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MINUTES OF THE ACRS SUBCOMMITTEE
MEETING ON
BULLETINS & ORDERS AND ECCS
JANUARY 3&4, 1980
LOS ANGELES, CA

The ACRS Bulletins and Orders and ECCS Subcommittees held a joint meeting in Los Angeles on January 3 and 4, 1980 to continue the discussion of the NRC/industry response to NRC Bulletins and Orders issued following the TMI-2 accident. The notice of the meeting appeared in the Federal Register on December 19, 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.

Dr. Plesset, Subcommittee Co-Chairman, opened the meeting at 8:30 am 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.

OPENING COMMENTS/EXECUTIVE SESSION

Subcommittee members and consultants indicated a number of concerns and suggested a number of areas that they would like to see reviewed during the meeting. These included use of the PORV block valve, automatic initiation of auxiliary feedwater, the use of small scale test facilities (Semiscale, TLTA) to obtain information on small breaks, use of the single failure criteria, and the possibility of reactor coolant pump operation at reduced speed.

PRESENTATION BY THE NRC STAFF

Dr. Ross, Bulletins and Orders Task Force Director, briefly reviewed the work undertaken and indicated that the product would be contained in five reports, one of which is a task force summary report plus four generic reports covering each of the NSSS vendors. The reports are to be issued in late January 1980. He indicated that the task force wanted to get the ACRS comments and/or concurrence on the recommendations contained in the task force report. Dr. Ross indicated that the recommendations contained in the B&O Task Force reports would be coordinated with the work that was ongoing with the NRC Action Plans and that the overall priorities would be established by the Action Plan.

The Task Force efforts covered operating reactors and concentrated on improvements in the auxiliary feedwater systems, additional analyses of small break LOCAs, revised emergency operating procedures for the plant operators, and improved operator training. In response to a question from Mr. Etherington on PWRs with low head safety injection pumps, Mr. Ross indicated that the NRC would be investigating the flow rates through PORVs and other means of reducing system pressure so that HPIS flow could be obtained. One of the items that the Staff will be looking at in the near term involves a decision as to whether or not loss of all feedwater should be a design basis event. A number of calculations are planned over the next several months to decide whether feed and bleed would be an acceptable method for core cooling. The NRC Staff is also asking that a number of natural circulation tests be performed in operating reactors. The Staff is asking that two phase natural circulation be included in this work. Tests in LOFT may help resolve this issue. A number of different solutions exists for improving the capability of getting water into the reactor system. These include high head HPI pumps and increased relief capacity. Another possibility under consideration is high pressure RHR systems.

Mr. W. Kane reported on the response of the utility owners groups to the NRC L&E Bulletins and on the status on the B&W plant responses to NRC long term requirements. Following the TMI-2 accident the NRC issued a number of I&E Bulletins asking operating utilities to address problems seen at TMI-2. The Bulletins & Orders Task Force was formed to follow the progress on these items. The evaluation of the Bulletin responses was performed over a six-month period by an interdisciplinary review team including individuals familiar with reactor systems, operator licensing, and inspection, and enforcement. The final product of the Bulletin evaluation effort is a series of reports that will be issued as NUREGs. In response to a question, Mr. Kane indicated that in the original bulletins there was a requirement that the reactor coolant pumps be left on in situations involving small breaks. Later it was shown by a number of calculations that by leaving the pumps on there was difficulty in complying with Appendix K for a certain range of small breaks. Consequently, a new I&E Bulletin was issued requiring that the reactor coolant pumps be tripped following small breaks. In response to an additional question, Mr. Kane indicated that flow indication was being required for auxiliary feedwater systems rather than just indication of pressure increase at the pump discharge.

One area where there was some difficulty with the initial Bulletin and Orders requirements was in establishing 50 degrees in subcooling prior to termination of high pressure injection. After a number of discussions the NRC Staff and W agreed on a subcooling requirement that was less than 50 degrees and was based upon achieving at least 20 degrees of subcooling plus an allowance for instrument uncertainty. This was done in an attempt to prevent overfilling of the pressurizer for some transients.

Shortly after the TMI-2 accident there was a special study performed on the B&W plants. A number of short- and long-term improvements were initiated. The first was an upgrade of the auxiliary feedwater system, the second was the submittal of a failure mode and effects analysis of the integrated control system, the third was an upgrade of the anticipatory reactor trip, and the fourth was continued operator training and drilling to insure a high state of preparedness. The auxiliary feedwater upgrade included installation of additional emergency feedwater pumps at Oconee, connection of existing auxiliary feedwater pumps to vital busses, provision for control room annunciation of all auto start conditions of the emergency feedwater system, and requirements to provide auxiliary feedwater flow verification in the control room. The NRC Staff has also asked the licensees study and improve the reliability of their auxiliary feedwater systems. Dr. Zudans suggested that equal attention should perhaps be given to main feedwater systems in order to prevent the necessity of using the auxiliary feedwater systems.

With regard to the B&W integrated control system the Oak Ridge National Laboratory was contracted by the NRC to do a review and analysis of the system. A number of findings came out of this study. The first indicated that the ICS system itself shows a low failure rate, however, the interfacing equipment such as power supplies are prone to failures. In fact, the ICS apparently prevented more plant upsets than it caused. Recommendations included the upgrading of non nuclear instrumentation and integrated control system power supply reliability and design and review and upgrading of the reactor protection system. Mr. Michelson indicated that attention should be given to the integrated control system and a possible failure mode that would cause the steam generator to overfill. Consequences of overfill include sticking open of the secondary valves and severely overcooling the primary system.

