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Dr. E. Igne
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Seventh Water Reactor Safety Research Information Meeting, 5 to 9 November 1979.

Dear El:

In the enclosure I have summarized what I heard in the subject meeting. Delay in submitting this report was caused by my effort to obtain a complete set of copies of papers presented. This turned out to be not possible inasmuch as I was informed that some presentations had not been put on paper.

Material presented at this meeting indicates that significant progress has been made in NDE technology, materials fracture toughness characterization and in thermal fuel behavior program. There were many thermal hydraulic computer codes under development last year and there continue to be many this year. The real progress in this area however, is the program ability to adjust to unanticipated needs in the field of analysis (TMI-2, SBA).

I have also enclosed, under separate cover, a large (but incomplete) set of material presented.

Thank you for the opportunity to attend this meeting.

Very truly yours,

Zenons Zudans
Senior Vice President,
Engineering

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encl.
Summary

U.S.C.
Papers

cc: Dr. Paul G. Shewmon
Ohio State University

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from: Z. Zudans
31 January 80

SEVENTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING
November 5-9, 1979

Summary of Selected Presentations

5 Nov 79

S. Levine, NRC, Director, RES

Special Session for the first day was dominated by the discussions on and around TMI-2. Mr. Levine announced his retirement and introduced Dr. Budnitz as 99% certain successor (confirmed at this time). NRC research for the future to emphasize need for results, faster data feedback into regulation process. Conservatism filed upon conservatism - a situation unfavorable to everybody, needs to be resolved by research to bring things back to reality.

Since 1978, improved safety research has been conducted: alternative containments studied and RHR studied by risk assessment methodology. As a consequence of TMI-2 accident, it was realized that sequences leading to core damage (beyond that of DBA) must be studied. Better computer programs are needed to study small breaks, both fast running, simple and slower complex codes. Better systems for plant status information display are being studied now by Sandia. Similarly, integrated reliability program has also been started. Improvements in feedwater system reliability are in process. Greater emphasis on creating event trees for all operating reactors, followed by simplified fault tree analysis looking for obvious risk outliers - will be the subject of a course to be given at the ANL.

Dr. L. S. Tong, NRC, RSR

Gave highlights of Water Reactor Safety Research (WRSR) achievements in 1979. Five main objectives of RSR were quoted as follows:

1) Accident Defection and Prevention.

2) Separate Effects Understanding and Modelling.

3) Reactor Systems Testing and Evaluation.

4) Analytical Tool Development, Assessment and Application.

5) Accident Consequence Evaluation and Mitigation.

Several instances of accomplishments were given in each of the above areas.

D. Eisenhut, NRC, DOR, stressed need to better utilize operating reactor experience. Stated that reactor operations and licensee qualifications will be improved significantly in a long run. In short term, Technical Support Center, a place near control room, will be set up to follow the plant state vector and advise operators on consequences of actions. Discussed results of short/long term lessons Learned Reports and reasons for licensing pause.

W.V. Johnston, NRC, I&E, discussed I&E report on TMI-2 (NUREG-0600), NSAC 1 and other reports by EPRI on TMI-2. Conclusions - systems approach to licensing process is needed with broader definition of safety related items, better coordination of research with regulatory process, improvements in simulation and simulations systems are needed.

METALLURGY AND MATERIALS RESEARCH (MMR)

C.Z. Serpan, Jr., NRC, RSR, gave brief overview of research in MMR.

P.C. Paris did not make a presentation on Tearing Instability.

J.P. Gudas, NSRDC, discussed validation of Computer Interactive Unloading Compliance Test Method. Substantial progress was indicated. Results indicate that 15-20% side grooves eliminate crack front tunnelling and saturates Tearing Modulus. Development of Key Curve Analysis (see next page) has lead to estimating procedures allowing correction of J_I for crack growth, and inter-laboratory comparability of results has been achieved. His tests are not multispecimen tests. Fracture characterization of primary reactor piping alloys is in progress by this method.

