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SUBCOMMITTEE MEETING MINUTES
NATURAL CIRCULATION HEAT REMOVAL
MARCH 26 1980

DATE ISSUED: 30 May 80
ACRS-1724

The ACRS Subcommittee on Natural Circulation Heat Removal met on March 26, 1980 in Room 1046 at 1717 H Street, N. W., Washington, D. C. The purpose of the meeting was to review the present state of knowledge on various modes of natural circulation heat removal in PWR's which might be important to decay heat removal during accident or transient conditions. Notice of the meeting was published in the Federal Register on March 10 and March 19, 1980 (Appendix A). Dr. Andrew Bates was the Designated Federal Employee for the meeting. A list of meeting attendees is attached as Appendix B. No written statements or requests for time to make oral statements were received from the public.

NRC STAFF PRESENTATION ON CURRENT REQUIREMENTS FOR AND KNOWLEDGE OF NATURAL CIRCULATION IN COMMERCIAL PLANTS

Mr. W. Hodges indicated that Regulatory Guide 1.68 required testing for natural circulation (water solid) during the start up phase of each reactor unless tests at a similar reactor could be shown to be equivalent. The major problem in running the tests is in measurement of the natural circulation. Flow is too low to be detected so verification is dependent upon establishment of ΔT values across the core and steam generator as well as a system heat balance. Verification of natural circulation is dependent upon examination of the system temperatures, verification of stable hot and cold leg temperatures with a ΔT across the core and steam removal from the steam generators. All of the current requirements relate to natural circulation under water solid conditions (bubble in pressurizer); there are no requirements for any tests under two-phase conditions and such testing would be avoided as possibly damaging to the reactor fuel. Tests have been run under steady state conditions and transient conditions from high power. The startup tests are initiated at about 5% power with a reactor trip and a coolant pump trip. A number of events have also been initiated at near full power due to loss of power to the reactor coolant pumps or other transient events leading to pump trip. In either case natural circulation has been proven to function. In response to questions the Staff indicated that the tests have been initiated from RCP coast down conditions and there have not been tests of the self starting capability of natural circulation. The Staff indicated that they did not see any potential

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problem in W and CE designs since the Steam Generator Elevation was sufficiently above the core, on B&W reactors the secondary side water level would have to be high enough to initiate the natural circulation flow.

Dr. B. Sheron indicated that the NRR Staff had a number of unanswered questions with regard to two-phase natural circulation. Certain small break LOCAs require heat removal by the steam generators. Calculations indicate that two-phase loop flow and reflux boiling should take place; however, no experimental data is available. NRR has asked Research to experimentally verify that two-phase natural circulation will take place. At the present time significant problems are not foreseen; however, the capability of natural circulation to remove decay heat when large enough quantities of non-condensable gases are introduced into the primary system is still of concern. The Staff indicated that gas may arise from accumulator nitrogen, dissolved gases, core oxidation, or failure of the UHI cut off valves. For very small breaks where the steam generators are required to remove decay heat the pressure does not decrease to the accumulator set point, for breaks which do turn on the accumulator the RCS pressure must fall to about 150 psia before significant quantities of N_2 enter the system. At this pressure the LPIS and RHR system can provide adequate core cooling. The Staff does not believe that there is a significant problem with non-condensables, unless there is significant core oxidation or failure of the UHI valves which should block N_2 injection. (See also W presentation on UHI to ECCS Subcommittee on March 25, 1980).

The Staff also reviewed the guidelines given to the plant operators for verifying natural circulation. The guidelines require verification of reactor trip, start of the diesels if offsite power is lost, verification of natural circulation for core ΔT values, verification of steam dump setpoint, starting of auxiliary feedwater pumps, and actions to prevent a too rapid cooldown of of the primary system. Instrumentation available to the operator includes T hot, T cold, pressurizer level, pressure, incore thermocouples, secondary steam pressure, and steam generator level. The procedure also contains corrective actions to establish natural circulation if the verification procedures fail to indicate natural circulation.

