PROGRESS IN FUEL BEHAVIOR RESEARCH

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The Fuel Behavior Research Branch welcomes you to the Seventh WRSR Information Meeting.

This past year has been very productive in all portions of our programs. The Three-Mile Island Accident involved all the members of the Branch in supporting Commission efforts to estimate the extent of core damage and to determine that it was safe to shut off the pump and go on natural circulation. Special thanks to all of our contractors who were called upon for aid during TMI. We also found a great deal of use for the data developed earlier in the program. Worthy of special mention is a code written by G. Marino to calculate the temperature rise and oxidation kinetics during the initial uncovery of the core and the detailed estimates of core condition by Marion Picklesimer. The past 5 months have been particularly difficult for the Branch because M. Picklesimer and I have been detailed away, on a full time basis, to the NRC Special Investigation Group and, the Branch has had to continue their own work, as well as ours, and plan new programs in response to TMI. Special thanks should go to Don Hoatson, who is acting in my place.

A. TESTS IN THE PBF REACTOR

Since the last Information Meeting, Hank Zeile, Phil MacDonald, and their EG&G coworkers have performed ten LOCA blowdowns (includes four expected TC-1 blowdowns) and one RIA test in the PBF. They have just drafted a report which summarizes the experimental data for DNB, RIA, and LOCA tests performed in the PBF during the past 5 years. This report compares the experimental data with engineering test predictions, made using the FRAP-T4 code, and represents a good summary of each.

Between February and May 1979, five LOCA blowdown tests were performed in the PBF to determine whether planned LOFT tests L2-3, L2-5, and L2-4, with unpressurized rods, could be run without threatening the integrity of the LOFT fuel bundles. These tests included blowdowns from local rod power levels as high as 53 KW/M and with measured peak clad temperatures (PCT's) as high as 1175K (1655°F). No creepdown was observed in any of the rods despite the exposure of several of the test fuel rods to all four blowdown - heatup quench cycles.

Two PBF-LOCA tests, performed this summer, (LOC-3 and LOC-5) used both fresh and preirradiated (15,000 MWD/T) rods which were pressurized to match anticipated fill gas pressures of commercial rods at beginning of life and at or near end of life. The tests were planned to examine the relative clad ballooning associated with clad rupture for peak cladding temperature in the 1051K-1500K (1430°F-2240°F) range. Maximum clad ballooning and resultant flow channel blockage may be possible in this temperature range. In test LOC-3, circumferential clad ballooning as high as 50 percent was observed, but this would not threaten local channel blockage, because more than 70 percent average circumferential coplanar ballooning of all four adjacent rods is required to block a flow channel in a commercial power reactor and some flow through any cladding ruptures is possible even then. Test LOC-5 was performed near the end of the fiscal year, and the rods have not yet been examined.

Just before this meeting, four thermocouple quench tests were performed in the PBF to measure the effect of external thermocouples on clad quench characteristics after nuclear blowdown and heatup. These tests will be discussed by subsequent speakers.

The PBF RIA test was performed with four preirradiated rods (5,000 MWD/T) at an energy deposition level (200 cal/gm) which was near the clad damage threshold level, i.e., the level expected to cause incipient clad failure and resultant release of fission products into the coolant. Only one of the four test rods failed, but it showed a series of small longitudinal clad cracks which looked like pellet-cladding-interaction (PCI) induced clad failures. This rod had not been opened after preirradiation while the companion rods had been, so the effect of fill gas chemistry will be studied in future RIA damage threshold tests using unprocessed preirradiated rods, i.e., test instrumentation will not be added if it requires breaching the clad prior to the test.

EG&G speakers will discuss the results of this low damage threshold RIA test and also the PIE data from the higher rod enthalpy RIA test which was completed just before the last Information Meeting, but which could not be examined until after the meeting. The PIE of the earlier RIA test showed significant swelling of the preirradiated fuel as a result of its increase in enthalpy to about 260 cal/gm radial average. The relationship of attainable enthalpy and associated swelling at very high burnup levels (40-60 GWD/T) will have to be examined in order to assess the associated threat to post-accident core coolability.

Court Hann, Ed Courtright, and their PNL coworkers provided the hardware for the EG&G RIA tests and will provide hardware and test procedures for other planned reactor tests which will be discussed Thursday afternoon and Friday.

