ISSUE DATE: April 17, 1980
ACRS-1729

MINUTES OF THE ACRS B&W WATER REACTORS SUBCOMMITTEE MEETING

WASHINGTON, D. C.

April 8, 1980

The B&W Water Reactors Subcommittee of the ACRS met on April 8, 1980 at 1717 H St., N.W., Washington, D.C. The main purpose of the meeting was to continue the Subcommittee's review of the sensitivity of B&W reactor systems to feedwater transients.

Notice of the meeting was published in the Federal Register on March 24, 1980. Copies of the notice, meeting attendees list, and meeting schedule are included as Attachments 1, 2, and 3, respectively. No written statements nor request for time to make oral comments were received from members of the public.

EXECUTIVE SESSION

Mr. H. Etherington, Subcommittee Chairman, convened the meeting at 8:30 A.M. and introduced the ACRS members and consultants (Attachment 2) who were present. The meeting was conducted in accordance with the Federal Advisory Committee Act and the government in the Sunshine Act. Mr. Peter Tam was the Designated Federal Employee. Mr. Etherington indicated, in his opening statement, that Harold Denton of NRR has recommended that construction of B&W plants not be halted. (Memo, Denton to Commission, Jan. 22, 1980). The Subcommittee would review this recommendation and document its findings in an ACRS letter.

Members and consultants made no comment at this point and the meeting proceeded as scheduled. The first presentation was made by an ACRS task force formed to study the B&W NSSS sensitivity question. Mr. Etherington noted that a presentation in a Subcommittee meeting by the ACRS staff is "unusual."

8006130055

CERTIFIED
5/2/80

PRESENTATION BY THE ACRS TASK FORCE

The ACRS task force has documented its findings in an 'official use only' draft report, distributed to the Subcommittee a few days before the meeting. Revised pages of the report were provided during the meeting and the task force indicated that the report will be finalized after ACRS comments are received.

The task force report, entitled "Review and Evaluation of the Babcock and Wilcox Nuclear Steam Supply System", was summarized as follows:

Mr. Ed Abbott described the B&W steam generator design. There are three control modes: the boiler following mode (primary system response changes according to steam demand), the turbine following mode (steam flow corresponds to primary system heat rate) and the optimized mode (integrated master control which uses a composite system of controls for the turbine, steam generator and reactor subsystems). B&W OTSG uses the last mode.

Mr. G. Young addressed the B&W OTSG dryout time question. The water inventory in a B&W steam generator is much less than that in a W or CE steam generator. Therefore, a loss of feedwater results in relatively fast dryout of the B&W OTSG. Mr. Young stated that dryout, however, is a problem only if the result is a safety issue. The Crystal River-3 and Rancho Seco incidences have shown that core damage did not occur as a result of dryout. The only way steam generator dryout can occur at a B&W plant now under construction and designed to the post-TMI standards is for multiple failures to the safety-grade emergency feedwater system (EFW). The loss of the EFW system at TMI-2 can be attributed to non-safety grade system interaction.

Mr. J. Stampelos addressed the integrated control system (ICS). He said that B&W performed a failure mode and effects analysis (FEMA) of the ICS. The report of the analysis has been evaluated by ORNL. Mr. Stampelos stated that the ICS is an asset to plant operators, based on operating history, but ICS interface with the balance-of-plant needs improvement.

Mr. J. Bickel made a presentation on dynamic response characteristics of the B&W NSSS. He stated that:

. The ability to change load in a B&W NSSS with its feed forward control mode is at least twice as fast as NSSS designs using UTSGs (CE and Westinghouse). This capability is the result of the ability to effectively eliminate the effects of the moderator temperature reactivity defect via proper regulation of feedwater flow and enthalpy.

. B&W NSSSs have a characteristically more sluggish pressurizer pressure and level response to T_{avg} change due to their inherently small $\frac{V_{rcs}}{V_p}$ ratios and operating pressures, and temperatures.