Mr. R. Baer outlined the special low power tests to be performed at the Sequoyah plant during start up. At the time of the meeting the Staff indicated that they still did not have detailed information on the procedures to be used in conducting the tests or a detailed safety analysis for the tests. The proposed tests include:

(1) Natural circulation under normal conditions
(2) Natural circulation (N/C) with simulated loss of offsite AC
(3) N/C with loss of pressurizer heaters
(4) Effect of steam generator isolation on N/C
(5) N/C at reduced pressure
(6) Cooldown capability of charging/letdown system
(7) Simulated loss of all AC
(8) Establishment of N/C from stagnant conditions
(9.a) Forced circulation cooldown
(9.b) Boron mixing and cooldown in N/C.

The criteria in choosing the tests to be performed include:

(1) The tests should provide meaningful information
(2) The tests should provide supplemental operator training
(3) The tests should not pose undue risk to the public
(4) The risk of plant damage should be low, and
(5) The post test radiation levels should not preclude implementation of the future TMI-2 requirements.

The Staff noted that none of the tests is expected to enter a region of two-phase fluid flow. In response to questions Mr. Baer indicated that TVA would be adding some supplemental instrumentation for the tests but not a lot. The consultants and Subcommittee members expressed concern over the adequacy of instrumentation and the value of any data (or lack of data) that might be obtained from the tests. The consultants also questioned how meaningful the tests would be if they did not include two-phase flow. Mr. Baer indicated that the Staff viewed the tests more as a method of operator training than as needed experimental data.

Mr. Baer reviewed the NRC Staff's present positions with regard to feed and bleed heat removal. The B&O Task Forc concluded that the capability of plants with low head HPI pumps to depressurize with the PORVs to initiate HPI in time to preclude core damage is highly uncertain and that a diverse decay heat removal path independent of the steam generators was desirable. They recommended consideration be given to increased PORV capacity, high head HPI pumps, or high pressure residual heat removal systems.

At the present time there are no NRC requirements for plants to have a feed and bleed capability. There are a number of different opinions within the NRC Staff as to whether it is better to have feed and bleed capability, a high pressure RHR system, or increase the reliability of the auxiliary feedwater system.

Mr. J. Ebersole indicated that the reliability of natural circulation would limit the extent to which improvements in the secondary side would be useful.

The Staff indicated that for cases where there were no non-condensable gases they did not forsee any interruption of natural circulation. In the case of UHI injection you would not be injecting N_2 until the system pressure was below about 600 psia. At that point the steam generator would be acting as a heat source rather than a heat sink. W has also submitted a calculation that the Staff is reviewing (See ECCS meeting March 25, 1980) which indicates that even with the UHI gas there is sufficient area and volume in the steam generator tubes to maintain the heat transfer process. The Staff is still reviewing this submittal and the Subcommittee consultants indicated that they have some questions with regard to the degradation of the heat transfer coefficient.

The Staff stated that there could be a number of problems with feed ar[illegible]ed even in reactors with high head HPI pumps; these include lack of safety [illegible]ade controls to PORVs, reliability of the valves under two phase or liquid flow, and the dumping o[illegible] large amounts of water into the containment. (There is currently an NRC research program which is following tests which are planned by EPRI on PORV and safety valve discharges. These tests should address some of the unanswered questions on flow rates under different conditions of the fluid).

Dr. S. Fabic reviewed the present status with regard to the ability to calculate natural circulation under water solid two-phase loop flow, and reflux conditions. He also discussed the experimental data needs for assessment of the codes calculational ability. Experimental knowledge of hot and cold leg flow rates, stratification, fluid densities, liquid and vapor velocities, froth levels, and fluid temperatures in both the primary side and secondary side are important. Some data is available from LOFT Tests L3-1 and L3-2 for comparison to the calculations. Review of the data shows the need to take into account stratified flow, counter-current flow, and fluid conditions upstream of the break orifice. Knowledge is needed on the flow regimes which will occur in the steam generator tubes and in the hot leg in order to properly model the flow. Present codes (RELAP 4) are basically homogeneous flow, more advanced codes (TRAC and RELAP 5) should be able to eventually handle the two-fluid flow involved with counter current conditions, stratification, and refluxing. TRAC will also eventually include a non-condensable gas field. Dr. Fabic indicated he felt that with the recent experimental uncertainties and lack of needed measurements he could not say whether the codes were doing a good job or a bad job in calculating two-phase natural circulation.