J.A. Joyce, U.S. Naval Academy, discussed Key Curve Analysis of Ductile Shelf Fracture Toughness and demonstrated that Key Curve Analysis technique of Paris* correlates with the results by other methods. The principal advantage of this method is that it develops the J-R curve directly from the load displacement record without requiring a separate determination of crack length by unloading compliance. Further advantage of this method is that it can be used with load-displacement curves developed for any rate of displacement. Key Curves are developed by testing a series of geometrically similar subsize specimens of the same steel. He developed Key Curve surface

$$\frac{PW}{3b^2} = F\left(\frac{a}{W}, \frac{\Delta}{W}\right)$$

For ASTM A533B steel and utilized this surface for J-R curve development. J-R curves thus develop compare well with these developed with Unloading Compliance method. The Key Curve method allows to make a correction of J-R curves to account for crack extension as well.

F.J. Loss, NRL, (paper was presented by Many), described a program on fracture toughness determination of A533-B submerged arc weld deposit with irradiation followed by two full cycles of post irradiation annealing and reirradiation. This research is intended to identify the potential merits of the annealing for control of radiation induced embrittlement in reactor pressure vessels. Showed that the proposed ASTM multispecimen procedure for determination of J_{IC} is inappropriate (J-R exhibits power law and not a linear behavior as presumed earlier).

W. H. Culler, NRL, Influence of Critical Variables on Fatigue Crack Growth Rates of PVP materials in PWR Coolant Environments.

In these tests single material (A-508 Steel) was used in environment with fixed chemistry (PWR) to study the effect of the load rise time (1-30 min), hold time (1-60 min), temperature (93-288°C) on crack growth rate. All data

*Ernst, H., Paris, P.r., Rossow, M. and Hutchenson, J.W., "The Analysis of Load Displacement Relationships to Determine J-R Curve and Tearing Instability Material Properties," ASTM STP 677, 1979.

fall into one of two clearly defined categories: one essentially on ASME Section XI air environment line, the other (a factor of 3 to 5 above the first) approximately midway between air and water lines.

S. Moore, ORNL, made no presentation of his paper.

R.J. Eiber, BCL, discussed reevaluation of criteria for postulating cold leg break. He had performed deterministic analysis of cold leg piping. Several questions (such as model boundary conditions, vibration loads used, how margins were defined, neglect of high stresses in sensitive areas) were not satisfactorily discussed to support the conclusions reached by the speaker. Author proposed leak rate model based on crack path length and equivalent "hydraulic diameter."

D. Tomasko, Sandia, discussed analytical prediction of two-phase jet loads. The objective of this work is to develop an approximate engineering model for characterization of two-phase jets from piping cracks. The work today uses large computer codes (BEACON/MOD2-INFL, CSQ, TRAC-P1A) and experimental data from KWU. Gave analysis vs. experiment evaluation concluding that TRAC-P1A gave the best agreement with the experiment. Importance of modelling friction effects was identified.

G.H. Powell, Univ. Cal./Berkeley, discussed non-linear code for pipe whip analysis. This is a 3 year program with a year gone by. This code uses special purpose Data Base Manager for all data storage and recovery modular in design. In principle, it is extension of an existing code (FACTS - developed by SSD, Inc., Berkeley, Cal.) into non-linear dynamics field. Initially, code is developed for batch processing (minicomputers in mind), later for interactive I/O. Beam and shell modelling of piping and support structures is used.

Yet, another computer program added to already rich selection of claimed "perfect" structural codes!

G. Irwin, Univ. of Maryland, spoke on the subject of Standard Crack Arrest Specimen and Analysis. He pointed out that differences of opinion existed since the very beginning of this program. Gave a description of proposed static and dynamic (K_{I_a}, K_{I_D}) K value determination. This is a joint effort by 28 Laboratories. Results will be documented in a NUREG document and ASTM Std test procedures. Professor Irwin feels plausible explanations for determination of crack arrest toughness will be forthcoming.

R.D. Cheverton, ORNL, gave a discussion of experimental results in behavior of deep flaws in a thick-wall cylinder under thermal shock loading. Conclusion was that, until further characterization, LEFM and the developing crack arrest methodology are valid for deep and shallow flaws under severe thermal shock conditions.

J.W. Bryson, ORNL, discussed photoelastic and FE studies on determination of K_I distributions along arbitrarily shaped flaw fronts at pressure vessel nozzle corners (see recent HSST quarterly report for details).