. B&W NSSSs (with their OTSGs), however, are subject to more adverse T_{avg} changes for the most limiting feedwater and steam flow transients than are plants using UTSGs.

. The need for safety-grade anticipatory reactor trip on turbine trip and loss of feedwater is clearly evident in view of the larger reduction in heat removal capability which can be experienced by an OTSG.

. With proper anticipatory reactor trip operation, the B&W OTSG can be operated safely. B&W should develop a better trip that would permit the ICS to function properly. The current anticipatory FW trip is only good for the interim; for the future, a sustained-loss-of-heat-sink trip should be developed.

DISCUSSIONS WITH THE NRC STAFF

Mr. Novak indicated that Mr. Denton of NRR has sent a memorandum to the Commission to recommend that construction of B&W plants should not be halted. The Staff has requested that the ACRS review this recommendation and document its concurrence (or lack of) in a letter to the Commission. In addition, since draft NUREG-0667 (to be described below) was published prior to the meeting, the Staff would also like the ACRS to comment on the report. Mr. Etherington indicated that since members and consultants have just received the report in the mail (date of report, April 2, 1980), the Subcommittee has no way of doing a fair review in this meeting. Such review would take place in a future meeting.

1. NUREG-0667 - Mr. R. Tedesco

Mr. Tedesco is chairman of a Staff task force formed to provide an assessment of the apparent sensitivity of the B&W designed plants to transients and the consequences of malfunctions and failures of the ICS and non-nuclear instrumentation (NNI).

The task force, consisting of about a dozen individuals, spent about two weeks studying the issues and documented its findings in NUREG-0667, "Transient Response of Babcock and Wilcox-Designed Reactors." General findings are:

. B&W designed plants are more responsive to secondary side perturbations than other light water reactors.

- The once-through steam generator design is basically sound; however, it requires a highly interactive and responsive control system (i.e., the integrated control system).
- A high degree of overall plant interaction is inherent in the integrated control system and the once-through steam generator.
- Based on the design features and the faster response of B&W plants during transients and upset conditions, the operators may be required to take more rapid action and have a better understanding of instrument response than operators on plants having other designs.

Implementation of the 22 specific requirements would be incorporated into TMI-2 action plan. Details of the implementation, to be presented in Section 7 of the report, were not finalized yet. The 22 requirements fall into four action areas, and are described in detail in the report:

AUXILIARY FEEDWATER SYSTEM

1. Classify AFW system as engineered safety feature. Where upgrade may not be feasible, consideration would be given to the addition of a dedicated AFW system (i.e., a separate train).
2. Automatically initiated and controlled by engineered safety features which are independent of the ICS, NNI, and other non-safety systems.
3. Installation of a diverse-drive AFW pump for Davis-Bessee.
4. Reevaluate and modify such that system is capable of differentiating between steam line break and overcooling and undercooling transients.

INSTRUMENTATION AND CONTROL

5. Improve reliability of instrumentation and plant control, separate and channelize power buses and signal paths for NNI and associated control systems.

Prompt followup actions should be taken on:

- . BAW-1564 (ICS reliability analysis)
- . NSAC-3/INPO-1 recommendations (evaluation of CR-3 incident)
- . IE Bulletin 79-27 (loss of NNI power supplies)

6. Establish prompt implementation of select data set of principal plant parameters for operator (safety grade).
7. Increased usage of incore thermocouples.
8. Provide a safety-grade containment high radiation signal to initiate containment vent and purge isolation.

DESIGN AND OPERATIONAL MATTERS

9. Plant operating and control functions should be modified to maintain pressurizer level on scale and pressure above HPI actuation setpoint.
10. Perform sensitivity studies of possible modifications which reduce the response of the OTSG to feedwater flow perturbations. (Consider active and passive measures).
11. Modifications should be made, to the extent feasible, to reduce or eliminate manual immediate actions for emergency procedures.
12. Provide a qualified I&C technician on duty with each shift.
13. Operator training on CR-3 event as well as plant-specific loss-of-NNI/ICS analysis and procedures.
14. B&W develop generic guidelines for loss-of-NNI/ICS.
15. Mandatory one-week simulator training for operators as part of re-qualification program.
16. Accelerated staff evaluation of B&W RCP restart criteria during small break LOCA.
17. Staff review alternative solution to PORV unreliability/safety system challenge rate concerns.
18. Expeditious completion of Crystal River 3 IREP study.

