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ACRS-1649

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MINUTES OF THE
ACRS SUBCOMMITTEE MEETING ON
EMERGENCY CORE COOLING SYSTEMS
JUNE 19-20, 1979
WASHINGTON, D.C.

The ACRS Subcommittee on ECCS held a meeting on June 19-20, 1979 in Room 1046, 1717 H Street, N.W., Washington, D.C. The purpose of this meeting was to review the small break analysis performed with the B&W model subsequent to the Three Mile Island 2 accident and the NRC Staff review of the B&W work. The second day of the meeting was devoted to a review of the proposed FY 1981 budget for ECCS related Safety Research and the proposed Supplemental to the 1980 budget. The notice for the meeting appeared in the Federal Register on Monday, June 4, 1979. A copy of the notice is included as Attachment A. A list of meeting attendees and a meeting schedule are included as Attachments B and C. No written statements or requests for time to make oral statements were received from members of the public.

Executive Session - Open

Dr. Plesset, Subcommittee Chairman, opened the meeting at 8:40 and indicated that it was being conducted in accordance with the Federal Advisory Committee Act and the Government in the Sunshine Act. Dr. A. Bates was the Designated Federal Employee for the meeting.

During a short executive session the Subcommittee members and consultants discussed the topics to be reviewed at the meeting. Dr. Plesset indicated that the Research Budget should be examined to determine the appropriateness of the items being funded in FY 1981 and requested in the supplemental 1980 budget. Dr. Catton indicated that he had some questions on the B&W models used in the steam generator and in the hot leg as they relate to heat transfer, bubble growth, and natural circulation.

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Meeting with the NRC Staff

Dr. Rosztoczy, NRC, indicated that subsequent to the Three Mile Island accident the Staff met with B&W, Westinghouse, and CE a number of times to review small break calculations. The area of concentration has been on very small breaks and natural circulation cooling. Also reviewed were the vendor guidelines for emergency procedures and operator retraining programs. In response to a question, Dr. Rosztoczy indicated that all of the Appendix K conservatisms apply to the small break analysis, the 1.2 ANS decay heat value is the most significant of the conservatisms.

Dr. Rosztoczy indicated that there were some differences in opinion regarding the advantages and disadvantages of tripping the reactor coolant pumps following initiation of the LOCA. Discussions are continuing on this point. The other change made to the operating reactors has been the provision for reactor trip prior to actuation of the relief valves for the various transients which produce over pressure events. It is not believed that this will cause any problems with system operation, W and CE already run their plants this way, B&W plants have reversed their PORV and Trip setpoints. Dr. Zudans questioned the wisdom of replacing 148 PORV actuations with 148 reactor trips; the mechanical effects on the primary system may be more undesirable for the reactor trips. Dr. Rosztoczy indicated that efforts were being made to reduce the number of reactor trips also.

Mr. Michelson suggested that one alternative to the PORV set point changes might be the automatic closure of the down stream block valve following actuation of the PORV.

Mr. B. Sheron reported on the NRC review of the B&W calculation of small breaks and the capability of natural circulation heat removal (see Attachment D). The concerns arose from a report prepared by Mr. C. Michelson, TVA, on very small breaks in B&W reactors where repressurization of the primary system may occur if natural circulation cooling is lost or the steam generators are not available. B&W responded to these concerns and others in a topical

Report to the NRC on May 7, 1979. In the B&W plants very small breaks may not provide adequate enthalpy flux at the break to cope with core decay heat. If the sytem depressurizes far enough to produce a steam bubble in the primary system, natural circulation cooling to the steam generators will be lost. The system will then repressurize until the enthalpy flux at the break provides adequate heat removal, or natural recirculation is restored. Similar effects may be produced in U-tube steam generators. Mr. R. Jones, B&W, indicated that the higher elevation of cold water in the raised loop plant provides a greater potential to reestablish the natural circulation. The lowered loop plants go through one cycle of repressurization and water solid natural circulation, the following depressurization ends up in a reflux boiling mode with condensation of steam in the steam generators when the secondary side water level exceeds the primary side water level. The raised loop plants go through three cycles of repressurization. The calculations are sensitive to how the hot leg and steam generator are modeled. The bubble rise, the separation of steam and water, and the interface levels in the steam generator are very important in calculating heat transfer. The actual heat transfer coefficients in the steam generator are not as important as are the water and steam elevation and interface areas.