In addition to photoelastic studies a computer code NOZ-FLAW is being developed for direct evaluation of K along flaw fronts. This will employ 3-D crack tip elements with proper square root and inverse square root singularities for displacements and stresses, respectively. A distinct feature of this singular element is that the K appears as an unknown and is solved for directly. Results of NOZ-FLAW have been compared to BIFIG (EPRI program for nozzle flaw analysis) and to photoelastic tests with good agreement for certain types of flaws and not so for others. NOZ-FLAW computer program should be in ANL Software Center around March, 1980.

H.R. Hawthorne, NRL, discussed test results intended to evaluate notch ductility degradation of low alloy steels with low-to-intermediate neutron fluence exposures. An objective was to obtain experimental assessment of the level of conservatism in current embrittlement projection methods (Reg. Guide 1.99). It was found that Reg. Guide 1.99 may be overly conservative in projecting upper shelf reduction at fluences less than 5×10^{18} n/cm^2. A lesser degree of conservatism for transition temperature. In general, increased embrittlement with increased fluence (in test reactor) was observed (no saturation). Also showed annealing to be beneficial.

W.N. McElroy, HEDL, gave progress report on LWR pressure vessel surveillance dosimetry improvement program. This is a large multinational program involving various professional organization. The major benefit of this program is in significant improvement in the accuracy of the assessment of the remaining safe operating lifetime of LWR pressure vessel. ASTM Practice for use of DPA (displaced atoms per atom of displacing agent) is now out as ASTM Std. Other results of this research are ASTM Std drafted: 5 practices, 5 guides, 5 methods.

F.B.K. Kam, ORNL, discussed Validation of Flux and Spectrum Predictions for Pressure Vessel Wall Environment and indicated that factor of 2 to 3 differences appear in calculations by different Laboratories. He described PCA-PV and ORR-PV facility work at ORNL, designed to reach the neutron exposure determination accuracy goal of 10-15%. This is part of the program described by McElroy in previous paper.

D. Pachur, Nuclear Research Center, Julish, F.R.G., presented results of irradiation and annealing of three (3) different steels - subjected to different neutron fluences at different irradiation temperatures. He found saturation to be present at various combinations of conditions, not being clearly identified which set of prerequisites causes saturation to occur more readily. He identified four (4) governing processes (characterized by activation energies) affecting materials properties occurring during annealing (graphically distinguished by different time dependence of phenomena at various temperatures) and stated that irradiation behavior can be described (but not understood) by growth of these processes. A very effective presentation and valuable contribution towards study of saturation phenomenae.

R. Clark, PNL, described work in Steam Generator Tube Integrity program. Chemically and mechanically defected specimens have been produced and tested. Has problems in defining defect before test. Has developed empirical correlation for burst data for uniform thinning, elliptical wastage and EDM slot with corresponding error bands. Discussed plans for dissection and examination of a retired Surry-2 Steam Generator (NRC controlled, joint NRC-EPRI sponsored project).

W.L. Clarke, GE, described progress with EPR method for detection of sensitization in stainless steel. This technique is accepted now internationally, ASTM adaption is in process. Pa (C/cm^2) activation charge density is indication of sensitization. For Pa < 5.0, almost no sensitization. This technique is in use now by GE and a number of steel suppliers. Field services will also be available, $4,000/instrument, 1 hour to perform the test in field.

D. Van Rooyen, BNL, stress corrosion Cracking Predictability in Steam Generator tubes. He stressed the need for capability to predict SG tube cracking. Started with the Arrhenius relationship for crack rate to activation energy

$$r = K \exp^{-Q/Rt},$$

developed various further crack rate relationships and described the use of slow strain rate experiments for the calibration of the proposed rate equations. Could not find conclusive evidence that microstructure relates to cracking. Cold work even to a small degree can be quite adverse to stress corrosion cracking in pure H_2O. Crack velocity increases with temperature, has confidence in method of extrapolation to high temperature. An interesting presentation on fundamental phenomena in SCC.

