ACRS-2402

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SUMMARY/MINUTES OF THE ACRS
METAL COMPONENTS SUBCOMMITTEE MEETING
FEBRUARY 27-28, 1986
WASHINGTON, D.C.

The ACRS Subcommittee on Metal Components met in Washington, D.C. on February 27-28, 1986 to review the 1) proposed broad scope rule change to GDC-4 (the application of leak-before-break concept to all high energy piping systems of nuclear power plants), 2) technical report on material selection and processing guidelines for BWR coolant pressure boundary piping (draft NUREG-0313, Rev. 2) and 3) format and content of plant-specific pressurized thermal shock safety analysis reports for pressurized water reactors (draft Regulatory Guide, Task SI02-4).

Notice of the meeting was published in the Federal Register on February 7, 1986 (Attachment A). The schedule of items covered in the meeting is in Attachment B. A list of handouts kept with the office copy of the minutes is included in Attachment C. The meeting was entirely open to the public. There were no written or oral statements received or presented by members of the public at the meeting. E. Igne was the cognizant ACRS Staff member for the meeting.

Principal Attendees

ACRS

P. Shewmon, Chairman
D. Ward, Member
H. Etherington, Member
W. Kerr, Member
M. Bender, Consultant
I. Catton, Consultant
E. Rodabaugh, Consultant
T. Kassner, Consultant

NRC Presenters

J. O'Brien
R. Bosnak
W. Hazelton
R. Woods
C. Johnson
J. Reyes
M. Vagins
N. Randall

Other Presenters

R. Cloud, R. L. Cloud Assos., Inc.
J. McInerny, Westinghouse
T. Chang, Westinghouse
S. Bernsen, AIF
D. Norris, EPRI

DESIGNATED ORIGINAL

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Proposed GDC-4 Modification (Broad Scope Rule)

J. O'Brien, RES, discussed this matter. This proposed rule allows application of leak-before-break (LBB) technology to all high energy (275 psi or 200°F) fluid system piping to demonstrate that specific pipe ruptures need not be treated in the design basis. The rule depends on advanced fracture mechanics techniques which have been experimentally validated and include evaluations of water hammer, corrosion, leak detection and indirect sources of pipe rupture. All reactor piping in all reactor types which satisfy rigorous acceptance criteria can take advantage of the rule. Only dynamic effects associated with pipe ruptures are excluded from the design basis. Containment design, ECCS performance and environmental qualifications are not impacted. The removal of pipe whip restraints, jet impingement barriers and other related facility design changes in operating plants, plants under construction and future designs are permitted. Averted worker radiation exposures are measured in several (possibly many) 10,000's of man-rem. Cost savings are measured in several (possibly many) $100 millions. Public safety is believed to be enhanced.

It was stated by the NRC Staff that the detailed acceptance criteria for the selection of piping systems and their basis to which the broad scope rule will be applied is still being developed. The subcommittee, during the meeting, presented the following comments on application of the broad scope rule:

- The rule should not be applied in the foreseeable future to piping systems operating at temperatures above 650°F because of possible degradation due to creep mechanism.

- In evaluating the piping systems the NRC Staff should consider the consequences of a crack opening along a weld. This is a credible failure mode and should be considered.

- The material properties and fabrication processes of the weld in piping systems where LBB criteria are applicable should be given at least as much consideration as properties of the base metal of the piping system.

- The LBB argument may also be usefully applied to matters related to ECCS, containment design and equipment qualification with similarly accompanying benefits.

The following are foreign practices with respect to the use of the broad scope rule.

- UK: NII shows a strong inclination to reject LBB for Sizewell based on concerns with stress corrosion cracking and inadequate NDE of cast stainless steel piping and components. (CEGB at odds with NII on this.)

- France: Undecided, but weakly inclined to reject LBB at this time partly because of commitment to standardization, although research on LBB in progress.

- FRG: Strong commitment to LBB in new PWR main coolant, main feed and main steam line inside of containment.

- Italy: Close to FRG practices.

- Japan: Inclined to accept LBB, even in BWRs. Heavy investment in LBB research.

- Canada: Inclined to accept LBB for certain piping systems at the Darlington facility.