In response to a question from Dr. Theofanous, Dr. Fabic indicated that the present vendor evaluation models which do not include two-fluid models were probably inadequate for calculation of some of the small break LOCA sequences. He indicated that Research was using both advanced codes and codes which were similar in model (but best estimate) to the vendors codes in order to gain knowledge as to the important phenomena that need to be modeled.

Dr. Catton recommended that consideration be given to doing a simple experiment to determine the effect of non-condensables on reflux boiling. He indicated that such a test could be done as a gross basis to determine when enough gas had been injected to shut off the refluxing, this could be done using a heat balance without need to measure where the gas went or other parameters that might be of interest to the advanced code development people. The details could be worked out in the longer term.

Dr. Fabic pointed out a number of modeling problems with the codes. These include the need to model stratified flow in pipes, counter-current steam-liquid flow in steam generator tubes and the hot leg, ECC mixing at the injection nozzle, and system leakage flows and bypass flows. All of these items have been shown to be important in LOFT tests.

The Subcommittee consultants and members generally agreed that decisions were going to have to be made as to the level of detail necessary in the code, otherwise, with a purely microscopic approach the experimental program would never yield sufficient data to accurately calculate the desired transients.

Dr. H. Sullivan, NRC Research, reviewed a number of programs in the Systems Engineering Branch that will be looking at natural circulation. In response to a question he indicated that some tests would be conducted from stagnant conditions to verify that natural circulation would start by itself without any forced flow. Dr. Sullivan indicated that present calculations with RELAP 4/MOD 7, and RELAP 5 have indicated that natural circulation under water solid conditions is not a problem. The codes also predict that the reactors can enter into a two-phase natural circulation.

The Research program will be conducting a number of tests in experimental facilities to study natural circulation under various conditions. This will include single phase, two-phase - one component, and two-phase - 2 component (non-condensable and water) tests. Test results will be used to study the adequacy of the calculational methods. The goal is to obtain codes in which we have adequate confidence so that the NRC can use them to perdict the behavior of large PWRs.

In repsonse to questions Dr. Sullivan indicated that studies were being conducted to determine how much non-condensable gas might be injected or produced in a reactor system. This would then be used as a basis for running tests in Semiscale or other facilities to determine whether or not it can casue problems in interrupting heat removal through the refluxing process. Dr. Sullivan also indicated that they have not planned any tests on feed and bleed since NRR has not requested them. They do have the capability of doing some tests if requested. (NRR subsequently asked for some tests on loss of AC and DC power and core boil off of water. A preliminary test has been conducted in Semiscale.) Tests will be conducted to look at the heat transfer in the core and steam generator, phase separation, transitions between phases, mass injection and removal, imbalance between steam generators, two-phase mixture levels, and counter-current flow effects. A number of tests will be conducted in various facilities including LOFT, Semiscale, Flecht-Seaset, PKL, CCTF, THTF, TLTA, and in various university programs. A number of the facilities are scaled to be full height and they are are capable of running at a variety of test pressures. The univeristy program involves looking at steam generator U-tube behavior using transparent U-tubes for flow visualization.

One major problem that Research is working on with each of the experimental facilities is the adequacy of the instrumentation. In many cases improvements are needed for the phenomena that need to be measured, i.e.; low flow, counter-current flow, and two-phase flow.

The Subcommittee members and consultants expressed their opinions on a number of items that they thought should be investigated furhter. Mr. Bender indicated that the capability to depressurize the primary system would solve many of the problems associated with needing natural circulation or feed and bleed. Dr. Lienhard indicated that he felt more could be done with proper use of dimensionless groups in scaling of flow between the small experiments and larger systems.