Dr. Catton indicated some concern over the models used for the B&W Steam generators. Dr. Rosztoczy indicated that he believed they were adequate.

Dr. Zudans suggested that the several instances of small breaks in actual reactors provided a good source of information for benchmarking calculations of small breaks. Dr. Rosztoczy indicated that the TMI-2 accident was probably the only event that was adequate for doing detailed calculations, other small LOCA's have occurred at low power or during start-up. Also not all of the desirable information is available.

Drs. Catton and Theofanous indicated that they believed that some thought and hand calculations like those contained in the Michelson report would be more adequate than large codes for analyzing some small break situations. Dr. Catton indicated that with the big codes one may loose sight of important effects which are buried in the mass of computer calculations. Small changes in the model and nodalization might have significant effects; if time is devoted to thinking about the processes and doing some hand calculations insights unavailable from computer codes may be developed.

A number of questions arose as to the use of a volume balance for the calculations for B&W small breaks. Both mass and enthalpy balances are needed to determine losses from the primary sytem and possible core uncovery.

Mr. Sheron indicated that the calculations show that a condensing surface is reestablished in the steam generator prior to any core uncovery. At the Davis-Besse plant the high head safety injection pumps shut off (zero discharge) at 1600 PSIA. Under these conditions repressurization could prevent additional HPIS water from being added. The make up pumps can provide some water above 1600 PSIA but they are not safety grade. Auxiliary feedwater is required in order to depressurize the sytem below 1600 PSIA and allow operation of the HPIS pumps.

The Staff agreed that pressurizer level indication is not a good indication of system water inventory for certain pressurizer breaks. They expect to see the pressurizer drain for cold leg breaks. Mr. Michelson and others cautioned the Staff that detailed analysis of the pressurizer heat losses are necessary to determine if it will drain for all small breaks. Small changes in temperature are important; if the upper head of the vessel or the hot leg are several degrees hotter than the pressurizer it will not drain due to the loop seal.

Mr. Sheron reported that they reviewed the possible system effects of the operator isolating a break during depressurization. The Staff have concluded that the system is adequate for this type of situation provided that make up flow or HPIS is provided to depressurize the system and auxiliary feedwater is available. Various opertor actions could cause problems. The procedures should provide adequate guidance on what to do. Negative guidance (i.e. prohibitions of certain action) are generally not included in procedures.

The Staff indicated that they have looked at tests of ECC injections in LOFT and Semiscale for water hammer-pressure oscillation type effects and have not seen any indication of potential damaging pressures. RELAP calculations show oscillations which have been removed with control of the water packing problems that the code had.

A review of sources of non-condensable gases indicates that the primary system would have to depressurize to about 400 PSIA before enough hydorgen and nitrogen would come out of solution to fill the hot leg U bends. About 190 cubic feet of gas are involved in order to fill the U-bends. The CE and W plants require similar volumes to fill the U-tubes in the steam generators. For very small breaks in the B&W plants the Staff does not see the accumulators emptying and there should be no nitrogen injection. There should not be problems with non-condensables in the hot legs. This is provided that the core is not uncovered (as in TMI-2) and there is no metal-water reaction.

Mr. Michelson cautioned that under some circumstances operator actions to isolate a break could produce undersirable consequences such as injection of non-condensable gases, repressurization of the primary system, core uncovery, or interruption of natural circulation.

Mr. R. Audette, NRC, reviewed the results of the small break calculations performed by B&W following the TMI-2 accident. The recent calculations were preformed for break sizes between .07 ft^2 and .0005 ft^2 with various combinations of auxiliary feedwater, no auxillary feedwater, HPIS flow with one or two pumps going and different initiation times, and with and without the primary coolant pumps running. Results show that for breaks sizes below 0.02 ft^2 repressurization will occur. In one case, with no auxiliary feedwater, with 1.2 ANS decay heat, a stuck open PORV, 1 HPI pump, and the RCPs off the core was shown to uncover; if 1.0 ANS was used it did not uncover. The Staff concluded that if auxiliary feedwater was provided at 20 minutes core uncovery would be prevented for both raised and lowered loop plants for breaks smaller than .02 ft^2. Other conclusions are shown in Attachment E-1.