Mr. Kane indicated that one of the short term items for B&W plants was the provision for control grade trip to trip the reactor upon loss of feedwater or turbine trip. In the long term this trip will have to be made safety grade. The new operator training and drilling program involves increased emphasis on thermodynamics, hydraulics, heat transfer, fluid flow, small break phenomena, inadequate core cooling, and transient training including loss of feedwater events. The passing grade on licensing and requalification exams will now be 80& overall with no section below 70%. As a part of the NRC long term requirements the capability of simulators will have to be upgraded in order to improve the representation of severe secondary side transients.

Mr. Phil Matthews, NRC Staff, summarized the status of the auxiliary feedwater reliability study made on operating plants. The Staff has reviewed the reliability of the W and CE plants and letters concerning Staff recommendations for upgrading of reliability of the systems were sent out in September and October. The Staff is now in the process of getting responses to their letters. For the B&W operating plants a number of immediate modifications were made during the summer of 1979. Subsequent to those modifications, a long term program was set up to embark on additional reliability improvements. Along with Staff recommendations was a requirement tc make auxiliary feedwater initiation automatic in those plants where it is now manual. Initially, the initiation of the auxiliary feedwater should be control grade, and in the long term they should be made safety grade. ACRS consultants indicated that some utilities had safety concerns with regard to making auxiliary feedwater systems automatic. Among the safety concerns raised by the utiltities was the impact on containment if a main steamline break occurs and the auxiliary feed system continues to feed the steam generator. The NRC Staff indicated that they were reviewing the issues involved but at least, initially, believed that automatic initiation of feedwater was more reliable for most conditions than was manual. Automatic initiation of auxiliary feedwater would require manual intervention in the case of the feedwater or main steam line break inside of containment, however, this is less likely than the requirement of manual initiation for various loss of feedwater transients. In one case in Yankee Rowe, the Staff has allowed, at least initially, manual initiation of auxiliary feedwater. This is because the steam generator inventory is such that the plant has approximately one hour to initiate auxiliary feedwater flow. Dr. Catton raised a question as to whether or not automatic feedwater initiation woold aggravate water hammer problems in the auxiliary feedwater systems. The NRC Staff indicated that they had requested the utilities to address this matter in their responses to the NRC. At least in some cases automatic initiation of auxiliary feedwater would tend to help the water-hammer situation rather than make it worse.

The NRC Staff indicated that Dr. Okrent had previously raised a question with regard to the basis for seismic qualification of auxiliary feedwater sources. The Staff indicated that their position on siesmic qualification of auxiliary feedwater systems was that the plant should either have a Seismic Category 1 water source or a water source with seismic design equivalent to that which was considered seismic at the time that the plant was licensed. Of the plants that the Staff studied, they found for the most part they do meet this requirement. The exceptions fall in the areas of SEP plants which are being reevaluated. This will be included as a portion of the overall seismic evaluation capability of the plant.

In response to a question the Staff indicated that the recommended pump endurance run of 72 hours for auxiliary feedwater systems had been reduced down to a 48 hour test. The Staff indicated that they wanted to run the pumps for long enough to have the bearing oil pump temperature come up to equilibrium and to identify any problems that might be associated with long term running, however, they did not want to wear out the pumps. The 72 hour requirement was reduced because in some plants it contradicted the technical specification.

Dr. Zoltan Rosztoczy reviewed the work of the Task Force in the analysis area. This included small break loss of coolant accidents, the anaylsis methods associated with a small break, the variation of the evaluation of the small break with single failure, the evaluation with some degraded protection, and the frequency of small break loss of coolant accidents. Also considered was inadequate core cooling, evaluation of off normal transients and accidents, HPI termination criteria, reactor coolant pump trip, and various needed experimental programs to gain better understanding of small break loss of coolant accidents. The purpose of the review was four fold and included a desire to see if there was sufficent understanding and knowledge of plant behavior in the case of a small break LOCA and whether this knowledge had been properly utilized in the preparation of generic operating guidelines and in the training of reactor operators. In addition, the review was intended to see if licensees had taken necessary actions to substantially reduce the likelihood of

TMI type events and that the near term actions necessary for the safe operation of operating plants have been implemented. In response to a question Dr. Rosztocy indicated that the Staff had not reviewed an accident consisting of a secondary side blowdown with a stuck open relief valve on the primary side. Mr. Michelson indicated that this was a very likely event if one considered that the primary system might be overfilled and the relief valves might have to open following a secondary site blowdown.