R. Bosnak, NRR, discussed plans for the implementation of the limited and broad scope proposed rule to GDC-4. With respect to the limited scope rule, schedular exemptions have been granted to all recently licensed plants covering protection against dynamic effect, i.e., pipe whip restraints, jet impingement barriers, loads in unbroken portion of loop and branch lines, and pressurization transients. No change is permitted in containment design, ECCS and equipment environmental qualification. Recently, Crystal River-3 and Surry have requested that downsizing/or removal of large snubbers required to resist pipebreak loads is possible with an overall result of improved system performance and reliability. The Staff indicated that an independent design and fabrication assurance is a prerequisite.

The use of LBB applied to all high energy lines (broad scope rule) has been requested by lead plant Beaver Valley-2 in their Whipjet program. Others are expected to submit their programs to use the broad scope rule. R. Bosnak stated that the acceptance criteria (Regulatory Guides, SRPs) will be developed following public comments, but should incorporate the following major items:

° Pipe rupture probability should be extremely low $\sim 10^{-6}$. Alternatively, a deterministic evaluation with verified design and fabrication, and adequate ISI is necessary.

° Leakage detection systems should be reliable, redundant, diverse and sensitive. (This requirement is more difficult to apply to piping systems outside of containment.)

° A margin of at least 10 on detection of leakage from through-wall flaws should exist.

In conclusion, R. Bosnak stated that 1) removal of dynamic pipe rupture protective devices and deletion of large dynamic pipe rupture loads is beneficial, 2) decoupling of SSE and LOCA is acceptable and 3) use of code allowable stresses is sufficient.

ACRS input and comment are sought by the NRC Staff in the following areas:

° Guidance in the assurance of extremely low probability of pipe rupture, i.e., levels and failure modes.

° If probability is above a given threshold level, what, if any, methods of protection should be used.

° Need for inservice inspection augmentation.

° Design load combinations and allowable limits for future plants.

° Equipment qualification and the use of designated environmental profiles instead of bounding conditions.

° Leak detection and reliable prediction of leakage through stable cracks.

T. Chang, Westinghouse, discussed their efforts in the application of the proposed broad scope rule. The technology used has been accepted for eliminating PWR reactor coolant loop breaks. Westinghouse has been the industry leader in the development and application of LBB technology. It was stated that an application to apply the broad scope rule was submitted to the NRC Staff in 1984. The NRC Staff has not reviewed the application because the broad scope rule has not yet been promulgated. The Westinghouse LBB methodology is similar to the proposed NRC rule under consideration.

Westinghouse has stated that the proposed broad scope rule should be expedited in order that plants under construction can fully realize the benefits.

R. Cloud and members of Duquesne Light Co. presented an updated status report on the application of the LBB concept to high energy piping systems located inside and outside of containment. R. Cloud and Associates, Inc. is the major contractor for this program called "Whipjet." Briefly, Whipjet will do the following:

- Satisfy DEGB postulation with engineering analysis showing a detectable leak-before-break is assured.

- Reduce hardware in the plant in order to minimize plant cost, facilitate access for ISI and reduce time in performing inspections and maintenance to enhance ALARA position.

- Increase plant safety through more complete knowledge of material properties and capabilities.

Whipjet is not intended to change the use of DEGB for establishing design criteria for ECCS, containment and equipment qualification.

A brief discussion to address the failure mode at Mohave and Monroe fossil plants was presented. It was stated that the catastrophic failure at the longitudinal weld of the reheat steam line was caused by creep rupture. The subcommittee agreed that if the upper temperature limit of the material is less than 650°F (as in LWRs) the creep rupture failure mechanism does not seem to occur.

D. Norris, EPRI, briefly discussed their involvement in the area of the application of LBB to all high energy piping systems. He stated that enhanced plant safety and economic benefits to plant owners would occur if the proposed broad scope rule was promulgated. EPRI is involved with an ongoing program in this area with all NSSS vendors. Between 1975 and 1984 EPRI has spent $31 million in structural mechanics and NDE studies. Between 1985 and 1989 they plan to spend about $18 million.