The meeting adjourned at 5:30 pm.

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Additional information is available from the meeting transcript available in the NRC Public Document Room, at 1717 H Street, N. W., Washington, D. C. A complete copy of the slides presented is on file at the ACRS office with the reocrd copy of the minutes.

ATTENDEES LIST

ACRS MEMBERS

M. Carbon
M. Plesset
H. Etherington
J. Ebersole
M. Bender

ACRS CONSULTANTS

J. Lienhard
V. Schrock
Z. Zudans
I. Catton
T. Theofanous

ACRS STAFF

A. Bates, Designated Federal Employee
E. Abbott, ACRS Fellow
G. Young, ACRS Fellow

NRC STAFF

W. Hodges, RSB
B. Sheron, RSB
H. Ornstein, AEOD
T. Speis, RSB
P. Norian, AB
R. Baer, DPM
R. Auluck, STD
H. Sullivan
R. Landry
Z. Rosztoczy, NRR
C. Graves, NRR

Others

R. Turk, GE
J. Gloudermans, B&W
D. Lambert, TVA
R. Olson, Balt Gas & Elec

**Anticipated Transients Without Scram (ATWS).* March 26, 1980, Washington, DC. The Subcommittee will meet with representatives of the NRC Staff to continue discussion of the proposed resolution of ATWS. Notice of this meeting was published March 11.

**Three Mile Island, Unit 2 Accident Action Plans.* April 1–2, 1980, Washington, DC. An Ad Hoc Subcommittee will review Draft 3 of NRC NUREG–0660, "Action Plans Developed as a Result of The TMI–2 Accident." Notice of this meeting was published March 17.

**Babcock & Wilcox Water Reactors.* April 8, 1980, Washington, DC. The Subcommittee will complete its review of the NRC Staff study to determine whether construction should be halted on certain B & W plants because of sensitivity of the once-through-steam generator (OTSG) to feedwater transients.

**Three Mile Island, Unit-2 Accident Implications.* April 9, 1980, Washington, DC. An Ad Hoc Subcommittee will discuss implications of the TMI–2 Accident as they relate to construction permit applications.

**Regulatory Activities.* April 9, 1980, (8:45 am), Washington, DC. The Subcommittee will review regulatory guides and revisions to existing regulatory guides; also, it may discuss pertinent activities which affect the current licensing process and/or reactor operation.

**Power and Electrical Systems.* April 9, 1980 (2:00 pm), Washington, DC. The Subcommittee will consider the nuclear data link (NDL), presently being developed by NRC as part of the TMI–2 Accident Actions Plans.

**Reactor Safety Research.* April 9, 1980, (4:00 pm), Washington, DC. The Subcommittee will review the NRC office of Nuclear Regulatory Research response to ACRS recommendations to Congress on NRC research (NUREG–0657); also, the FY–82 budget review and preparation of the ACRS report to NRC will be discussed.

**General Electric Test Reactor (GETR).* April 18–19, 1980, Sunol, CA. The Subcommittee will continue its review of the geologic and seismologic aspects of the GETR plant site. Also, other matters related to the NRC Order to Show Cause may be discussed. Notice of this meeting was published February 22.

**Licensee Event Report (LER).* April 22, 1980, (8:30 am), Washington, DC. The Subcommittee will review plans of the new NRC Office of Analysis and Evaluation of Operational Data, and NRC action in response to the ACRS LER Report, (NUREG–0572).

**Site Evaluation.* April 22, 1980, (1:00 pm) Washington, DC. The Subcommittee will review the proposed Emergency Planning Rule (10 CFR, Part 50), published in the Federal Register December 19, 1979 and NUREG–0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Notice of this meeting was published February 22.

**Concrete and Concrete Structures.* April 22–23, 1980—Rescheduled from March 25–26, Washington, DC. The Subcommittee will review "user needs" in structural engineering and the way in which these needs have been met.