Questions revealed that the pump heat added to the primary system was not accounted for in the analysis. This may have an effect as it is comparable to the decay heat after extended periods of time.

Mr. N. Lauben reviewed the audit calculation the NRC Staff had performed for the B&W small break LOCA. Two cases were calculated by EG&G using RELAP 4/ MOD 7. The first case was a 0.01 ft^2 break with Auxiliary Feedwater (AFW) delayed 20 minutes, the second case assumed normal AFW and one HPI pump. In the first case the HPI actuation point is reached and repressurization occurs, the second case produces voids in the candy cane of the hot leg and loss of natural circulation occurs. The Staff concluded that RELAP and Craft produce differences in key varibles that are calculated and that additional study is needed. Core uncovery was not calculated with either code.

Mr. B. Wilson, NRC Staff, reviewed the new B&W Guidelines given to the utilities for the operator procedures for coping with a small break (Attachment F). Two items were discused, the first was the development of the procedures by B&W and the Staff review of them, the second are related to the utility implementation of the procedures for their operators.

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The guidelines consisted of symptoms and indications of small breaks, immediate action to be carried out, precautions, and followup actions. One major item was to provide criteria underwhich HPI flow could be terminated. This was based upon maintaining hot and cold leg temperatures 50°F below saturation or having the LPIS operating at 1000 gpm for 20 minutes. Recommendations for action to follow given reactor coolant pump operation or tripping and availability of auxiliary feedwater were also made. A number of consultants recommended that the simulator training of operators be reviewed to observe the mistakes made by operators. These than could be used to augment the training program and procedures as things one does not do.

Mr. Wilson indicated that the Staff is looking at the human factor in control room engineering and there may be future NRC requirements or recommendations in this areas in the future.

Mr. B. Wilson reviewed the NRC Objectives used when reviewing utility procedures. The intent is to assure conformance with vendor guidelines and workability for the operators. The problem areas identified when reviewing procedures included lack of adequate knowledge of small break phenomenon, utility exceptions to the guidelines, and the adaption of the new procedures to existing procedures.

A number of questions arose as to how an operator determines whether these is a break and how big it is. The amount of make up flow and HPIS flow as well as system de-pressurization are important criteria in the determination.

Mr. B. Bogar reviewed the operator training that has been carried out since TMI for B&W plants. It has included review of the TMI-2 accident, TMI-2 simulater training, formal classroom training, written exams, an NRC audit of the training, follow up training and requalification training.

Facility changes and procedural changes have been incorporated in the training as well as review of the auxiliary/emergency feedwater system operation.

Subcommittee members and consultants suggested that when grading the tests consideration should be given to the seriousness of the mistake when grading - not knowing an answer may not be as serious as doing the wrong thing - especially when procedures can be looked up. Also discussed was the information available to the operator and how they are able to decide the exact plant situation so as to respond properly.

During a summary session of the meeting various consultants expressed their opinion in the need for additional work. Several felt that additional thought is needed to assure that there are not other accident sequences and operator mistakes that could lead to TMI-type situations.

TMI-2 is being thoughly studied, but other possible accidents should also be looked at just as thoroughly.

The meeting was recessed at 5:45 p.m. to reconvene the following day.

Summary of the Meeting June 20, 1979

Topic	Presentation	Transcript Pages
1. Opening Comments	M. Plesset	262-265
2. Introduction	T. Murley	265-305
3. Separate Effects Research		305-375
a. SEMISCALE	A. Serkiz	305-331
b. Blowdown/Reflood Heat Transfer	A. Serkiz	331-349
c. 2D/3D Program	G. Bennett	349-362
d. Model Development Experiments and Technical Support	A. Serkiz	362-375
4. LOFT Research	D. McPherson	376-396
5. Analysis Development Branch	S. Fabic	396-441
6. Closing Comments	M. Plesset	441-462

ECCS Meeting
Wednesday, June 20, 1979

Opening Comments - M. Plesset (Transcript pages 262-265)

The House Committee preparing the budget for the NRC has proposed three changes to increase the utility of the ACRS report.