He stated that EPRI is supportive of NRC/ACRS initiatives in this area. Some significant issues that need to be addressed are: leak detection methodology and its reliability outside containment, availability of weld toughness data, applicability to BWRs with 316NG with IHSI of the welds, and definition of acceptance criteria.

S. Bernsen, AIF, spoke briefly about AIF work in this area. He stated that the following criteria were proposed to the NRC and ACRS three years ago:

- ° No intermediate breaks need to be postulated.
- ° No breaks are assumed at terminal ends where leak-before-break criteria are satisfied, unless location is susceptible to unstable cracks from corrosion, thermal fatigue or water hammer.
- ° Eliminate the SSE + LOCA loading combination for piping and support structures.

NUREG-0313, Rev. 2 (Implementation of BWR Pipe Crack Recommendation)

W. Hazelton, NRR, discussed the long-range approach for dealing with stress corrosion cracking in BWR piping as described in draft NUREG-0313, Rev. 2. Revision 2 expands Revision 1 coverage to include all stainless steel piping systems (class 1, 2, and 3), requires formal qualification of NDE examiners and procedures, provides guidelines for evaluation and repair of cracked welds, and upgrades leakage limits and monitoring. Revision 2 of the report generally follows the recommendations of the NRC Piping Review Committee as found in NUREG-1061, Vol.1.

Draft Regulatory Guide on PTS

R. Woods, NRR, discussed the draft Regulatory Guide implementing the PTS rule. Briefly, the PTS rule defines the screening limit (270°F, 300°F), describes the calculation of $RT_{PTS}$, discusses the flux reduction programs and determines when the PTS/PRA analysis needs to be performed.

The draft Reg. Guide suggests the methodology of the PTS/PRA analysis. The PTS/PRA analysis methodology is based on work done at H. B. Robinson, Calvert Cliffs and Oconee plants.

ACRS Questions on PTS

The Subcommittee next discussed the ACRS concerns with respect to PTS. R. Woods, NRR, led the discussion of the NRC Staff's responses to the following ACRS questions.

Q.1: Is there any reason to believe that the issue of PTS cannot be treated generically, e.g., that some classes of plants or particular designs are subject to a significantly higher frequency of severe vessel overcooling transients?

A.1: The NRC Staff based on B&W Report 1791, published in 1983, recommended to the Commission that the same screening limits should be applicable to B&W plants because, although the challenge frequency might be higher, the severity or risk rate, appears to be less. Hence, the NRC Staff was unable to justify a different screening limit for the B&W plant. [SECY-83-288, dated July 15, 1983 contains the NRC Staff's SER on this matter.]

With respect to pressure vessel material properties (copper/nickel content) this is properly accounted for automatically in the rule.

Q.2: How well justified is the crack distribution used in developing the NRC position on PTS? Is there a sufficient bases to justify the distribution used? Also, has allowance been made for crack growth? Should there be?

A.2: The NRC Staff feels that the crack distribution assumed for the PTS study (modified Octavia and Marshall codes) is conservative, although the NRC Staff admits that it is one of the largest uncertainties that exist in the PTS study. It was also mentioned that current pressure vessel inspection indicates a more conservative flaw distribution was assumed in the analysis. Crack growth was accounted for in the study by assuming flaws of various depths representing flaw size at the beginning and end of life of the pressure vessel.

Q.3: Throughout the transition temperature range, the Staff appears to permit use of unirradiated data with the upper shelf as a ceiling for fracture toughness. Is this so? If so, is this best estimate? May it not be unconservative? If so, by how much?

A.3: Current studies are being performed to correlate the Charpy curve to the $K_{Ic}$ curve, specifically if the Charpy curve drops and changes shape does the $K_{Ic}$ curve change shape? Information obtained thus far indicates that some margins have been eroded, but that the NRC Staff feels comfortable because the margins are still large.

Q.4: Dr. I. Catton, an ACRS consultant, questions whether HPI recovery following partial core uncovery is covered adequately under PTS (or, if not, elsewhere). Dr. Catton suggests that HPI following partial core uncovery will lead to low temperatures and possible water hammer. Can the Staff provide estimates on the frequency and severity of such an event? What are the major sources of uncertainty in these estimates, and their magnitudes? Is human error very important? Is plant design important? Is thermal hydraulic prediction adequate for the purpose?