In the area of small break LOCA analysis methods the NRC had a number of concerns. These included the ability of the computer programs to correctly predict natural circulation and interruption of natural circulation, the appropriateness of the pressurizer and surge line models to correctly predict pressurizer pressure and water level, the appropriateness of reactor core coolant level and heat transfer calculations, the details of system modeling necessary to predict pressurization, the use of the equilibrium assumptions during repressurization of the partially voided system, and the discharge rate of two phase fluid through the relief and safety valves. One of the difficulties in evaluating small break analyses is that the experimental verification of the small break methods is limited and there are large uncertainties involved. The NRC reviewed the evaluation of system modeling, sensitivity studies on noding, selective submodels and assumptions, and looked at the experimental data available to verify the models. The NRC also ran a number of audit calculations. During the NRC review of the analysis methods a number of shortcomings were identified in various areas. These include the need to correctly node the primary vessel, the fact that equilibrium pressurizer models tend to underestimate system pressure when there is an insurgent to the pressurizer, surge line models do not have a flooding check, drainage from various portions of the system (Upper plenum, guide tubes) could be slow and they need to be modeled appropriately, the uncertainty of break flow calculations is large, the discharge of two phase relief and safety valves is unknown, models tend to overestimate flow in the two phase region, codes cannot handle non-condensable gases, modeling of safety injection tank discharges (condensation model) an unrealistic, and detailed core nodilization is needed to correctly predict steam superheat and cladding temperatures. Also identified was some anomalous behavior in the system pressure and pressurizer level in B&W plants during certain transients. Under certain circumstances pressure appears to hang up for short periods of time rather than continue a smooth depressurization that would expected from calculation. It is felt the unexplained behavior might be due to flashing of isolated hot fluid in regions of upper portions of the hot leg. Safety concerns arise because behavior of the system is not adequately predicted by the analysis methods, and voids may be formed which could migrate to the top of the candy cane and stop natural circulation if offiste power was lost and the pumps were stopped. Flashing could also mask small breaks and prevent timely ECC actuation. These concerns continue to be addressed by B&W and the NRC Staff. Dr. Rosztoczy indicated that another area of concern was with the nodilization of the accumulator injection point. Depending on how the nodilization is done various pressure oscillations can occur in the cold leg. Under some conditions the accumulator water is sucked into the cold leg due to the depressurization caused by the cold water, under other conditions large pressure oscillations may occur in the calculations. This is one area where information from experiments in LOFT and Semiscale may help provide answers as to discharge flow from the accumulators.

Dr. Rosztoczy reviewed the audit calculations that the NRC has had performed by EG&G at Idaho. In general the calculations showed generally good agreements with the calculations performed by NSSS vendors. Some of the audit calculations and the vendor calculations differences could be attributed to variations in the input assumptions in the calculations. The NRC Staff concluded that the methods presently being used are generally satisfactory for the purpose of predicting overall plant behavior. However, they have found a number of shortcomings and found a need for improvements in the models so that more accurate calculations can be performed. They also found that probably there has not been enough attention paid to specific small break types of problems over the last several years and they believe it is appropriate to review the methods and the verifications of the methods that are being used presently for small breaks. Three basic recommendations came out of the NRC analysis review. The first is that the analysis methods for small breaks should be revised, documented, and submitted to the NRC for review within six months, The second recommendation provided for plant specific calculations based on the new approved methods to show that the complete spectrum of small breaks are meeting the present Appendix K acceptance criteria. The third recommendation would include a review and a revision of the conservatisms required in small break analysis by the NRC Staff. The first two recommendations are included in the Bulletins and Orders Task Force Report. The third recommendation, that the NRC Staff revise required conservatisms will be included as one of the Task Action Plans. Dr. Plesset and other Subcommittee members raised several questions regarding the timing of recommendations number 1 and number 3. They believe it may difficult for the NSSS vendors to revise their calculations when the NRC Staff has not yet specified the required conservatisms in the small break. Mr. Waters of Carolina Power and Light representing the W Owners Group also expressed the concern with the tightness of the schedule on revising various analysis calculations. He indicated that a great deal of effort and work has gone into doing calculations over the last six to eight months and that perhaps a higher priority needs to be placed at the present time on revising the procedures and guidelines for the operators so that they better understand what is happening in their plant and can control it.

In response to a question Dr. Rosztoczy indicated that improper operator reaction was not addressed in the various calculations performed in their studies. Another recommendation made by the analysis review group involved specifications on allowable down time for various safety systems. Specifically mentioned was a requirement for availability on BWR high pressure safety injection systems. Present technical specifications allow them to be down or inoperable for up to fourteen days. However, there is no cumulative down time requirement on the system. Thus, the system may be down for fourteen days, restarted and then be taken down again for fourteen days making it largely unavailable. Dr. Ross indicated that this item was going to be a general recommendation that cumulative down time of various systems be studied.

The Subcommittee and Staff discussed the ability of various PWR systems to depressurize through use of the PORV and at the same time get high pressure injection water into the core in order to cool it in the event of the loss of auxiliary feedwater. It is questionable in the number of designs as to whether or not adequate relief capacity is available to reduce the pressure sufficiently in order to get injection water into the core. This is an area that will need further study in the future, particularly in those plants with low HPI pumps.

A number of recommendations came out of the review by the analysis group. The first relates to reactor coolant pumps. The Staff believes until a better solution can be found the reactor coolant pumps should be tripped automatically. The Staff is also recommending that for BWR transients which open the relief valves that the licensees should demonstrate that the core remains covered or provide assurance that there will be no fuel damage. Another area of recommendation involves the HPI injection capability and relief capacity. The Staff is recommending that additional relief valves be provided so that sufficient depressurization capability will be installed in the event loss of all feedwater high head be provided so that sufficient decay heat can be removed through the code safety valves at high pressure. Dr. Rosztoczy noted that the increased relief capability would tend to agree with recommendations that are coming out of the ATWS review about the need for additional relieving capability. Another possibility is the addition for high pressure RHR system which would eliminate the need to depressurize the primary system to get rid of the decay heat from the core. Dr. Ross indicated that the additional relieving capability of high head HPI pumps or a high pressure RHR system would be included within the NRC Action Plan rather than as an item that must be addressed through the Bulletins and Orders Task Force Report. Additional recommendations made by the B&O Task Force include the following:

1) That BWR licensees should evaluate other possible depressurization modes besides full actuation of the ADS system This would perhaps allow additional depressurizations - the ADS system can be used only twice.