**Reactor Radiological Effects.* April 23, 1980, Washington, DC. The Subcommittee will review the radiological protection programs at nuclear power plants.

**Natural Circulation and Feed and Bleed Heat Removal.* April 24, 1980. An Ad Hoc Subcommittee will continue its review of information presently available on natural circulation and feed and bleed heat removal systems. Also, areas where inadequate information is available will be examined, and planned NRC (or other) tests that will be run to obtain needed information will be studied and the possible need for other tests will be explored.

**Metal Components.* April 24–25, 1980, Oak Ridge, TN. The Subcommittee will meet with representatives of the NRC and ORNL Staffs to review the Heavy Section Steel Technology (HSST) Program.

**Reactor Fuel.* April 29, 1980, Washington, DC. The Subcommittee will discuss the status of reactor fuel development and use in commercial LWRs, including particular problems or unusual events that occurred in the past year.

**Reliability and Probabilistic Assessment,* April 30, 1980, Washington, DC. The Subcommittee will continue its evaluation of the need to develop quantitative safety goals for nuclear power plants and consideration of the actual form these goals may take and what they should accomplish.

**Consideration of Class 9 Accidents.* May 6, 1980, Washington, DC. The Subcommittee will discuss consideration of low probability, high consequence accidents (including core melt) as part of the licensing process. The Integrated Fuel Melt Research Program will also be discussed.

ACRS Full Committee Meetings

April 10–12, 1980.

A. *Floating Nuclear Plant—Core ladle conceptual design.
B. *Sequoyah Nuclear Power Plant, Unit 1—Augmented Low-power test program.
C. *Anticipated Transients Without Scram (ATWS)—Proposed plant modifications.
D. *Pressurized Water Reactors with Once-Through-Steam Generators (OTSG)—Dynamic response to transients.
E. *NRC Action Plans to Implement Lessons Learned from the TMI–2 Accident—Proposed implementation of long term items.
F. *Code of Federal Regulations (10 CFR, Part 50, Appendix K, ECCs Evaluation Models)—Proposed changes in techniques to calculate clad-ballooning.
G. *Proposed Replies to NRC Commissioner V. Gilinsky re ACRS Report dated Dec. 11, 1979 on the pause in licensing and proposed use of the nuclear data link (NDL).
H. *Meeting with NRC Chairman J. Ahearne and other NRC Commissioners who may be present re ACRS activities as an NRC advisory committee, including ACRS comments on the NRC Action Plans.

May 1–3, 1980—Agenda to be announced.

June 5–7, 1980—Agenda to be announced.

Dated: March 14, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 8334 Filed 3-18-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Emergency Core Cooling Systems, Change of Date

The March 26, 1980 meeting of the ACRS Subcommittee on Emergency Core Cooling Systems has been rescheduled to be held on March [illegible]. All items pertaining to this meeting remain the same as announced in the Federal Register March 11, 1980 (45 FR 15[illegible]).

Dated: March 14, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-8335 Filed 3-18-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Ad Hoc Subcommittee on Natural Circulation Heat Removal; Change of Date

The March 25, 1980 meeting of the ACRS Ad Hoc Subcommittee on Natural Circulation Heat Removal has been rescheduled to be held on March 26, 1980. Notice of this meeting was announced in the Federal Register March 10, 1980 (45 FR 1534?).

In addition, the title of this Ad Hoc Subcommittee has been changed to "ACRS Ad Hoc Subcommittee on Natural Circulation and Feed and Bleed Heat Removal."

All other items regarding this meeting remain the same as announced in cited notice.

Dated: March 14, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-8336 Filed 3-18-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Concrete and Concrete Structures; Change of Date

The meeting of the ACRS Subcommittee on Concrete and Concrete Structures announced in the Federal Register March 10, 1980 (45 FR 15346) has been rescheduled from March 25–26, to April 22–23, 1980. All other items regarding this meeting reamin the same as announced in the cited notice.