1. The ACRS should prepare its report in accordance with a schedule that permits it to be used by the NRC in preparation of the fiscal year 1981 authorization request.
2. The ACRS should prepare a clear statement of research priorities including specification of projects to be added or dropped.
3. The ACRS should include a discussion of the specific manner in which the NRC's reactor safety research projects are expected to affect the NRC's reactor regulations.

Introduction - T. Murley, NRC (Transcript pages 265-305)

The FY 80 supplement for safety research budget was presented. This budget ($29.8 million) is for:

1. better understanding of transient and small LOCA accidents (13.4 million)
2. enhanced operator capability (3.8 million)
3. plant response under accident conditions (5.1 million)
4. post mortem examination and plant recovery at TMI (2.7 million)
5. improved risk assessment (3.1 million), and
6. improved reactor safety (1.7 million).

(A breakdown of each of the above six categories is presented in the meeting handouts.)

The budget for the LOCA-ECCS program is scheduled to rise and then peak in 1981 as scenarios that go beyond design basis accidents are considered.

Current plans are to upgrade SEMISCALE by adding a secondary system. It is tentatively planned to upgrade the Two-loop test apparatus (TLTA) facility to study BWR transients and small LOCAs.

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I. 's planned that a data bank be established for each operating reactor that would include applicable computer codes which could be employed for calculations following an accident or event at a plant. This probably will be done at National Labs.

An important area is enhanced opeator capability. The diagnostic system used at Halden (Norway) cou'd be installed at LOFT. This diagnostic system is being considered for the Groven-Rhinefeld reactor in Bavaria.

The auxiliary feedwater pumps are not considered engineered safety features (ESFs). A study of ESFs should consider heat removal without the availability of the turbine condenser after the reactor scram.

Dr. Catton requested further information on the funding for the better understanding of transients and small break LOCA events. Items discussed in addition to the detailed breakdown of this area (provided in the attachments) were: re-examination of RETRAN, secondary system treatment for TRAC, study of the CE code IRT, addition of a noncondensable gas model to various codes, incorporation of COBRA into TRAC, modification of the SSC code (for liquid metal systems) to handle water reactors, and the creation of immediately operable computer code decks for each operating reactor. Calculations performed on postulated event trees should provide a greater understanding of transients and small break LOCA events.

A review of all interconnected systems should be included in the improved reactor safety area of this budget. This interaction step should include an examination of environmental aspects and a determination of the adequacy of instrumentation between interconnected systems.

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Dr. Murley presented the FY 1981 budget (please see handouts). Dr. Murley will provide a list of priorities for this budget and the supplement before mid-July.

The German computerized monitoring system is not really a great advance in the state-of-the-art since this type of plant monitoring is performed in refining operations. The nuclear industry, however, is generations behind in terms of control and display diagnostics. This system could be implemented for LOFT. The Germans have been working on it since about 1971.

The need for larger scale test results was pointed out. The use of commercial power reactors for specific tests (e.g. small break LOCAs) could be accomplished without hazard to the plant. Special accident following instrumentation could be installed in commercial reactors to aid the study of abnormal transients.

Separate Effects Research - (Transcript pages 305-375)
SEMISCALE - A. Serkiz, NRC (Transcript pages 305-331)

The budget for the SEMISCALE is 6.7 million (FY 80) and 10.5 million (requested for FY 81). The upgrading of the facility includes the addition of once-through and U-tube steam generators estimated at 3.5 million.

ECC bypass studies are being phased out and will be essentially completed by FY 1981. Technical support is being redirected into instrumentation and diagnostics.

SEMISCALE has been set up for upper head injection (UHI) experiments. The facility will be modified with additional insulation since it has been postulated that excessive heat from the downcomer walls has led to voiding associated with core back-flow and high core steaming rates. This modification will correct surface area to volume ratio effects. A test run at this facility considering the relationship of pressurizer level and core level has been accomplished. This Subcommittee will get copies of this report.