B. OUT-OF-REACTOR EXPERIMENTAL PROGRAM

1. Multirod Burst Test Program - ORNL (MRBT)

During the past year, MRBT has run 17 single rod burst tests with the new facility having a heated shroud controlled to be at the same temperature and rate of temperature rise as the specimen. The burst strains obtained have been higher than with an unheated shroud, and are more consistent with those observed in the multi-rod bundle tests. The most recent tests indicate that the influence of the thermohydraulic conditions is controlling at the lower heating rates. The new tests need to be run at constant power conditions with steam flow rate varied to control the heating rate. Such tests are now underway.

The examination of MRBT No. 3 (16 rods) has been completed and reported. The burst strains ranged from 42-77 percent (greater than the 31-59 percent of MRBT B-1), while the volume increase was 27-55 percent (29-55 percent in B-1), meaning that the deformation was more localized in B-3 than in B-1. This is consistent with the flow blockage estimated from the measured loss-of-cross-sectional flow area to be from 76-90 percent at the most deformed region (49-58 percent in B-1), the value depending on the method of measurement. These data are consistent also with the measured pressure drops in flow testing of a maximum of 75 percent in B-3; in comparison, to a maximum of 55 percent in B-1. One rod had a ballooned region of more than 32 percent circumferential strain of about 28 original tube diameters while the longest in B-1 was about 10 percent. All of these data agree with the differences expected with a change in heating rate from 30°C/second (B-1) to 10°C/second (B-3); the slower heating producing more ballooning and greater burst strains.

2. Cladding Creepdown - ORNL

Six in-pile creep tests under external pressure have been completed and a portion of the data analyzed. The temperatures have been 650°F (five tests) and 700°F (one test) at pressures ranging from 1900 to 2500 psi. In all tests to date, the data are consistent with out-of-pile tests under external pressure, in that ovalization increases until the mandrel is contacted, and cladding then wraps itself around the mandrel. Both the rate of ovalization and the rate of circumferential compressive straining are determined. The rate of circumferential compressive strain decreases significantly when the mandrel is contacted. Before mandrel contact is made by ovalization, the average circumferential compressive strain ranges from 2×10^{-5} hr^{-1} to 1×10^{-7} hr^{-1} for faster "before" rate and much less than 1×10^{-7} hr^{-1} for the slower "before" specimen (data reduction not yet completed); all values being for a test temperature of 650°F and the same fast flux. Comparison of the 650 and 700°F data indicates that the creepdown is still mostly thermally activated at 650°F. Mandrel contact has been definitely seen in 450 hours at 700°F for a differential pressure of 1900 psig. The data for locating the time of mandrel contact in the other tests is not as conclusive, and analyses are continuing. The last three tests of the series are scheduled to be stress-reversal tests to examine the effect of having internal pressures higher than system pressures in high burnup fuel rods.

1605 147

3. Properties of Irradiated Cladding - BCL

All PWR spent fuel cladding testing has been completed, and final reports are in preparation. Mechanical properties of spent fuel cladding from two PWR's have been determined as functions of many experimental test parameters. One of the major conclusions reached is that there is no essential difference in the mechanical properties of irradiated and unirradiated Zircaloy-4 cladding once the irradiated cladding has been heated to temperatures circa 1200°F in transient heating and about 1100-1150°F for more than 1-2 minutes in isothermal annealing. A second conclusion is that there is no major or essential difference in the responses of cladding from different vendors (in those lots examined).

Spent fuel cladding from a BWR is being obtained for similar tests in a scoping study to see if there are major differences in the responses of the cladding from the two types of LWR's.

4. Properties of Zircaloy-Oxygen Alloys - ANL

Although the originally scheduled experimental work was completed and reported last year, additional work was conducted to evaluate the conditions for, and the influence of, steam starvation and hydrogen atmospheres in the gap of the ballooned and burst specimen as reported by Kawasaki at last year's meeting. This work has now been completed and final reports are in press. The embrittlement criteria, now being recommended for LWR fuel element cladding, are stated as follows: (a) For resisting thermal shock during a LOCA reflood, the calculated thickness of wall containing less than or equal to 0.9 percent of oxygen by weight, based on average wall thickness at any axial location, shall be greater than 0.1mm (0.004 inches). (b) For handling, transport, and interim storage of oxidized fuel assemblies, the calculated wall thickness containing less than or equal to 0.7 percent of oxygen by weight, based on the average wall thickness at any axial location, shall be greater than 0.3mm (0.012 inches). The two criteria are based on the thermal shock and impact load failure boundaries of 0.03J and 0.3J, respectively, for the PWR cladding used. The influence of hydrogen to 2200ppm dissolved at high temperature in the cladding is included in the low temperature deformation and shock properties upon which these recommendations are based. In a LOCA, the hydrogen would have been dissolved in the burst fuel rod due to steam starvation occurring in the narrow gap between fuel and cladding in the less ballooned sections of the fuel rod.