D.W. Prine, GARD, discussed quantified flow detection during welding by use of acoustic emissions (AE) methodology. Flow characterization in Laboratory environment was demonstrated feasible 100% for cracks, 5-80% for porosity and less for other defects. Today's status: hardened design and fabrication complete, software done and debugged, ASTM recommended practice submitted for ballot. Still to do - final software debug, field testing.

P.H. Hutton, PNL, described a program aimed at using acoustic emission (AE) device for continuous monitoring of reactor pressure boundary. In this AE use patterns are developed for normal operation. These patterns are later compared into actual operating plant produced patterns for flaw detection purpose. Correlation by energy range appears to be more successful than correlation by peak time.

L.L. Yaeger, Daldalean Assoc., described incipient crack detection technique (non-destructive) based on internal friction damping (IFD-NDE) BWR piping equivalents are used to quality and test the method and feasibility has been demonstrated an 4" SS pipe. In principle, this method consists of feeding a signal in pipe wall at a resonant frequency and measuring the attenuation from which the energy dissipated in one cycle and additional damping capacity are computed.

V. Van den Broek, Univ. of Michigan, described research progress in Ultrasonic nondestructive testing of pressure vessels. His work is concerned with development of a Synthetic Aperture Focusing System for Ultrasonic Pulse-Echo Flaw Evaluation. This is a sophisticated approach, critical need being that of processing data in the computer in real time. In principle, the method consists of defining how signals at two locations affect each other, than shifting the signal from the second location and phasing it with that of the first location. Signal at the first location gets stronger in this way. Signals are used to develop images, holographic terminal is employed. There is hope to get imaging in real time (speed of computer processing is the determining factor). Conversion of ultrasonic signal amplitudes into color image gives better quantitative information. Sophisticated device.

C.V. Dodd, ORNL, described improved multi-frequency eddy current (MFEC) test and analysis for in-service inspection of steam generator tubes. The main objective of this development is to overcome problems associated with single frequency EC devices where tube support clearance varies and/or wall thickness changes. This MFEC system is able to sort these things out. The method is rather simple (but effective): develop instrument response as a function of a number of governing parameters (wall thickness, support gap, diameter), invert these responses (least square) to produce these parameters in terms of response functions. It works.

R. Clark, PNC, Proposed improved test standard for steam generator tube inspection for use with single frequency eddy current devices. Better correlation with known defects was obtained by using these new standards.

L. Becker, PNL, gave first year report on a five (5) year program on ultrasonic inspection reliability and the probability of flaw detection for primary piping systems. First year's effort resulted in setting up a test matrix designed to separate the influence of six components affecting the flow detection reliability: 1) flaw, 2) instrument with coupling geometry, 3) ASME Section XI, 4) test procedure, 5) records, and 6) baseline. Survey of industry has identified in general similar practices, but variable ultrasonic transducers, which affect the results significantly.

SESSION A-2 - CODE DEVELOPMENT PROGRAM

D.R. Liles, LASL, TRAC is an advanced, BE computer program for analysis of LWR accidents. It features nonhomogeneous, multidimensional fluid dynamics, nonequilibrium thermodynamics, heat transfer and reflood models and flow-regime dependent constitutive equations, all geared to better describe physical phenomena in a reactor.

TRAC hydrodynamics and heat transfer improvements this year are:

- improved thermodynamic representation
- changed inter-phase condensation model
- two-fluid one dimensional version (working).

During last year eliminated drift-flux in favor of 2-fluid model. Priorities now are for fast running version and inclusion of non-condensible gas. Constitutive equations are now made dependent on flow regime.

D. Mendel, LASL, continued with presentation of TRAC heat transfer package including fluid-to-wall heat transfer, gap conductance model and reflood heat transfer model. An interesting feature is automatic renoding in reflood regime.

Discussion of goodness CHF correlations used developed after presentation. It was pointed out that both local and Iloeje's CHF correlations do not work well (later predicts rewetting at wrong location) may be because of axial heat flux. Boiling length correlation for CHF does a better job than the above two in cases of nonuniform (axially) heat flux. I recommend to consider correlation on the bases of local and adjacent point temperatures (would introduce heat flux gradient effect). LASL may look into that.