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The Proposed schedule for the SEMISCALE includes:

1. small break testing through August
2. rerun test S-06-7 to test the hot wall effect with the new insulation installed
3. feedwater transients during late fall-early winter.

All break effects (e.g. pressure, 2D-3D) cannot be studied at any one facility. They must be integrated using codes and engineering judgement. Subcommittee members questioned the relation of SEMISCALE to a full sized LWR. Questioned raised concerning pressure effects, 2D-3D effects, externally mounted thermocouples on fuel pins, scaling, and two-phase pheonmena are under study. The shift from the large break to the small break LOCA require some redirecting of the facilities.

Mr. Ebersole pointed out potential problem areas (vortex formation in an accumulator, activating valves under dynamic heads, discharge of nitrogen into a primary loop from the accumulators, and UHI).

Mr. Michelson pointed out situations where large breaks are changed to small breaks as in the case wherein isolation valves fail to close fully. Some PWRs have loop isolation valves.

Blowdown/Reflood Heat Transfer - A. Serkiz, NRC (Transcript pages 331-349)

Each of the LOCA/ECCS tests facilities have their own capabilities. SEMISCALE is a PWR facility whereas TLTA is a BWR facility. FLECHT-SEASET has a testing capability of up to 2300°F and was designed to primarily handle flow blockage effects during reflood. Preliminary analysis of steam cooling tests at FLECHT show a 50% enhancement over current models.

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Several current Oak Ridge articles have addressed the concerns that electrical heaters cannot really simulate the thermal hydraulic boundary conditions or simulate fuel pins. Mr. Serkiz has the information on these articles.

The ATWS program is related to LOCA analysis. A BWR reflooded with cool, clean water requires the insertion of control rods. The rod drive control insert and exhaust tubes are in direct line of the muzzle blast from LOCAs inside the drywell. There may be a mechanism for tube damage that will degrade rod insertion capability. It is difficult to model nuclear feedback effects with electrically heated bundles.

<u>2D/3D Program</u> - G. Bennett, NRC (Transcript pages 349-362)

The 2D/3D program is an international cooperative program (Japan, Federal Republic of Germany, USA). The program analyzes steam binding that may occur during a PWR LOCA, flow distribution effects within the core, flow hydrodynamics in the downcomer and upper plenum during core uncovering, and natural circulation.

The facilities involved are the German upper plenum test facility (full-scale vessel), the Japanese cylindrical test facility (2000 electrical rods, full height core, 4 loops), the Japanese slab core facility (full radial capability), and instrumentation and analytical work (via TRAC) at Los Alamos. The German PKL facility (full height core - 3 loop capability) is not formally part of the program but has been used to test instrumentation for the other facilities. These facilities are in various stages of construction and testing. The program is currently on schedule.

New instrumentation (e.g. level detectors and improved pressure, temperature instrumentation) to be developed in this program should be usable in operating LWRs instead of only in research.

This program has considered the possibility of the study of an evaporative-reflux-condensation steam generator if natural circulation is determined insufficient in U-tube steam generator systems.

Three dimensional effects such as countercurrent flow, cocurrent flow, cross flow, and the chimney effect may be expected in this program (The chimney effect occurs when steam is generated in the central region of the core, rises up the hot channels, and then falls back along the core periphery).

Model Development Experiments and Technical Support - A. Serkiz, NRC (Transcript pages 362-375)

The researchers and institutions involved in developing, benchmarking, and verifying heat transfer correlations were listed. Various examples of technical support (e.g. advanced instrumenation) for the 2D/3D project were presented.

Additional knowledge is needed in the following areas; the relationships and characteristics of mass, volume, and energy transport through orfices and relief valves, the performance of boilers at reduced pressure and reduced heat transfer surfaces, and the determination of heat sinks for the break and secondary system.

LOFT Research - D. McPherson, NRC (Transcript pages 376-396)

The LOFT program has completed all non-nuclear tests in the L-1 series, two nuclear loss-of-coolant experiments in the large break series, and an isothermal small break loss-of-coolant experiment. A new series of experiments will focus on system response to off-normal conditons. An assessment of conventional process information and a comparison to the special research instrumentation will be conducted.

