PROGRAM UNDER EVALUATION.

PROGRAM ELEMENTS: DEFINE FUNCTIONAL DESIGN REQUIREMENTS FOR A FISSION PRODUCT TRANSPORT CODE VERIFICATION TEST FACILITY.

INVESTIGATE THE CAPABILITY OF EXISTING FACILITIES, SUCH AS STCF, NSPP, ETC., IN MEETING THESE REQUIREMENTS.

CONSTRUCT (OR MODIFY) A FACILITY TO PERFORM TESTS.

CONDUCT FISSION PRODUCT BEHAVIOR TESTS IN PROTOTYPIC ACCIDENT ENVIRONMENTS.

FUNDING: FY 82 - 1000K

TMI FISSION PRODUCT RELEASE DATA EXAMINATION

OBJECTIVE: TO PROVIDE FUNDING FOR FISSION PRODUCT RELEASE AND TRANSPORT DATA GATHERING ACTIVITIES AND ANALYTICAL SUPPORT DURING TMI RECOVERY.

STATUS: JOINT NRC, DOE, EPRI, GPU DATA GATHERING ACTIVITY UNDERWAY.

DOE HAS COMMITTED TO PROVIDE GOVERNMENT SHARE OF FUNDING.

FUNDING INDICATED BELOW REPRESENTS A CONTINGENCY FOR DATA ACQUISITION AND ANALYSIS NOT AGREED TO BY JOINT COMMITTEE (AND NOT FUNDED BY DOE).

FUNDING: FY 80S - 175K -- FY 81 - 85K -- FY 82 - 200K

CLASS 9 ACCIDENT RESEARCH: PROGRAM LOGIC

CLASS 9 ACCIDENTS CHALLENGE CONTAINMENT. PROGRAM OBJECTIVE: DETERMINE BEST ESTIMATE OF RISK; ANALYSIS AND ASSESSMENT OF SPECIAL FEATURES.

NATURE OF CHALLENGES IDENTIFIED IN WASH-1400:

1. CAN PRESSURES IN PRIMARY SYSTEM BREACH THE SECONDARY? (EVENT V; SG TUBE RUPTURE)
2. CAN A MELTED DOWN CORE BREACH THE PV AND OVER-LOAD THE CONTAINMENT? (DEBRIS BED COOLABILITY; STEAM SPIKE)
3. CAN A HYDROGEN EXPLOSION BREACH THE CONTAINMENT? (HYDROGEN LOADS; HYDROGEN CONTROL; CONTAINMENT RESPONSE)
4. CAN A STEAM EXPLOSION BREACH THE CONTAINMENT? (EXPLOSION EFFICIENCY; PV LOADING)
5. CAN A HOT CORE MELT THE BASEMAT? (CORE CONCRETE INTERACTIONS; CORE CATCHERS)
6. CAN THE CONTAINMENT SLOWLY HEAT UP AND BE OVER PRESSURIZED? (AUXILIARY SPRAYS; FVCS)
7. CAN MAINTENANCE OF VITAL FUNCTIONS BYPASS CONTAINMENT OR THREATEN ITS INTEGRITY?
8. CAN FAILURES IN I & C COMPROMISE SAFETY SYSTEMS?

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