A.4: R. Woods stated that the accident scenario of partial uncovering of the core followed by core recovery is a severe accident concern and is not regarded as a PTS issue.

Q.5: Are there any steam generator overfill scenarios which the Staff considers significant for PTS?

A.5: Yes, there are steam generator overfill scenarios that are important to PTS. Oak Ridge identified the sequence of a break in the steam line followed by overfeed by the auxilliary feedwater as an important, but not dominant, sequence for the Robinson analysis, although it is dominant for the Oconee analysis. The NRC Staff indicates that the Oak Ridge analysis supports the PTS rule.

With respect to loop flow stagnation concerns, J. Reyes stated that this problem is being addressed by T. Theofanous.

Q.6: Are there any reactors for which the data on chemical composition of critical welds is not well determined? If so, how is a judgment made? Is the difference between the composition accepted and the worst possible significant? If so, how much less likely must the worst possible be? How is this judgment made?

A.6: N. Randall, RES, stated that updated chemical composition of critical welds were recently documented--on January 23, 1986. In general, the NRC Staff stated that they are reassured in the sense that the justification for numbers used in the PTS analysis is getting much better.

Four categories of weld chemistry data are available. The first category is the actual measured value of the critical welds, which is true for nearly all pressure vessels. The second category is called generic chemistry where a plant had only one or none of the measured values for their critical welds, but through searches have found other typical vessels. These are then sampled to determine its chemistry. B&W and Westinghouse Owners Groups have determined weld chemistry by this method. It was stated that about half of the plants approaching the screening criteria are in this category.

The third category is that of historical numbers. In this case, all they have is a statement indicating that for vessel welds fabricated in this time period a certain distribution of copper/nickel content is reported. From this data a conservative upper bound value is obtained. A quick look at the data in this category shows that ample margins exist before the screening value is reached. In the fourth category, if no data is available the NRC Staff dictates an upper bound value which is very conservative.

Q.7: What is the expected consequence of a through-wall crack? What is the likelihood of: (a) core melt, (b) late containment failure, and (c) early containment failure?

A.7: In part, the answer relates to a severe accident scenario. But the NRC Staff did fund a study by Pacific Northwest Laboratory to develop a vessel failure model. This model was then applied to Oconee in order to determine the containment failure modes. The result is that 4/10-percent of the through-wall cracks would lead to early containment failure if we assume no containment spray and 1/10-percent would lead to early containment failure with containment sprays; these would lead to isolation containment failures. The study indicates that the "objective of individual risk of early and late fatalities are met." This study is reported in a paper by R. Barrett and E. Throm which was presented at the 12th Annual Water Reactor Safety Meeting at NBS in Germantown, Md.

## Future Action

The subcommittee decided that a subcommittee report on GDC-4 (Broad Scope Rule), NUREG-0313, Rev. 2 (Implementation of NRC Pipe Crack Review Committee on Long Term Fix of BWR Pipe Cracking) and Draft Reg. Guide on PTS (Implementation of PTS Rule) should be presented to the ACRS at its 311th Meeting in March 1986.

The meeting adjourned about 1:05 p.m.

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NOTE: A transcript of the meeting is available in the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., or can be purchased from ACE-Federal Reports, 444 North Capitol Street, Washington, D.C. 20001, (202) 347-3700.

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In addition, the licensee also was informed of the NRC concern about procedural controls in high-radiation areas via several information notices and a circular (Information Notice 84–19 dated March 21, 1984, Information Notice 82–51 dated December 26, 1982, and Circular Notice 76–03 dated September 13, 1976). These notices emphasized the importance of ensuring that radiation protection procedures and radiation protection training and retraining programs specifically address the matter of control and access to such areas and initiate appropriate retraining of all plant personnel. They also recommended that entry be allowed only after appropriate management review and approval. Further, they recommended periodic audit of these actions to ensure their continued effectiveness. Many of the actions noted in the Notices are similar to those in the Confirmatory Action Letter issued by the NRC to Vermont Yankee on September 9, 1985. In addition, there have been a number of escalated enforcement actions for similar violations at other plants of which the licensee should have been aware. A purpose of publishing escalated enforcement actions in NUREG–0940 and Orders Imposing Civil Penalties in the **Federal Register** is to give licensees notice of other enforcement actions which may bear on their own operations. (See Vol. 4, No. 1, p. I.A–94 and Vol. 3, No. 2, p. I.A–1 of NUREG–0940.)