A two-phase flow calibration facility will be completed at LOFT in 1980. It will have the capacity to calibrate flows in full size pipes.

The development of audio detection devices (e.g. loose parts monitoring, onset of boiling) was addressed since the Subcommittee members didn't recognize it as part of the NRC instrumentation development program at the LOFT facility. The development of these instruments is not currently ongoing in the 3D program. The 3D program will be looking at acoustic monitoring.

Preliminary studies at LOFT on noise in ion chambers and self-powered neutron detectors indicate that two-phase flow in the downcomer can be monitored.

<u>Analysis Development Branch</u> - S. Fabic, NRC (Transcript pages 396-441)

A faster version of TRAC (TRAC-PF1) applicable to PWRs will be available in late 1979.

Since the complex codes are unwieldly for extensive mapping of great varieties of accidents, two suggestions have been offered - the development of hybrids and very fast digital routines (intelligent shortcuts using microprocessors).

The codes presently in use and in development do not provide for operator action. Some preprogram options are available which allow us to control, for example, when a pump will stop or valve will open.

The aspirator in the main feed line inlet which brings the water into the downcomer of B&W steam generators is not modeled in any of our or B&W's codes. This may be important in natural circulation calculations. The auxiliary feedwater in B&W steam geneators enters the generator and is spread across the tubes by a perforated baffle. This heat transfer mechanism is not treated in either RELAP or TRAC and is probably the reason our comparisons have not been good.

Comparisons with the reconstruction of the TMI accident and computer code evaluation have generally not been good. A few problem areas have been identified:

1. importance of the steam geneator as a part of the thermal hydraulic system
2. two-phase flow
3. flow through relief valves
4. the mass of auxiliary feedwater into the steam generators
5. homogeneous model assumption, and
6. model of the "candy-cane" loop configuration.

Key indicators (e.g., events, quench times, time to empty the pressurizer) measured in LOCA tests and calculated by TRAC were in reasonable agreement. This relationship is used to assess the LOCA codes.

Closing Comments - M. Plesset (Transcript pages 441-462)

The SEMISCALE facility simulated a PORV TMI transient in a Westinghouse U-tube steam generator design. The surge line, however, was piped to match the B&W loop seal design. The pressurizer remained full during the small break LOCA (break at the top of the pressurizer). This indicates that the pressurizer may give false indications of liquid inventory in all PWRs. The small size of the pressurizer scale surge line and volume raise questions as to the meaning of results obtained at SEMISCALE for small break LOCAs.

Members of the Subcommittee felt that the activities presented at this meeting in response to TMI-2 were not organized. Risk assessment could be employed to order the priorities for experiments in order to organize the efforts at these facilities.

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Notes:

(1) For additional details a complete transcript of the meeting is available in the NRC Public Document Room, 1717 H St., N.W., Washington, D.C. 20555, or from Ace-Federal Reporters, Inc., 444 North Capitol Street, N.W., Washington, D.C. 20001.

(2) Materials provided to the Subcommittee at this meeting are on file in the ACRS Office. In general, these materials include:

a. Budget summaries for FY 1980 and FY 1981 of all projects discussed.

b. Program outlines and a schedule of testing for the facilities discussed.

c. ECCS, LOCA, ATWS Computer Code availability.

d. ECCS Computer Code Applications.

e. NRC Sponsored Calculations of the TMI-2 Accident Scenario.

f. Nodalization models of reactor systems used in TRAC.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee on Emergency Core Cooling Systems (ECCS); Meeting

The ACRS Subcommittee on Emergency Core Cooling Systems will hold an open meeting on June 19–20, 1979 in Room 1046, 1717 H St., N.W., Washington, DC., 20555, to review ECCS models for small breaks in Babcock and Wilcox reactor systems. The Subcommittee will also review the proposed FY–81 NRC budget figures for ECCS-related research activities. Notice of this meeting was published in the Federal Register on May 24, 1979.

In accordance with the procedures outlined in the Federal Register on October 4, 1978 (43 FR 45926), 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, June 19 and Wednesday, June 20, 1979, 8:30 a.m. until the conclusion of business each day.*

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 and to formulate a report and recommendations to the full Committee.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of Babcock and Wilcox, the NRC Staff, and their consultants, pertinent to the above topics.