2) Plant simulators should be updated and they should offer as a minimum at least each of the basic types of small breaks that have been identified during the past months.

3) The isolation logic of the HPCI and the RCIC system should be modified to prevent unwanted isolation of the systems.

4) Two reactor operators should be designated to restart the HPCI and the RCIC systems until the previous recommendation is implemented.

5) BWR ADS initiation should be automatic for those transients where the dry well does not increase in pressure. At the present time drywell pressure is one of the coincidence signals that is needed for ADS and for certain small breaks the pressure increase may not be noted.

An additional area that the NRC Staff reviewed was the actuation frequency of PORVs and safety valves. In reviewing the available information from the PWR vendors, the NRC Staff noted that the number of relief valve challenges is a strong function of the system design. The estimated relief valve failure rate per reactor year is 0.2, 1.0, and 2.0 for B&W, CE, and W designs respectively. BWRs have a significantly higher relief valve challenge rate, approximately 15 valve openings per year. In general, the available data supports the above estimates, there is, however, a major discrepancy between the W analysis and the W data. The observed failure frequency of valves per opening varies between 1 and 20 and 1 and 100. The WASH-1400 value is one per hundred. The probability of a stuck open relief valve is the same order of magnitude as a small pipe failure for B&W plants, it is an order of magnitude higher than pipe failure for W and CE plants and it is two orders of magnitude higher for GE plants. Careful selection of relief valves and overpressure reactor trip setpoints together with anticipatory reactor trips on turbine trip and loss of feedwater can reduce the expected frequency of stuck open relief valves in PWRs by an order of magnitude. Automatic closure of PWR block valves in case of a small LOCA, or operating PWRs with closed block valves could also reduce the risk associated with stuck open relief valves. The various ways available to reduce the SRV challenge rate of BWRs have not yet been fully evaluated by the licensees. The records kept on relief and safety valve challenges are unsatisfactory. The data provided by licensee on relief and safety valve challenges are incomplete and need to be improved. The NRC Staff is recommending that the frequency of relief valve challenges should be reduce substantially in CE, W, and GE plants. The discrepancy between the W analyses and W data on relief valve challenge rate during feedwater transients should be resolved. A report should be prepared on safety and relief valve challenge rate and failure rate based on past history of the various designs. All future safety and relief valve challenges should be recorded and reported to the NRC.

The NRC Staff indicated that are a number of possible ways to reduce BWR SRV challenge rates. These include additional anticipatory scram on loss of feedwater, revised relief valve actuation setpoints, increased ECC flow, lower operating pressures, earlier initiation of ECC systems, heat removal through emergency condensers, offset valve setpoints to open less valves per challenge, and installation of extra relief with a block or isolation valve feature to eliminate the opening of the normal SRVs consistent with ASME code requirements.

In summary Dr. Rosztoczy concluded by indicated that the possible failure of a relief valve in a nuclear plant to close as it happened at TMI is a generic industry-wide problem applicable to all U.S. designs including PWRs and BWRs. In the present mode of operation of the plants the GE BWRs are expected to experience the largest number of relief valve failures followed by the W PWRs, the CE PWRs and finally the B&W PWRs. BWR relief valves are typically ten times larger than PWR relief valves. Consequently, failure of the BWR valves to close could have more serious consequences and could result in reactor core uncovery while uncovery is not expected for PWRs. PWRs with low cut off head HPI pumps are not protected for the extended loww of all feedwater and for the extended loss of natural circulation events. All CE plants, half of the W designs, and Davis-Besse among the B&W designs fall into this category. Calculational methods used for small break LOCA analyses have not yet been properly verified and have large uncertainties. The uncertainties of the calculations possible exceed the ECCS acceptance criteria. Appropriate corrections have been recommended for the existing shortcomings. The recommendations when implemented will provide reasonable assurance that continued operation of the plants does not represent an undue risk to the public health and safety.

Another area that was initiated by the Bulletins and Orders Task Force but was not completed by the B&O Task Force involves inadequqte core cooling. Under the Lessons Learned Task Force, recommendations were mad for instrumentation on detection of inadequate core cooling for PWRs adn BWRs. A number of meetings were held with the utility Owners Group to specify the extent of the inadequate core cooling study. The purpose of the study is to develop emergency procedures and to identify the essential instrumentation needed to follow the procedures. The inadequate core cooling procedures will be independent of the initiating event. Operator action will be determined by the observed condition of the core in the reactor system. The Onwers Groups have completed the initial evaluation of inadequate core cooling and submittals have been received from each group during the months of November and December. Staff evaluation of the ICC submittals will be starting shortly, however, this work will not be completed under the B&O Task Force.

The NRC Staff briefly reviewed the effects of leaving the reactor coolant pumps on for small break LOCAs. (This matter was reviewed in detail at an October ECCS Subcommittee meeting). The basic problem involves the loss of additional mass out the break during a small break LOCA when the reactor coolant pumps are left on.