1605 148

5. True Stress - True Strain - Constant True Strain Rate Testing of Zircaloy - University of Florida

Testing technique and equipment for elevated temperature tests have been completed, and most of the uniaxial tensile testing at temperatures to 600°C have been completed. These are the first data ever obtained in continuous testing in tensile stress at constant true strain rates to about 5×10^{-2} sec^{-1}. The stress - strain data are readily analyzed to true strains approaching one and very close to fracturing. The strain anisotropy observed in room temperature tensile tests persists to at least 600°C, although the strain ratios differ to some extent. In addition, the strain ratio remains constant in a given test until void generation in the fracture area begins just before fracture occurs. The data also indicate that significant adiabatic heating occurs in the neck of a straining tensile specimen, decreasing the stress required at a given strain below the true value. Testing in biaxial stress will begin shortly.

C. CORE MELTDOWN AND FISSION PRODUCT RELEASE AND TRANSPORT

1. Steam Explosion Phenomena

A series of 48 intermediate scale steam explosion tests have been conducted under the direction of Dr. Larry Buxton at Sandia Laboratories. In these tests, where from 1 to 27 kilograms of molten iron/aluminum oxide mixture was poured into an open tank of water, the largest observed thermal-energy-to-work conversion efficiency was less than 1.5 percent. This value is nearly a factor of 20 less than the maximum efficiency that is theoretically achievable. There was no observed relationship between the explosion efficiency and the mass of the melt. However, it was observed that the explosion efficiency increased with increasing depth of water in the tank.

Sandia researchers, under the direction of Dr. Michael Corradini, have been evaluating the mechanical response of the reactor vessel and containment building to steam explosions. Their preliminary observations are that it appears unlikely that a steam explosion would result in the generation of a large mass missile (such as the upper reactor vessel head). Additionally, their analysis indicates that the generation of small mass missiles (such as a control rod drive assembly) could be mitigated by the presence of upper reactor vessel internal structures.

2. Molten Core Interaction

Dr. Dana Powers, at Sandia Laboratories, has completed a series of small scale inductively heated and sustained tests where core melt materials were contacted with various types of reactor building concrete. These tests have provided valuable data for development of the advanced molten core/concrete interactions code, CORCON, which is being developed by Dr. James Muir. The initial version of the CORCON code has just recently been completed, and a user's manual describing the code will be issued within the next few months. This work will be presented during the morning session on Thursday.

Sandia has also developed a technique for analyzing the composition of the aerosols released during melt/concrete attack. In several melt/ concrete interactions tests nonradioactive fission product mocks were introduced into the melt. Aerosols released from the melt during these tests were collected and analyzed by spark mass spectroscopy. This new capability is providing data needed for the development of fission product release models for that period by hypothetical meltdown accidents when the molten core materials are attacking the lower reactor cavity concrete basemat. Dr. Powers will describe this technique and the results of these measurements during the Fission Product Release Workshop on Thursday afternoon.

3. Fission Product Release from LWR Fuel

At ORNL, a series of experiments designed to measure the release of fission products from defected fuel rods under accident conditions have been completed. In these tests, segments of commercial fuel rods irradiated in operating light water reactors were inductively heated in flowing steam atmosphere. For those tests where the maximum fuel rod temperature remained below 1200°C (corresponsing to a successfully controlled LOCA), the measured release of fission products, cesium and iodine, were one to two orders of magnitude less than the "gap release" assumptions employed in the Reactor Safety Study. Four high temperature tests were conducted where the peak fuel rod temperature was taken to between 1200°C and 1600°C. Results of these tests indicate that above about 1350°C the release of cesium and iodine increases dramatically. These tests were directed by Dr. Malinauskas and Dr. Lorenz at ORNL, and will be described during the Fission Product Release Workshop.