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 therefore can be obtained by a prepaid telephone call to the Designated Federal Employee for this meeting, Dr. Andrew L. Bates (telephone 202–634–3267) between 8:15 a.m. and 5:00 p.m., EDT.

Dated: May 29, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-17083 Filed 6-1-79; 8:45 am]

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ATTACHMENT A

ACRS SUBCOMMITTEE MEETING ON
EMERGENCY CORE COOLING SYSTEMS
JUNE 19-20, 1979
WASHINGTON, D.C.

ATTENDEES LIST

ACRS

M. Plesset, Chairman
J. Ebersole
I. Catton
K. Garlid
W. Lipinski
C. Michelson
R. Shumway
H. Sullivan
T. Theofanous
T. Wu
F. Zaloudek
Z. Zudans
A. Bates, Staff*

*Designated Federal Employee

BNL

K. R. Perkins

B&W

R. C. Jones
J. J. Cudlin
E. R. Kane

PICKARD, LOWES & GARRICK

M. Schwartz

MPR

D. M. Chapin

CREARE

J. Block

WYLE LABS

R. Cummings

NRC

R. Audette
B. Sheron
Z. Rosztoczy
P. Norian
M. Lauben
W. Lyon
R. F. Smith
R. Anderson
R. Lee
A. L. M. Hon
L. Thompson
Y. Hsu
G. L. Bennett
G. D. McPherson
L. Shotkin
W. H. Beach
R. Hoskins
W. D. Beckner
G. Rhee
A. Seriz

KEPCO

K. Ota
K. Noda

BBR

K. O. Layer

EPRI

R. H. Leyse

DUKE POWER

G. Swindlehurst

ACE-FEDERAL

L. Weinschel

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ATTACHMENT B

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TENTATIVE MEETING SCHEDULE
ACRS ECCS SUBCOMMITTEE
JUNE 19-20, 1979

Tuesday, June 19, 1979 - B&W Small Break Analysis

I.	Executive Session - M. Plesset	8:30 - 8:45
II.	NRC Staff Introduction - Z. Rosztoczy	8:45 - 9:00
III.	NRC Staff Review of Michelson Concerns - B. Sheron	9:00 - 10:45
IV.	Small Break Analyis - R. Audette	10:45 - 12:30
	- Lunch -	12:30 - 1:30
V.	Review of New B&W Guidelines for Small Breaks - B. Wilson	1:30 - 2:30
VI.	NRC Methods for Review of LOCA Procedures at B&W Plants - B. Wilson	2:30 - 3:30
VII.	Audit of Operator Training - B. Bogar	3:30 - 4:30
VIII.	Additional Questions and Adjourn	4:30 - 5:00

Wednesday, June 20, 1979 - NRC Water Reactor LOCA/ECCS Research

I.	Executive Session - Opening Comments - M. Plesset	8:30 - 8:45
II.	Introduction - T. Murley	8:45 - 9:00
III.	Separate Effects Research Branch a. Semiscale - A. Serkiz b. Blowdown/Reflood Heat Transfer - A. Serkiz c. 2D/3D Program - G. Bennett d. ECC Bypass - A. Serkiz e. Model Development Experiments - A. Serkiz/S. Fabic f. Technical Support - A. Serkiz/G. Bennett	9:00 - 11:00
IV.	LOFT Research Branch - D. McPherson	11:00 - 12:30
	- Lunch -	12:30 - 1:30
V.	Analysis Development Branch - S. Fabic a. Systems Codes b. Component Codes c. Code Assessment d. Code Sensitivity	1:30 - 3:00
VI.	Closing Comments - Adjourn	3:00 - 3:30

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CONCERNS

1. ACCEPTABILITY OF INTERMITTANT NATURAL CIRCULATION
2. TIME DELAY IN TRANSITIONING FROM NATURAL CIRCULATION TO POOL BOILING
3. PRESSURIZER LEVEL WAS NOT CORRECT INDICATION OF WATER LEVEL IN CORE
4. CONSEQUENCES OF SMALL BREAK ISOLATION/REPRESSURIZATION
5. PRESSURE BOUNDARY DAMAGE DUE TO BUBBLE COLLAPSE
6. BREAK ENERGY NOT REPRESENTATIVE OF CORE EXIT ENERGY
7. EFFECT OF NON-CONDENSIBLE GASES (FROM CE SYSTEM 80 REPORT)