Accordingly, the NRC maintains that the licensee had prior notice of potential problems associated with TIP rooms. Therefore, a basis would have existed for an increase in the civil penalty amount had it not been for the licensee's reporting of this event and prompt short-term corrective actions.

*Licensee's Assertion*

The licensee claims that at the time of the Enforcement Conference on September 5, 1985, significant efforts had been taken to assess the specific causes of the incident and develop long-term proposed corrective actions. In particular, on the day (August 9) following the event, the Plant Manager directed the Chemistry and HP technician to generate a Plant Information Report (PIR) so that the event could be analyzed and recommended long-long corrective action could be provided. The final PIR, which was issued approximately 6 weeks later on September 17, 1985, proposed six long-term corrective actions. On September 21, 1985 the Plant Manager dispositioned the long-term recommendations. The licensee contends that the development and finalization of this long-term corrective action program occurred in a prudent and timely manner.

*NRC Evaluation*

The NRC maintains that the long-term actions taken by the licensee were not particularly prompt in that some of the actions could and should have been in place at the time of the enforcement conference, namely, an upgrade of the procedures for entry into locked high-radiation areas in general, and the TIP room in particular. These items were not provided by the licensee at the Enforcement Conference and appeared to have been considered only after the Enforcement Conference on September 5, 1985 and the Region I Confirmatory Action Letter (CAL) issued on September 9, 1985. In addition, four of the six items in the licensee's PIR simply proposed evaluation of certain aspects of the program rather than describing specific actions taken or necessary to correct deficiencies and improve the program. It was not until September 21, 1985 after the Enforcement Conference and issuance of the Confirmatory Action Letter (CAL) that the licensee committed to take these actions.

For these reasons, the NRC maintains that the licensee's long-term actions were not unusually prompt and do not provide an adequate basis for mitigation of the civil penalty.

*NRC Conclusion*

After consideration of the answers received and the licensee's statements of fact, explanation, and arguments for mitigation of the proposed civil penalty, the staff concludes that any adjustment to the civil penalty amount is inappropriate. Therefore, the proposed $50,000 civil penalty should be imposed.

[FR Doc. 86–2740 Filed 2–6–86; 8:45 am]

BILLING CODE 7590-01-M

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

The ACRS Subcommittee on Babcock and Wilcox (B&W) Water Reactors will hold a meeting on February 25, 1986, Room 1046, 1717 H Street, NW, Washington, DC.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

*Tuesday, February 25, 1986—8:30 A.M. Until the Conclusion of Business*

The Subcommittee will consider the implications of operating experience on the adequacy of B&W plant designs, including consideration of the severe overcooling event at Rancho Seco on October 2, 1985. The Subcommittee may also review the NRC Staff's plans to reassess the long-term safety of B&W reactors.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. 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 ACRS staff member named below as far in advance as is practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, its consultants, and other interested persons regarding this review.

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 ACRS staff member, Mr. Richard Major (telephone 202/634–1413) between 8:15 A.M. and 5:00 P.M. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the schedule meeting to be advised of any changes in schedule, etc., which may have occurred.

Dated: February 3, 1986.

**Morton W. Libarkin,**

*Assistant Executive Director for Project Review.*

[FR Doc. 86–2761 Filed 2–6–86; 8:45 am]

BILLING CODE 7590-01-M

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## Advisory Committee on Reactor Safeguards Subcommittee on Metal Components; Meeting

The ACRS Subcommittee on Metal Components will hold a meeting on February 27 and 28, Room 1046, 1717 H Street, NW., Washington, DC.

To the extent practical the meeting will be open to public attendance.

ATTACHMENT A

However, portions of the meeting may be closed to discuss industry proprietary information.