The additional loss of mass inventory, if a reactor coolant trips at the proper time, leads to a deeper level of core uncovery and consequently higher peak clad temperatures. Each of the PWR vendors modeled the accident in slightly different ways. Each came to the conclusion that leaving the reactor coolant pumps on was worse than tripping them immediately. The W plants found that cold leg breaks were the worst location, B&W also found that cold leg breaks were the worst, however, CE found that hot leg breaks were the worst. Part of this may be due to modelling differences, CE allows counter current flow in the hot leg and because of this, fluid running from the steam generator back down the hot leg toward the vessel leaves the small break at the bottom at a rate greater than fluid exiting through a cold leg break. B&W and W models do not allow this counter current flow, and consequently there is a higher mass flux out the break for cold leg breaks. It's not clear that if the modeling was done in a consistent way that there would be these differences between the analyses. If the pumps could be left on for the duration of the accident, B&W and W indicates this would be acceptable, however, for CE plants leaving the pumps on for the duration of the accident also leads to peak clad temperatures that are above Appendix K acceptance levels. The NRC Staff has concluded that the pumps can not be guaranteed to run for the duration of the accident and thus must be tripped immediately after initiation of HPI. The break sizes which result in difficulties with the pumps running range in size from approximately one inch to three inches in diameter. Tests will be run in LOFT with pumps on and the pumps off in an attempt to clarify some of the problems noted in the calculations done by the NSSS vendors. In the first of the LOFT small break tests run in November of 1979 it was noted that the bypass area between the upper plenum and the downcomer region played an important part in phenomenum observed during the test. This bypass area allowed steam flow out the cold leg break from the upper plenum region without travelling through the steam generators and the rest of the loop piping. This bypass area has generally not been modeled in the various analyses and thus, the overall system behavior is considerably different than actually occurred in the test. The NRC Staff indicated that during the LOFT small break test in November the operators were not completely aware of what was happening and it was only through the analysis in the two months since the test that the phenomena involved have been identified. Dr. Okrent noted that it was interesting that even in a well instrumented test facility with extremely capable operators the events during the course of an accident were not readily identified. In this light, one wonders how operators in commercial plants will be able to identify various phenomena, especially if the accident does not follow a previously identified path. Another area of interest in the LOFT test was behavior during the accumulator injection. The observed test results show a smooth injection of accumulator water with no pressure oscillations caused by unstable condensation in the cold leg. There was no accelerated flow of ECC water into the cold leg due to depressurization caused by the cold ECC water mixing with hot steam. Additional tests to provide data for improved small break calculations will be provided by Semiscale and TLTA. GE has been asked to perform predictions for two small tests to be conducted in TLTA.

Dr. Harold Sullivan, NRC/RES, reviewed a number of programs that will be undertaken during FY 80, 81, and 82 to examine various small break phenomena. Plans include tests in LOFT, Semiscale, TLTA, THTF, Flecht-Seaset, and perhaps some pump tests at a Bingham-Willamette facility where they have the capability of testing full scale B&W pumps. LOFT, Semiscale and TLTA will include small break tests. THTF and Flecht Seaset will include various tests on natural circulation, fluid boil-off and heat transfer. The natural circulation in Flecht-Seaset will include both single phase, two phase, and reflux boiling. A pump performance test may be run at the Bingham-Willamette facility in conjunction with B&W and EPRI.

Dr. Plesset recessed the meeting at 7:30 pm to reconvene at 8:30 am the following day.

Mr. Bruce Wilson, NRC Licensing Branch, reviewed the new guidelines and emergency procedures for the operators during small break loss of coolant accidents. The guidelines were reviewed with respect to the following critical operational actions; ECCS termination criteria, reactor coolant pump trip for PWRs, clarification of safety system actuation and verification of the heat sink. The termination criteria as a minimum was a specified amount of subcooling and pressurizer level indication for the PWRs and multiple confirming indication of vessel levels for the BWRs. Tripping of the reactor coolant pumps was required on all PWRs at or slightly below the primary pressure setpoint for engineered safeguards actuation. I&E Bulletin 79-05A and 06A required, in the event of an ECCS actuation due to low pressure, HPI must be kept on for a minimum of 20 minutes with all hot and cold leg temperatures 50 degrees subcooled or until low pressure injection system injection at 1000 gpm has been stabilized for 20 minutes. The Staff recognized that keeping HPI on for 20 minutes would probably result in challenging the PORV or safety valves for most ECCS actuations, so this was the first of the criteria that they changed. The 20 minute criteria was removed and 50 degrees of subcooling was made the basis for termination of HPI flow. The B&W guidelines were reviewed and approved within a period of about one week following their issuance whereas with W, CE, and GE it took several months. The diversity of plant designs with the later three vendors made it particularly difficult to develop guidelines that were generically applicable. For W two sets of guidelines were developed. One for the standard 412 plant and the other for the plants having high pressure injection pumps with a relatively low pressure injection head. Additional time was also called for W plants due their objections to meeting 50 degree subcooling prior to HPI termination. This requirement has been modified for W plants and HPI termination criteria based upon temperature and pressure measurement inaccuracies that are added to a 20 degree farenheit subcooling requirement. W also has a requirement that auxiliary feedwater flow be verified by assuring the steam generator water level has been raised up into the narrow range instrumentation span. The NRC Staff also checked to assure that the guidelines prepared by the NSSS vendors were applied by the utilities in revising the small break LOCA procedures. In addition, the NRC Staff reviewed the training that was given to the operators on the small break phenomena, the procedures, and they also audited the operators themselves to look at their understanding of the procedures that have been recently implemented and the events of the TMI accident. Mr. Wilson indicated the original purpose of the audit was to insure that the revised emergency procedures for small breaks had conformed to the approved guidelines. As a by-product of that examination, the Staff realized it was also necessary to talk to the operators and assure that they themselves understood what the procedures contained. The NRC Staff found a number of problems with the implementation of the guidelines in the individual procedures during their audits. These included several places where utilities had not implemented the guidelines because they felt that they were unsafe, misinterpretation of what guidelines required, and lack of understanding of what the guidelines required. The audits of the operators themselves covered the TMI-2 accident, small break phenomena, LOCA procedures, and facility design changes related to the procedure changes. The audits themselves were conducted by the Operator Licensing Branch and the I&E inspectors in an oral review of the new procedures. A number of problem areas were found, these included some difficulty in the operators explaining why the pressurizer level was going up at TMI-2 as the pressure was decreasing as well as a great deal of difficulty understanding and explaining thermohydraulics involved in maintaining a subcooled system.