P.B. Bleiweis, LASL, gave a number of illustrations showing how TRAC performs in analyzing 2D/3D facility experiments. In particular, analysis of UPTF for its design purposes and instrumentation was done by TRAC (German facility - Upper Plenum Test Facility, to be operational in 1982). Also, cylindrical core test facility (CCTF), (Japan) and slab core test facility (SCTF) (Japan), have been analyzed by use of TRAC.

J. Ireland, LASL, presented TRAC analysis of TMI-2. This was a rather complex model and contained numerous assumptions about inputs received from components interfacing with the model. In spite (or because) of these, results appeared to match actual TMI-2 results well for key variables.

FUEL BEHAVIOR RESEARCH PROGRAM, 8 Nov. 1979

J.A. Gieseke, BCL, discussed TRAP-MELT computer code designed to calculate the transport and deposition of aerosol by superheated steam flow through various control volumes. Radionuclide history thus can be traced with TRAP-MELT. At this time baseline calculations have been made for selected accident sequences. Model improvement are needed and lack of definite input data suggest large output uncertainties.

D.A. Powers, Sandia, reported on results of studies on high temperature melt/concrete interaction for the purpose of better understanding and use in risk assessment. This program is made up of experimental and analytical parts. Experimental program is to identify qualitatively important physical and chemical processes. Prototype melts and concretes at realistic conditions were used in large-scale test. Data collected in these tests were: 1) rate and extent of concrete erosion, 2) rate of metal oxidation, 3) rate of melt solidification, and 4) concrete temperature.

CORCON computer code is being developed as the comparison part to experimental work. No results as yet. There is cooperative effort with KfK, KWU to evaluate this code.

G. Hofmann, KfK, Long Term Coolability of a partially blocked core. A 90% reduction of coolant flow area was assumed, for a large number of adjacent channels, extending over the distance between two spacer grids, as a limiting condition for this experiment.

Experiments showed that even with such large blockage for realistic power levels and flow conditions the center of the blocked core does not overheat.

How realistically does this model represent a real core blockage? I feel the bypass flow, having no heat generation, might produce more optimistic result than the reality would be.

H.J. Zoile, INEL, gave an overview of PBF test results, related to fuel behavior under off-normal accident conditions. Four test series have been performed during the last 12 months:

1. Reactivity Initiated Accident Series (RIA)
2. LOFT Lead Rod Tests (LLR)
3. LOCA Series Tests (LOCA)
4. Thermocouple Response Test Series (TRTS)

Detailed discussion of failures and modes of failure was given for these tests.

T.R. Yackle, INEL, gave LOC-3 results on influence of internal pressure and prior irradiation on deformation of Zircaloy cladding. Tests indicated that irradiated rod balooned more than not irradiated and pressure made little difference on balooning. It was noted that when the rod failed, it contracted (due to anisotrophy of Zircaloy). LOC-3 is one of the five (5) FBF LOC program tests designed to evaluate fuel cladding behavior during cold leg break.

Brouhton, EG&G, presented evaluation of influence of cladding surface thermocouples (TC) on thermal behavior of nuclear fuel rods during LOCA. It was established that external TC affect CHF time and temperature achieved during blowdown. Also, reflood comes sooner with TC's. When steam is passing TC's it has no effect.

P.E. MacDonald, EG&G, reported on light water reactor fuel response during RIA experiment. Maximum fuel rod enthalpy limit of <280 cal/gUO_2 is imposed by NRC on commercial reactors. Complex analysis is used to estimate the effects of postulated RIAs in LWRs. Existing fuel behavior data have been obtained a long time ago in the Special Power Excursion Reactor Test (SPERT) and Transient Reactor Test Facility (TRTF) programs. PBF facility is being used now to obtain data on irradiated fuel under coolant conditions typical of hot-startup in a BWR.

Findings from unirradiated rod test indicate the failure threshold of fuel rods tested under BWL hot startup conditions is slightly higher than observed in SPERT (205-225 cal/gUO_2); failure at 260 cal/gUO_2 and BWR hot startup conditions is as severe as previously observed in SPERT (loss of coolable geometry). Preirradiated rod tests indicate the failure threshold was less than for unirradiated fuel and was about 140 cal/gUO_2. Failure at 285 cal/gUO_2 and BWR hot startup conditions is more severe than previously observed in SPERT and NSRR (Japan). Upon failure at 285 cal/gUO_2 the rods swelled and blocked the flow channels.