SUMMARY

- NO DISAGREEMENT ON PHENOMENA DESCRIBED BY C. MICHELSON.
- CONCERNS UNDERSCORED IMPORTANCE OF NATURAL CIRCULATION FOR DECAY HEAT REMOVAL DURING SMALL BREAKS.
- B&W HAS PERFORMED DETAILED ANALYSES TO ADDRESS CONCERNS.
- RESULTS SHOW PHENOMENA OCCUR, BUT THAT DECAY HEAT REMOVAL IS NOT UNACCEPTABLY IMPACTED.

B&W SMALL BREAK GENERIC STUDY

BREAK	AFW	HPI	EC PUMPS	LONG-TERM COOLING
.07 FT^2	OFF	2	OFF	390 SEC.
.02	OFF	"	"	650
.01	1 @ 20 MIN.	"	"	1730
.01	OFF	2 @ 20 MIN.	"	2774
LOFW	2	1 HPI	ON	1000
PORV	"	"	OFF	1000
PORV (ANS*1.2)	OFF	"	"	--
PORV (ANS*1.0)	OFF	"	"	4700
.01	2	"	"	4900
.01 (ASYM)	1	"	"	4975
.005	2	"	"	5000
.01 (DB-1)	2	"	"	6000

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CONCLUSIONS

1. AFW AT 20 MINUTES PROVIDE CORE COVERING FOR LOWERED AND RAISED LOOP PLNTS FOR BREAKS SMALLER THAN .02 FT^2.

2. HPI ONLY AT 20 MINUTES PROVIDE CORE COVERING FOR LOWERED LOOPS FOR BREAKS SMALLER THAN .02 FT^2.

3. 1 HPI TRAIN PROVIDES CORE COVERING FOR STUCK PORV IN LOWERED AND RAISED LOOPS.

4. HOT LEG BREAKS BOUNDED BY RESULTS FOR COLD LEG BREAKS DUE TO ACTION OF VENT VALVES.

5. SINGLE STEAM GENERATOR OPERATION IS ADEQUATE TO MAINTAIN CORE COVERING FOR SMALL BREAKS.

IF THE HPI SYSTEM HAS BEEN ACTUATED BECAUSE OF A LOW PRESSURE CONDITION, IT MUST REMAIN IN OPERATION UNTIL ONE OF THE FOLLOWING CRITERIA IS SATISFIED:

1. THE LPI SYSTEM IS IN OPERATION AND FLOWING AT A RATE IN EXCESS OF 1000 GPM IN EACH LINE AND THE SITUATION HAS BEEN STABLE FOR 20 MINUTES.

OR

2. ALL HOT AND COLD LEG TEMPERATURES ARE AT LEAST 50^{0} BELOW THE SATURATION TEMPERATURE FOR THE EXISTING RCS PRESSURE. IF THE 50^{0} SUBCOOLING CANNOT BE MAINTAINED, THE HPI SHALL BE REACTIVATED.

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SMALL BREAK ACCIDENTS

RCP's	FW	RECOMMENDED ACTION
1. YES	YES	STOP ONE RCP PER LOOP. USE OTSG's TO COOLDOWN AT 100°/HOUR.
2. YES	NO	MAINTAIN MAX. HPI FLOW. STOP ONE RCP/LOOP. OPEN PORV IF RCS PRESSURE INCREASES. RESTORE FW ASAP.
3. NO	YES	COOLDOWN WITH NATURAL CIRCULATION. IF UNABLE, ATTEMPT TO RESTORE FORCED CIRCULATION WITH RCP's. IF UNABLE, CYCLE PRESSURE BETWEEN 2300 PSIG AND 100 PSI ABOVE OTSG PRESSURE.
4. NO	NO	OPEN PORV. MAINTAIN HEAT REMOVAL PATH FROM HPI THRU BREAK AND PRESSURIZER. RESTORE RW AND ECP's.