The agenda for subject meeting shall be as follows:

*Thursday, February 27, 1986—8:30 A.M. Until the Conclusion of Business*

*Friday, February 28, 1986—8:30 A.M. Until the Conclusion of Business*

The Subcommittee will review, but not necessarily be limited to, the following items: (1) NUREG–0313, Revision 2, entitled, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," and (2) Regulatory Guide XXX entitled, "Guide for License Preparation and NRC Staff Review of Plant Specific Analysis Required by PTS Rule." The Subcommittee will also hear a status report of the proposed broad rule to modify GDC–4 of 10 CFR Part 50 (the leak-before-break broad scope rule is applicable to all LWR high energy piping systems).

Oral statement may be presented by members of the public with concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. 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 ACRS staff members as far in advance as practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, its consultants, and other interested persons regarding this review.

Further information regarding topics of 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 ACRS staff member, Mr. Elpidio Igne (telephone 202/634–1414) between 8:15 a.m. and 5:00 p.m. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Dated: February 3, 1986.

Morton W. Libarkin,

*Assistant Executive Director for Project Review.*

[FR Doc. 86–2762 Filed 2–6–86; 8:45 am]

BILLING CODE 7590-01-M

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[Docket No. 50–445]

**Texas Utilities Electric Co. et al. Comanche Peak Steam Electric Station, Unit No. 1; Environmental Assessment and Finding of No Significant Impact**

The Nuclear Regulatory Commission (the Commission) is considering issuance of an extension to the latest construction completion date specified in Construction Permit No. CPPR–126 issued to Texas Utilities Electric Company, Texas Municipal Power Agency, Brazos Electric Power Cooperative, Inc. and Tex-La Electric Cooperative of Texas, Inc. (Applicants) for the Comanche Peak Steam Electric Station Unit No. 1 (the facility) located on Applicants' site in Somervell County, Texas.

**Environmental Assessment**

*Identification of Proposed Action:* The proposed action would amend the construction permit by extending the latest construction completion date to August 1, 1988. The proposed action is in response to Applicants' request dated January 29, 1986, as supplemented February 4, 1986.

*The Need for the Proposed Action:* The proposed action is needed because the construction of the facility is not yet fully completed. The Applicants state that, although construction on Comanche Peak Unit 1 was essentially completed early in 1985, major efforts to reinspect and reanalyze various structures, systems, and components is currently underway. These efforts are being conducted by the Applicants' Comanche Peak Response Team to verify both design and construction adequacy as well as to respond to numerous issues raised in the operating license proceeding, by the NRC's Technical Review Team, and by other sources. This activity has been ongoing since the fall of 1984. The Applicants anticipate that it will not be complete before the second quarter of 1986. In addition, the operating license hearings are not yet completed and will involve additional time for which the construction permit will be needed.

*Environmental Impacts of the Proposed Action:* The environmental impacts associated with construction of the facility have been previously discussed and evaluated in the NRC staff's Final Environmental Statement (FES) issued in June 1974 for the construction permit stage which covered construction of two units. Unit 2 is not affected by the proposed action.

Since the proposed action involves extending the construction permit, radiological impacts are not affected by this action. There are no radiological impacts associated with this action. The impacts that are involved are all non-radiological and are associated with continued construction.

Since the construction of the facility is essentially 100% complete, most of the construction impacts discussed in the FES have already occurred: construction-related activities have disturbed about 400 acres of rangeland, the Squaw Creek Reservoir has been built, as have transmission lines and corridors, and a railroad spur. These activities and their impacts occurred earlier and are not affected by this proposed action.

The reinspection and rework that may be required will not have any significant environmental impact. The impacts associated with the work are equivalent to those of a maintenance or repair program. This activity will all take place within the facility and will not result in impacts to previously undisturbed areas.

There are no new significant impacts associated with this extension. There are, however, impacts that would continue in order to complete plant construction in addition to rework discussed above. These are community and traffic impacts, and continued groundwater withdrawal.