Mr. Wayne Hodges reviewed the audit of procedures that were implemented from the BWR guidelines. Again, the purpose of the audit was to review some selected licensees and their procedures, the operator retraining, the awareness of the procedures and to look at some systems considerations as far as the implementation of the small break LOCA. The procedures were developed from guidelines that were developed by the BWR Owners Groups which consisted of all the operating licensees plus GE. The guidelines were then approved by the Bulletins and Orders Task Force and were developed into procedures. A number of different GE plants were visited to review the procedures and interview operating personnel. In reponse to a question Mr. Hodges indicated the GE simulator at Morris, Illinois did a poor job of simulating small breaks LOCAs. They do a good job on big breaks and steam line breaks but they cannot get a small break LOCA. They cannot even do a good job of simulating stuck open relief valve. Consequently, the simulator will not be a very good tool for training the operators at the present time. One of the NRC recommendations is that they should upgrade the simulator considerably. Mr. Hodges noted that the Browns Ferry simulator in Chattanooga, Tennessee was a much better simulator and was better equipped to handle small breaks. In general, the age of the simulator determines the degree of sophistication of the transients that it can model. Dr. Ross noted that under the NRC Action Plan there were a number of items related to simulators. These included short term study of training simulators, interim changes in training simulators, and the starting of a separate program by NRC on research on various training simulators. Finally, after enough study was done, the training simulator standards would be upgraded on a continuous basis, and the NRC may issue a Regulatory Guide on requirements for the simulators. The NRC has also considering procurement for a training simulator for itself as well an engineering simulator.

Mr. Hodges indicated that they looked at a number of things during their audit. These included a comparison of the guidelines with the procedures to make sure that the guidelines themselves were being implemented properly, the clarigy of the operation actions, and cautions, the flow of the procedures with respect to timely actions, and the operator retraining, both the formal retraining in the classroom and also the walk through by the shift supervisor and the training coordinators. The Staff also looked at the operator awareness of the procedures, the basis for the procedures and at the various systems considerations that went into the implemention of the procedures. A number of concerns were identified by the Staff and the various utilities were asked to address these concerns. At a number of plants the training was still incomplete at the time the visits were made. At a number of plants the NRC Staff identified concerns with the level of training that the operators had and their understanding of the reactor vessel level instrumentations and the temperature compensation that it receives. In some cases, operators within the same plant would be using different methods to verify containment isolation. The Staff did agree that there several ways to do it, each equally valid, and that each operator was taking his own particular way. The Staff suggested that it was probably a good idea to establish guidelines that called for verification of containment isolation by two independent means. In some cases, the Staff identified actions that the operators were required to take that may be difficult due to the positioning of various instruments and valve actuation switches. In some cases valves were located at least 20 feet from level recorders and it would be difficult for the operator to follow reactor vessel level while actuating the manual ADS function. The recommendations the Staff has made include at least two operators in the control room rather than the one that's required at the present time. Another problem with BWR level instruments is that the overlapping instruments generally do not have a common zero. The reference point of several instruments may be at different points in the vessel and thus lead to confusion.

With regard to the BWRs the Staff made a number of recommendations in the Bulletins and Orders Task Force. These recommendations include; (1) the separation of HPCI and the RCIC initiation signals, the isolation of the isolation condenser when a high radiation signal is received at the isolation vent rather than in the steam line leading to the isolation condenser, the changing of the pressure taps to avoid spurious isolation of HPCI and RCIC, the reduction of challenges to safety relief valves, possibly through the anticipatory scram on loss of feedwater or revising the setpoints on the relief valve actuation, the identification of water sources that are available for cooling at low pressure before manual actuation of the ADS system, the reporting of outages of ECC systems, that interlocks be installed on non-jet pump plants other than Humboldt Bay to insure that at least two recirculation loops are opened for recirculation flow for modes other than cold shutdown, for Big Rock Point the licensee should verify the acceptability of the consequences of a loss of service water supply to a central plant component in the event of loss of offsite power, the modification of the LPCI system so that they are able to restart automatically if necessary, and the requirement that operators review all emergency procedures that have been implemented or modified since the previous shift. As an added item the NRC Staff has asked GE to go back and review all the changes made in the Bulletins and Orders recommendations and do a study of the total impact on the plant to make sure that there are no synergistic effects between the changes that will lead to detrimental plant performance. The NRC Staff has also suggested that the RHR system and the spent fuel pool decay heat removal systems be separated.