WORKSHOP ON PLANS FOR IN-REACTOR FUEL BEHAVIOR EXPERIMENTS (8 Nov 1979 3:00 pm)

P.E. MacDonald, EG&G, PBF Experimental Program. Future test plans are designed to provide detailed data on fuel behavior in accident environment. PBF core can be operated to simulate typical accidents and off-normal conditions such as power-cooling - mismatch (PCM), LOCA, RIA, Operational Transients without scram (OPTRAN), SBL. The main objective is to provide validation data base for fuel rod analysis codes. Review of work beginning in 1974 was given with account of what has been learned so far. Future RIA test objectives include investigation of coolability of a bundle of preirradiated rods, testing of commercially irradiated BWR/6 fuel at alternate licensing criteria and identification of fuel behavior in accidents that are beyond design basis accidents (it appears desirable to move towards tests of larger damage - when material is fluidized there should be no length effect on heat transfer).

J. Randles, ISPRA, reported on SUPER-SARA program in ESSOR. He stressed the point that no other facility is available at present or in the foreseeable future providing the range of parameters for in-pile LOCA simulation offered by ESSOR. Until now this has been an Italian program. It is being proposed now to make it into a European Communities program and then into an International program. (Financial pressure induced change.)

Characteristic features of SUPER-SARA test program include 36-52 PWR rods, 2 m level length; 25 BWR rods, 2 m long; electrically heated parallel twin test section to verify thermal hydraulics; once through test section; well isolated from outside; range of hot cells adjacent to reactor.

Tentatively 10 tests are planned for SUPER-SARA.

Mohr, PNL, presented NRU Reactor (Chalk River Nuclear Laboratories) Experiment Plans. Will simulate heatup, reflood, quench, refill for full size rods. The objective of these tests to learn about thermal hydraulics, materials deformations, temperature distribution in full length bundle, quench front velocities, etc. Thermal hydraulic tests to be completed October, 1980, Material Tests to be completed April, 1982.

R.D. MacDonald, AECL, discussed in-reactor fuel transient behavior experiments at Chalk River Nuclear Laboratories (CRNL). This program is designed to verify particular aspects of the computer codes (ELOCA) used to assess fuel performance. Other objectives are to confirm that there are no unmodelled phenomenae (unsuspected mechanism) which would significantly affect fuel performance, study element failure mechanisms, measure fission product release. At present, there are two vertical in reactor loops, one for LOCA and the other for loss of regulation accident (LORA) (similar to US PCM program). Future plans call for a horizontal test loop in NRU reactor to be available in 1984 (if approved by management).

TFBP REVIEW GROUP MEETING, 9 Nov 79

Phil McDonald opened this session by giving a menu of things to come: a sampling of FBP program results, how to relate these to real fuel behavior in core at various thermal hydraulic conditions, library of Zr structures, molten fuel movement, RIA power measurement, flow instabilities and SB test program requirements.

F.S. Gunnerson, EG&G, presented evaluation of CHF propagation during PCM-5 test and comparison with PCM single rod test. Conclusions were: Film boiling sequence within bundles appear to be random. No rod-to-rod film boiling propagation. Individual rod DNB/post-DNB behavior correlated to power/coolant variations. Single rod-to-rod interaction as suspected:

$$\begin{array}{c} \text{quench and} \\ \text{rewet (cause)} \end{array} \longrightarrow \begin{array}{c} \text{onset of} \\ \text{DNB (effect)} \end{array}$$

Return to nucleate boiling (RNB): several heat transfer modes observed, rewet temperatures predictable. No rod-to-rod failure propagation. Conditions at onset of DNB on center rod consistent with single rod data.

D.E. Owen, EG&G, presented characterization and interpretation of PBF test rod cladding microstructure. Showed Zirconium-Oxygen equilibrium phase diagram and gave an interpretation of Zircaloy microstructure. Stated that it is a function of temperature and time, and showed various typical states with corresponding microstructures and observed modes of failure. Broad range of parameters was covered by the characterization:

$$920°K \leq T \leq 1085°K \quad (\alpha)$$
$$1085°K \leq T \leq 1250°K \quad (\alpha+\beta)$$
$$1250°K \leq T \leq 1700°K \quad (\beta)$$
$$1700°K \leq T \leq mp \quad \text{(melting point)}$$

and estimation of cladding temperatures provided.