19. Staff develop plant performance criteria for anticipated transients for all light water reactors.
20. Continue studies of need to trip RCP during small break loca. (Conducted jointly by industry and NRC).
21. Reevaluate location of AFW injection into OTSG.
22. Staff analysis to determine significance and cause of LER's due to licensed personnel error being higher for B&W plants than other PWR's.

During the discussion, Mr. J. Taylor of B&W stated that a lot of people have the perception that the B&W pressurizer level goes off scale all the time. He said that it does not; of 350 reactor trips B&W examined, there were indications of only eighteen occurrences of off-scale behavior.

Mr. C. Domeck of Toledo Edison commented on NUREG-0667. He commended the Staff effort in producing the report but pointed out that some of the requirements may overlap with others. He suggested that there should be active owner participation in the writing of Section 7, "Implementation of Recommendations."

Mr. J. Taylor of B&W commended the Staff for its NUREG-0667 and the ACRS task force for its report. He indicated he would like a copy of the ACRS report. B&W has made a number of specific recommendations to its utility customers to improve plant performance in light of the Crystal River-3 incident. The utilities are currently evaluating these recommendations for plant specific applicability. B&W plants under construction have many of the features that NUREG-0667 recommends. He said that there should be some criteria for the acceptability of certain transient response, and B&W is suggesting, as a first-cut, the following statements. He called these "transient success statements."

1. RCS pressure should remain above HPI setpoint.

2. RCS pressure should remain below safety valve setpoint.
3. RCS temperature decreases at rates within technical specification limits.
4. Primary coolant be contained within the primary system and pressurizer quench tank.
5. Pressurizer level remains on scale.
6. OTSG level remains on scale.

B&W would also like to participate in the writing of Section 7 of NUREG-0667 Regarding the auxiliary feedwater system, B&W recommends a reliability-oriented upgrading as opposed to just safety-oriented upgrade.

2. RELAP-4 SIMULATION OF A HYPOTHETICAL OVERCOOLING TRANSIENT - W. Jensen

Mr. Jensen performed a number of calculations using the RELAP-4 code. The analysis is designed to be a best estimate and is plotted for comparisons with similar analyses by B&W. B&W used a detailed steam generator model (MAXI-TRAP) and a less detailed model (MINI-TRAP). The purpose of the RELAP calculation is to provide an audit of B&W's TRAP computer code which has not been approved by the NRC.

The assumed sequence of events was a reactor trip and turbine trip with the failure of main feedwater to throttle back to maintain the shutdown level of 32 inches. Feedwater was assumed to continue to flow causing shrinkage of the primary system. Letdown was assumed to isolate and the makeup flow of one charging pump was assumed. The increase in steam pressure following the turbine trip caused the turbine bypass and the steam relief valves to open for a few seconds. The RELAP results show more cooling than B&W, but with less loss of reactor system pressure and pressurizer level than B&W's Maxi-TRAP model. No voids were formed in the primary system over the 100 second interval of the RELAP analysis.

Mr. Ebersole said that there is a lot of "unreality about the mechanical evolution" in Mr. Jensen's study. Mr. Catton and Mr. Zudans said that plants should be better instrumented to provide more complete transient response data; mere computer study is not a satisfactory substitute for good data.