Community impacts from continued construction would be similar to those impacts previously assessed. The total number of workers on-site for both units at the present time (about 5300) is about the same (although somewhat smaller) as during earlier peak construction periods. The number of workers specifically assigned to Unit 1 is small compared with the number associated with the completion of Unit 2. The number of workers on-site will decline as the reinspection program for Unit 1 is completed during 1986. Continuing construction does not involve community impacts different from those previously considered or significantly greater than those previously considered or experienced.

The construction permits for Comanche Peak Units 1 and 2 limit groundwater usage to 40 gpm on an

A-2

REV. 1, 2/20/86

# TENTATIVE SCHEDULE
# ACRS METAL COMPONENTS SUBCOMMITTEE MEETING
# FEBRUARY 27-28, 1986
# WASHINGTON, D. C.

February 27, 1986

| | | | |
|---|---|---|---|
| I. | Chairman's Opening Statement- P. G. Shewmon | 15 min | 8:30-8:45 |
| II. | Proposed Rule Change to GDC-4 | | |
| | 1. Limited Scope Rule, status and update - J. O'Brien | 15 min | 8:45-9:00 |
| | 2. Broad Scope Rule | | |
| | ° Presentation - J. O'Brien | 120 min | 9:00-11:15 |
| | * * * B R E A K * * * [with a 15-min. break at 10:30] | | |
| | ° NRR Plans - R. Bosnak | 45 min | 11:15-12:00 |
| | * * * L U N C H * * * | | 12:00-1:00 |
| | ° Implementation | | |
| | - Status at Beaver Valley-2 - R. Cloud | 1 hr | 1:00-2:00 |
| | - Westinghouse Plans - H. Clark | 1 hr | 2:00-3:00 |
| | * * * B R E A K * * * | 15 min | 3:00-3:15 |
| | 3. Report of EPRI's Study in this area - D. Norris | 1 hr | 3:15-4:15 |
| | 4. AIF Comments - ~~Sid Bernstein~~ S. Bernsen | 30 min | 4:15-4:45 |
| III. | Subcommittee's Discussion | 15 min | 4:45-5:00 |
| IV. | RECESS | | |

February 28, 1986

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|---|---|---|---|
| I. | Chairman's Statement-P. G. Shewmon | 15 min | 8:30-8:45 |
| II. | Presentation on NRC Technical Positions on BWR Pipe Crack, NUREG-0313, Rev. 2 - B. D. Liaw | 120 min | 8:45-11:00 |
| | * * * B R E A K * * * [with a 15-min. break at about 10:00] | | |
| III. | Presentation on Format and Content of Plant-Specific PTS Safety Analysis Reports for PWRs, (Reg. Guide Task SI 502-4) - R. Woods | 60 min | 11:00-12:00 |
| | * * * L U N C H * * * | | 12:00-1:00 |
| IV. | Response to ACRS Concerns on PTS - R. Woods/NRC Staff | 90 min | 1:00-2:30 |
| V. | Subcommittee Discussion | 30 min | 2:30-3:00 |
| | ADJOURNMENT | | 3:00 |

ATTACHMENT B

ATTACHMENT C
LIST OF HANDOUTS
ACRS METAL COMPONENTS SUBCOMMITTEE MEETING
FEBRUARY 27-28, 1986, WASHINGTON, D.C.

February 27, 1986

1. Status of Final Limited Scope GDC-4 Rule, J. O'Brien, RES

2. Broad Scope GDC-4 Rule, J. O'Brien, RES

3. NRR Plans for the Implementation of the Limited and Broad Scope Rule Revisions to GDC 4, R. J. Bosnak, NRR

4. Alternative Pipe Break Criteria (Leak-Before-Break Concept), John McInerny and Dr. T. Chang

5. Robert L. Cloud Associates:
Review of Whipjet, Swec Progress, R. L. Cloud Progress, EPRI Progress

February 28, 1986

1. Long Range Approach for Dealing with Stress Corrosion Cracking in BWR Piping Draft NUREG-0313, Rev. 2, Gen. Issue 86, W. S. Hazelton, NRR

2. PTS Regulatory Guide - ACRS Questions, Roy (H. W.) Woods, NRR

3. Pipe Break and Load Combinations, S. Bernsen, AIF Subcommittee on Load Combinations