4. Fission Product Transport Analysis - TRAP

Battelle Columbus Laboratory is continuing development and improvement of the TRAP-MELT code which models fission product transport behavior within LWR primary coolant systems under severe accident conditions. A user's manual for this code was issued during the past fiscal year. Also completed was a series of experiments designed to measure the deposition rates of submicron-sized aerosols on surfaces. Dr. James Geiseke, of BCL, will present a discussion of this work Wednesday morning.

D. FUEL CODE DEVELOPMENT AND ASSESSMENT

Several accomplishments of note were achieved in the area of fuel code development and assessment. The assessment of the FRAP-T4 code was completed and a research information letter was issued. This version of our transient fuel code now has the capability to analyze a LOCA event from blowdowm through the complete reflooding. The code is available from the National Energy Software Center (NESC) at Argonne National Laboratories. The first version of the FRAPCON steady state code was independently assessed and sent to the NESC. This code is the first version of the marriage of the FRAP-S steady-state code series and the GAPCON-THERMAL code series. It contains models from GAPCON-THERMAL and FRAP-S3 as well as improvements in code structure and useability, such as dynamic storage allocation and subrountine modularization. As a result, the code executes faster and gives a slightly overall better fit to the large data bank used in the assessment. The development of the next version of the code is underway, and it will contain improved mechanical models from EG&G and BNWL code developers, as well as improved fuel relocation modeling from the results of our Halden test rigs. This version, FRAPCON-2, is considered to be our final numerical update of our steady-state fuel code. Future improvements in the code, if any, will be issued as modifications of the basic FRAPCON-2 tape.

The FRAP-T5 transient code was completed, sent to the NESC, and is currently being independently assessed. Results of this assessment will be presented later this morning by T. Laats of EG&G. This version of the code contains improved thermal/mechanical models based on our separate effects tests and is linked to the transient gas release code GRASS-SST developed by ANL. It also contains a link to an automated response surface uncertainty analysis subcode, the results of which will be presented by M. Bohn, of EG&G, later this morning.

In the area of separate effects fuel behavior tests, work is proceeding roughly on schedule. The IF-430 transient gas flow rig began irradiation in the Halden reactor. A fission product detection system has been attached from which we are hoping to learn a great deal about fission product release and its flow characteristics during mild power changes and steady-state operation. Initial results indicate that although substantial fuel relocation occurs at 15 kw/m during the first power ramp, an effective gap of more than half the initial was still available for gas flow at powers around 30 kw/m. It appears, therefore, that the fuel cracks do not tightly close, but remain open enough to allow gas communication along the rod. R. R. Hobbins and T. Appelhans, of EG&G, are the cognizant engineers for this project.

The out-of-pile direct electrical heating transient testing of irradiated fuels to determine transient fission gas release during a rapid transient (PCM-type) has been completed by S. Gehl of ANL. A final integrated report on the results of the program will be issued in FY 1980. Moreover, these results are being used in the assessment and development of the GRASS-SST code, the latest version of which FASTGRASS was completed this summer by J. Rest at ANL. FASTGRASS is 10 to 100 times faster than its more sophisticated older brother, GRASS-SST, and requires 40 percent less computer memory space. Although extensive assessment has not yet been completed, results to date indicate it to be in reasonable agreement with GRASS-SST.

In addition to the Halden assembly, managed by EG&G, that I have just talked of, two other experiments are managed by Battelle Northwest (PNL) are in the Halden Reactor - these are IFA-432 and 513. These irradiation tests are refining our understanding of fuel stored energy and pellet relocation. IFA-432 has reached an average burnup of over 20,000 MWD/MTM, and we anticipate leaving it in-pile to an average burnup of 30,000. IFA-513 has been in-pile for one year. The rods from this assembly (co-sponsored by Halden project) will eventually be tested in the PBF. We are particularly pleased with the accomplishments at PNL, under the leadership of Don Lanning, to fully analyze the power-temperature data coming from both steady-state and power ramp operation in Halden.

The other program at PNL, laboratory measurement of gap conductance, has been completed. Many precise measurements have been made over a wide range of temperatures, pressures, and gas compositions by John Garnier and Steve Begy. A report was issued earlier this year, and the final report will be issued by the end of the year. Comparison of the data with currently used equations has shown these equations to be relatively good, but that improvements can be made. We expect revised equations will be available in 1980 from the analytical effort being conducted at the University of Missouri by Professor Loyalka.