Mr. Bill Kane summarized the items required by the Bulletins and Orders Task Force in both the short term and the long term that had been discussed in the prior presentations.

Mr. Chuck Domeck of Toledo Edison Company represented the owners of the B&W reactors. Mr. Domeck indicated that the B&W owners formed the TMI-2 Incident Technical Subcommittee in early April 1979. The Technical Subcommittee for TMI-2 was formed under an existing organization of B&W owners that were addressing other generic items related to B&W plants. Approximately montnly meetings have been since April of 1979. The B&W fuel owners group charter permits utility participation in any or all of the active issues as long as the utility is currently designing, constructing or operating a plant utilizing the B&W 177 Fuel Assembly Nuclear Steam Supply System. An extensive number of generic issues and analyses are being accomplished under the technical direction of this owners group. The TMI-2 Subcommittee of the Owners Group includes topics related to safety grade anticipatory reactor trips, auxiliary feedwater reliability studies, abnormal transient operating guidelines, reactor coolant system venting requirements, power operated relief valve actuation indication, and small break LOCA analyses. Tasks are undertaken by B&W under the direction of the TMI-2 Subcommittee if sufficient utility interest and financial support exist. On some tasks seven utilities are participating, on other tasks only four or five utilities are participating. Mr. Domeck noted that the TMI-2 Subcommittee does not constitute a legal entity and has no financial or legal obligations for its member companies. Formal requests are still made by the NRC to the licensee and formal responses in writing are provided on the individual utility dockets. In general, the owners group is coordinating the responses and helping each of the individual utilities develop responses to the items addressed by the Bulletins and

Orders Task Force. There are some differences between the various plants in implementing the NRC Staff requirements and recommendations, for example, on Davis-Besse the auxiliary feedwater system is controlled by a safety grade initiation and on other plants a control grade system is being installed by the first of January 1980 and the safety grade system will not be installed until January 1, 1981.

In response to a question from Mr. Michelson, Mr. Geisler of B&W indicated that the subcooling meter being installed in the B&W plants (manufactured by B&W) is strictly a subcooling meter and it does indicate degree of super heat. Mr. Michelson indicated some concern with this and the Subcommittee members felt that it would equally easy to indicated super heating as well as subcooling.

Mr. Domeck indicated that with respect to the criteria to be used for tripping reactor coolant pumps, the majority of the B&W technical subcommittee members agree with utilizing coincident input signals with low pressure emergency safety features actuation combined with low reactor pump coolant current or power. Sacramento Municipal Utility district however, believes that rather than low reactor system pressure coincident with low reactor coolant pump current, the important parameters should be evidence of subcooling or if the reactor coolant system is operating outside the temperature pressure bands of their applicable technical specifications. Overall, the B&W Subcommittee position is that tripping the reactor coolant pumps for certain small breaks does provide the least overall risks considering the spectrum currently analyzed events.

Mr. Domeck indicated that the B&W owners group was concerned about a number of things. These include extensive backlogs of license amendment requests, many of which relate to items addressed by the B&O Task Force, and the extensive requirements for analyses and work that the NRC Staff has asked the owners to perform since the TMI-2 accident. The extensive efforts being required may, in many cases, overtax the ability of B&W and the other groups to perform the needed analyses in a reasonable manner without overlooking items and making mistakes.

Mr. Domeck also reviewed a depressurization incident at the Toledo Davis-Besse plant that the Staff had commented on during the first day's portion of the meeting. Mr. Domeck indicated that the Toledo Edison staff had reviewed the incident and that they did not see any basis for flashing of hot water causing the abnormalities seen in the pressure decrease during the transient. Dr. Ross indicated that the Toledo Edison transient was not the only transient that the NRC Staff was concerned with. A number of other transients at B&W reactors have shown the same abnormalities during a depressurization incident. He indicated that the NRC Staff would be pursuing this with the other B&W owners as well as B&W.

In response to a question Mr. Domeck indicated that the B&W owners group Subcommittee on TMI-2 would continue throughout 1980 and probably into 1981. The Owners Group itself will probably continue indefinitely until such time that the owners decide that it is not a useful way to coordinate activities with the NRC Staff.

Mr. Rogers, Pacific Gas and Electric Company, provided the subcommittee with an overview of the work done by the BWR Owners Group. The BWR Owners Group membership consisted of all the utilities which own operating BWRs in the US and all of the so-called near term operating license applicants. In addition, two overseas utilities from Tokyo Electric Power and Japan Electric Power participated. The work of the owners group was formed to address NRC Staff concerns in the generic post-TMI-2 activities. This work involved the B&O Task Force and the Lessons Learned Task Force. The Owners Group addressed small break accident analyses and guidelines, loss of feedwater incidents, and stuck open relief valve analysis. In addition, they did a natural circulation study that provided the NRC with information on how water levels in the reactor are measured both on a generic and specific basis. The BWR owners group will also prepare a reliability fault tree analysis to look at some of the transients involving loss of feedwater in small break LOCAs. The intent of the study is to review operating procedures and guidelines based on existing Chapter 15 Analyses to make sure that procedures and guidelines will not mislead the operator into doing things other than what he should do or expect. The Owners Group is also working on a series of new symptom based operator guidelines and procedures. The new symptom oriented procedures would avoid the difficulty of asking and having the operator decide what the event was and then go into the emergency procedure and act accordingly. The symptom oriented procedure would allow the operator to look at the symptoms and address the symptoms rather than having the difficulty of diagnosing a particular event.