Further tests are planned (out of pile) to duplicate microstructures at 1700°K and higher. Finally will come out with a complete microstructure catalog for Zr. Apparently, KfK has devised method which allows them to measure the clad temperature to within ± 3°K (in terms of microstructure β-fraction).

M.S. El-Genk, EG&G, presented results of molten fuel movement research (RIA-ST-4 test, 700 cal/gUO_2 total energy deposition). In addition to experiment, analysis was performed using SINGLE Code. The test described indeed did melt the fuel and moved it all over the place! Details of physical model and analysis boundary conditions and simplifying assumptions were discussed. Some of the conclusions indicate, that freezing of molten debris on the inner surface of the shroud wall is governed by temperature and Zircaloy volume ratio in molten fuel debris, radiation cooling has minor effect on freezing process.

R.K. McCondell, EG&G, reviewed PBF RIA power calibration techniques and results. Measurement results by different method were discussed, such as

- core power chamber data
- self powered neutron detector (SPND) data
- shroud flux wire data
- core flux wire data
- Burnup analysis data

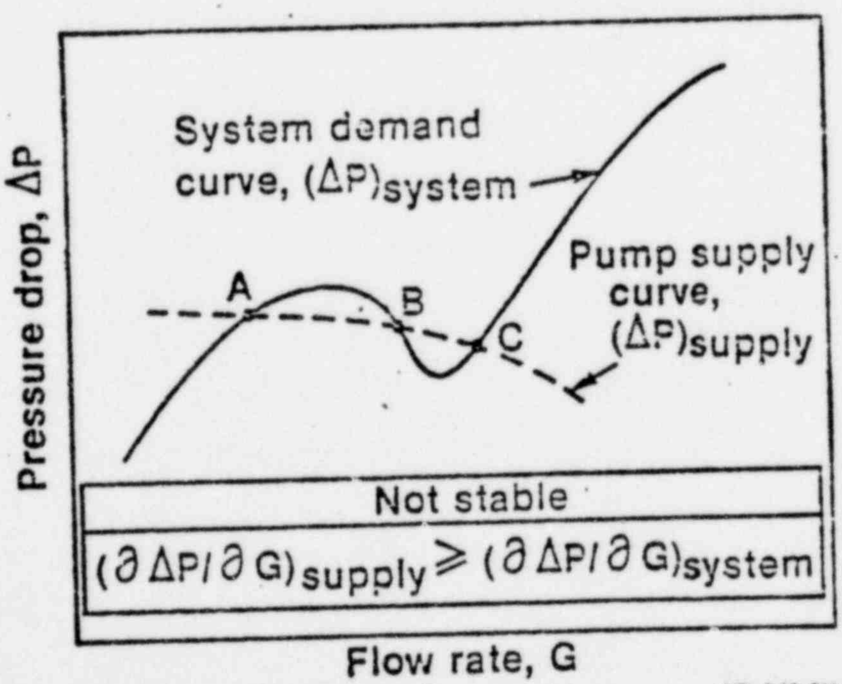
A committee had been formed to evaluate various measurement methods. It was concluded that none of the methods used were unreliable and a long list of recommendations for improvements was made. Some of these recommendations have been completed, some are in progress. As a result of this work the energy deposition uncertainty between methods has been reduced to about ± 7%. A tabulation of the best estimate total fuel energy data for various tests was given along with the standard deviation.

F.S. Gunnerson, EG&G, discussed rewet behavior, two-phase instabilities and test PR-1.

For typical rewet behavior: increase flow/decrease power results in rewet. Anomalies observed in PCM-2, PCM B-1 RF, PCM-5.

The type of instability considered was density wave (most common), Ledinegg (flow excursion), and relaxation (chugging). Hedinegg instability is associated with rapid increase/decrease of flow in cases where system demand curve is non-monotonic (see Figure).

Flow Excursion (Ledinegg) Instability



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