3. ANL SENSITIVITY STUDY ON B&W OTSG - B. Siegel

The objectives of this study is to determine the sensitivity of the cooling dynamics to perturbations in the secondary system, to determine effects of proposed applicant's modifications to reduce sensitivity of the coupling of primary to secondary systems, and to determine the effects of the use of secondary systems to generate steam (at Midland only). Task 1, to be completed by August, 1980, is a parametric study of the effectiveness of proposed modifications on transients (including the effects of location of AFW injection, MFW and AFW runback flow rates, time of initiation of runback, and OTSG water level). Task 2, to be done by July 1981, is an assessment of the change in sensitivity of the primary-to-secondary coupling due to the use of a tertiary heat exchanger (at Midland only).

All work is to start soon.

4. INTEGRATED RELIABILITY EVALUATION PROGRAM (IREP) - J. Murphy

The Staff has briefed the Subcommittee on this program in its Jan. 8, 1980 meeting. Mr. Murphy described new developments since then. The Integrated Reliability Evaluation Program is continuing. General results identified are:

- System interactions are significant (especially auxiliary cooling systems, DC and AC power sources).

. Likelihood of core damage and of high release categories may exceed those predicted in WASH-1400. However, analytical methods and data may differ and the analyses are not easily compared.

Eventually, IREP will be extended to study six plants: Indian Point, Zion, Oconee, Calvert Cliffs, Browns Ferry and Dresden. Also, plant owners may be asked to participate in the study.

After the Crystal River-3 incident, the Staff has decided that it should "de-emphasize quantitative risk assessment, but should emphasize diverse applicability of accident sequence analysis."

Mr. Taylor pointed out that the IREP should produce a "scrutable report with sufficient documentation to allow good peer review."

DISCUSSION WITH UTILITIES

1. TVA (D. Terrill)

   TVA has responded to Mr. Denton's 50-54 letter last year (as reported to the Subcommittee in its Jan. 8, 1980 meeting). TVA concurs with Mr. Denton's Jan. 22 letter which recommends to the Commission against halting of construction of B&W plants.

2. WPPSS (A. Hosler)

   Mr. Hosler presented a handout which provides a list of items WPPSS has committed to work on as a result of Mr. Denton's 50-54 letter. WPPSS has briefed the Subcommittee in its last meeting on these items (see minutes, Jan. 8, 1980). The current list describes the status of implementation of these commitments.

3. CONSUMERS POWER COMPANY - M. Salerno

   Mr. Salerno said that Midland construction is about 60% complete. He stated that the Staff should have been supplied sufficient information to make a decision with regard to construction stoppage. Although the 'sensitivity' issue is not closed, he urged that it be pursued during the normal licensing process such that there will be no more delay in licensing reviews.

EXECUTIVE SESSION

Mr. Etherington commended the ACRS task force for its efforts in writing the report.

The Subcommittee decided that all items on the schedule should be presented to the full ACRS, which would meet during the same week.

The Subcommittee agreed to recommend that the ACRS write a letter to the Commission on the Committee's concurrence with Mr. Denton's Jan. 22, 1980 memorandum, "Determination Whether B&W - Designed Plants Presently Under Construction Be Allowed to Continue."

No future meeting date was picked but subsequent to adjournment, Mr. Etherington picked April 29, 1980 for the Subcommittee to meet again to review NUREG-0667.
(Whereupon, the meeting was adjourned at 3:30 P.M.)
A complete transcript of the meeting is on file at the NRC Public Document Room at 1717 H St., N.W., Washington, D.C. or International Verbatim Reporters, Inc., Suite 107, 449 S. Capitol St. S. W., Washington, D.C. 20002, 202/484-3550

List of documents received before and during the meeting. Those that were in the public domain have been distributed in the meeting. (Members and consultants have received a copy of everything. For this reason, copies of these documents are not attached to these minutes; one copy of each, however, has been filed in the ACRS office).

1. R. Tedesco, "B&W Transient Response Task Force Presentation Before ACRS Subcommittee on B&W Water Reactors."

2. W. Jensen, "RECAP-4 Investigation of B&W Transients."

3. B. Siegel, "ANL Sensitivity Study on B&W OTSG."

4. J. Murphy, "Integrated Reliability Evaluation Program (IREP)."

5. A. Hosler, "Recommendations for WNP-1/4."

6. Tam to Subcommittee, "Status Report for the April 8, 1980 ACRS Subcommittee Meeting on B&W Water Reactors."

7. Draft "Transient Response of Babock & Wilcox - Designed Reactors", NUREG-0667, April 2, 1980.

8. Draft, "Review and Evaluating of the Babcock and Wilcox Nuclear Steam Supply System" (by the ACRS Staff task force).

of 5 U.S.C. 552b(c), Government in the Sunshine Act.

Authority To Close Meeting: This determination was made by the Committee Management Officer pursuant to provisions of Section 10(d) of Pub. L. 92-463. The Committee Management Officer was delegated the authority to make such determinations by the Director, NSF, on July 6, 1979.

M. Rebecca Winkler,
Committee Management Coordinator.
March 19, 1980.
[FR Doc. 80-8818 Filed 3-21-80; 8:45 am]
BILLING CODE 7555-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Babcock and Wilcox Water Reactors; Meeting

The ACRS Subcommittee on Babcock and Wilcox Water Reactors will hold a meeting on April 8, 1980 in Room 1046, 1717 H St., NW., Washington, DC 20555 to 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.

Notice of this meeting was published March 19, 1980.

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

The agenda for subject meeting shall be as follows:

Tuesday, April 8, 1980

8:30 a.m. Until the Conclusion of Business

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

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

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

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Peter Tam (telephone 202/634-1413) between 8:15 a.m. and 5:00 p.m., EST.

Dated: March 18, 1980.

John C. Hoyle,
Advisory Committee Management Officer.
[FR Doc. 80-8877 Filed 3-21-80; 8:45 am]
BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Ad Hoc Subcommittee on Three Mile Island, Unit 2 Accident Action Plan; Meeting; Change

The April 1-2, 1980 meeting of the ACRS Ad Hoc Subcommittee on Three Mile Island, Unit 2 Accident Action Plan will be held in Room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, MD, starting at 8:30 a.m. each day. Notice of this meeting was published on March 17 (45 FR 17097) and all other items pertaining to the meeting remain the same as published.

Dated: March 19, 1980.

John C. Hoyle,
Advisory Committee Management Officer.
[FR Doc. 80-8878 Filed 3-21-80; 8:45 am]
BILLING CODE 7590-01-M

Announcement by Automatic Telephone Answering Service

AGENCY: Nuclear Regulatory Commission.

During the next 60 days, the NRC will test the effectiveness of an automatic Telephone Answering Service as an additional method of providing current information to the public concerning the scheduling of Commission meetings. The telephone number to call is (202) 634-1498.

The schedule of Commission meetings will be recorded daily on or before 3 PM of the date preceding the meeting and updated as required. The meeting schedule will also continue to be distributed through the current mailing service, the Federal Register and the Public Document Room, as before; no change is anticipated in this distribution.

The recording will operate 24 hours a day. Because the Commission schedule is subject to late changes, those who are planning to attend a meeting should reverify the status of the meeting whenever possible.

Meetings will be at 1717 H Street unless otherwise indicated.

Further details of meetings are available from NRC staff during regular work hours at (202) 634-1410. At the end of the 60 day trial period, consideration will be given to extending this service to an "800" (Toll-Free) number.

Dated: March 18, 1980.

Walter Magee,
Office of the Secretary.
[FR Doc. 80-8879 Filed 3-21-80; 8:45 am]
BILLING CODE 7590-01-M

[Docket Nos. 50-325, 50-324]

Carolina Power & Light Co.; Issuance of Amendments to Facility Operating Licenses

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 28 and 50 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to (1) correct the table of safety related hydraulic snubbers, (2) provide for systematic implementation of instrumentation modifications, and (3) eliminate the requirement for removing the SRM "shorting links" during core alterations with control rods withdrawn.

The applications for amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendments will not result in any significant environmental

POOR ORIGINAL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
April 10, 1980

SUMMARY OF THE APRIL 8, 1980 MEETING OF THE B&W REACTOR SUBCOMMITTEE

Purpose of Meeting:

To continue the Subcommittee's review of the sensitivity of B&W reactor systems to feedwater transients.

Attendees:

ACRS

H. Etherington, Chairman
W. M. Mathis
J. Ebersole
J. Ray
S. Lawroski
I. Catton, Consultant
T. Theofanous, Consultant
Z. Zudans, Consultant
T. McCreless
G. Young, Fellow
J. Stamepelos, Fellow
E. Abbott, Fellow
W. Kastenberg, Fellow
P. Tam, DFE
J. Bickel, Fellow

TVA

L. A. Haack
Dennis L. Terrill
D. W. Wilson

OTHERS

N. S. Porter, WPPSS
A.G. Hosler, WPPSS
M. J. Salerno, CP Co.
R. M. Hamm, CP Co.
David Long, WSPR
Bob Leyse, NSAC
F. R. Miller, TE Co.
T. D. Murray, Toledo Edison
Chuck Domeck, Toledo Edison
Mary Gust, Shaw Pittman
Ken Wilson, Duke Power
N. M. Cole, MPR
S. Corsanico, IVRI

NRC

D. R. Quick, Reg. II
J. Murphy, PAS
A. Bournin, NRR
Bruce Wilson, NRR
F. Rowsome, PAS
T. M. Novak, DSS
B. L. Siegel, DSS
Z. P. Speis, DPM
Darl Hood, DPM
Hal Ornstein, AEOD
Eward Blackwood, IE
V. Panciera, NRR
S. Israel, NRR
G. Zech, NRR
Nina DiPaolo, IVRI

B&W

C. W. Connell
D. H. Roy
D. A. Womack
J. H. Taylor
R. R. Steinke
Bruce Karrasch
Roland L. Reed
John S. Shively
B.J. Short
R. O. Vosburgh
Al Ahamkhani

OTHERS

H. Filacchions, SAI
A. A. Garcia, SAI

ADVISORY COMMITTEE ON REACTORS SAFEGUARDS

B&W WATER REACTORS SUBCOMMITTEE

Enclosure 5

TENTATIVE MEETING SCHEDULE
APRIL 8, 1980

	APPROXIMATE TIME
EXECUTIVE SESSION	8:30 a - 8:45 a
° Chairman's Opening Statement (Mr. H. Etherington) ° Review the Meeting Schedule ° Comments by the Subcommittee	
DISCUSSION WITH NRC STAFF	
A. Introduction (Mr. T. Novak)	8:45 a - 9:00 a
B. Staff Analysis of Sensitivity of B&W Plants to Feedwater Transients	
1. New developments since the January Subcommittee Meeting. (W. Jensen)	9:00 a - 9:40 a
2. Progress Report: ANL Plant Sensitivity Program (B. Siegel)	9:40 a - 10:30 a
********************** BREAK **********************	10:30 a - 10:40 a
3. Pertinent Results from the Integrated Reliability Evaluation Program	10:40 a - 11:30 a
4. Report by B&W Reactor Response Vulnerability Task Force (2-week study of the Crystal River-3 event)	11:30 a - 12:30 p
********************** LUNCH **********************	12:30 p - 1:30 p
DISCUSSION WITH UTILITIES BUILDING B&W PLANTS: ANALYSIS OF B&W REACTOR SENSITIVITY TO FEEDWATER TRANSIENTS	
1. WPPSS	1:30 p - 2:15 p
2. Consumers Power Company	2:15 p - 3:00 p
3. TVA (Mr. D. Terrill)	3:00 p - 3:45 p
********************** BREAK **********************	3:45 p - 4:00 p
DISCUSSION WITH ACRS STAFF (T. McCreless, ACRS Fellows) Results of Analysis Performed by the ACRS Staff	4:00 p - 5:00 p
EXECUTIVE SESSION Discuss recommendations to be made in an ACRS letter to the Commission.	5:00 p - 5:30 p