Mr. Rogers addressed a concern that the BWR owners group had with the increasing number of requirements that NRR is placing upon the plant operators. The increase in number of changes in the guidelines and procedures and the increased number of instruments in the control room may not provide additional aid to the operators but rather may cause increasing difficulty for the operator in controlling the plant.

Mr. Rogers indicated that the BWR owners group has had some difficulty in a number of areas. The first area is in the number of requirements being made by the NRC and on the short deadlines available for implementing some of the changes in the plants. There have been cases where they have gotten conflicting points of view about the implementation of certain criteria from different people in NRR. In other cases, the Owners have not gotten timely approval of submittals to the NRC consequently, difficulty in meeting schedules is aggravated.

Mr. Domeck, Toledo Edison, indicated that an additional problem that the B&W owners saw and had a concern with was that scarcity of qualified operations people on the staffs of the architect engineers. Almost without exception, the various utilities have to rely on the architect engineers for advice on their plants, yet many of the architects have only one, two, or perhaps three people on their staffs with operating experience.

Mr. George Liebler, Florida Power and Light, representing the CE owners group, reviewed their position on the Bulletins and Task Force recommendations. Mr. Liebler indicated that the CE owners group also experienced a number of difficulties in implementing recommendations of the B&O Task Force due to the tight schedules required. They experienced difficulties in getting necessary hardware and had to design some systems around hardware available on an off-the-shelf basis. The CE owners group has also initiated symptomatic evaluation of plant event sequences for use in improving plant emergency procedures and operator training. Mr. Liebler indicated that the utilities often have difficulties in finding equipment for their plants that has improved reliability. For example, if the utility wants to buy improved PORVs with increased reliability, they go to one of the valve suppliers and indicate their desire. The often expressed answer by the valve supplier is "it's not a big enough order and it's not worth our time and effort to develop the improved valve." The QA requirements for nuclear equipment and the difficulties in developing special equipment for a limited number of sales often make it uneconomical to provide the equipment. Another of concern is that shortcuts being taken in the development and changes to operator procedures. Prior to TMI-2 the changing of a plant procedure required a period of two to three months between the proposal of the new procedure and its final implementation. This time was required to get proper review and walkthroughs of the procedure to make sure it really was an improvement. Subsequent to TMI-2 a large of procedures have been revised and approved and are being used without having had the thorough study and analyses that the earlier procedures have. There are some concerns that the new procedures, if examined in detail, will turn out to be ineffective or even worse may lead the operator into making mistakes. Mr. Liebler indicated that these procedures have also had an effect upon the morale of the plant operators. This is because procedures that they believe need to be changed often require six months before being implemented, yet TMI-2 related changes that the NRC is requiring are going through in several weeks or a month. The operators realize that the changes didn't get the review that is typically given to the procedures and they are concerned about the double standard that it seems to exist.

Mr. Dave Waters, Carolina Power and Light Company, presented the comments of the W Owners Group. He reported that the initial changes of the operating procedures have been made in a fairly smooth manner. The owners group and W is now going back and looking at some of the other emergency procedures and will be applying lessons of TMI-2 to those procedures as well as other abnormal operating guidelines. Mr. Waters indicated the W owners group may not continue in its format but rather may be reconstituted as a smaller group that serves as a steering committee for individual utilities. This steering committee would help to coordinate the various issues that need to be addressed by the utilities and the various requests that come from the NRC but would serve the function that the owners originally did of performing many of the tasks. Mr. Waters indicated that the W owners group was also concerned with the schedules and the other difficulties expressed by the other three owners groups.

Dr. Plesset and Mr. Mathis each urged that the owners groups and the NRC Staff attempt to work together to resolve their common problems. Each expressed some concern that the present appearance of lack of cooperation and arguing was certainly not the best method to improve reactor safety.

Mr. Mathis thanked the participants of the meeting and adjourned the Subcommittee at 3:40 pm.

Additional information on the Subcommittee meeting may be found in the meeting transcript which is available in the NRC Public Document Room, 1717 H Street, NW Washington, DC or from Ace-Federal Reporters, Inc., 444 North Capitol St., Washington, DC. A complete copy of all the slides presented at the meeting is on file in the ACRS office with a record copy of the minutes.

The NRC staff determined that the event demonstrated a weakness in the licensee's ability to control testing and maintenance activities, to develop and review procedures, to adhere to approved proceures and to conduct audit activities. The Director, Office of Inspection and Enforcement (IE), also determined that the potential public hazard had been high. As a result on November 9, 1979 the staff proposed imposition of civil penalties in the amount of $450,000.00 for the prolonged violation of containment integrity. On the same date, the staff issued an Order to require that appropriate review of checklists and procedures be performed to assure that engineered safety features are in compliance with the specifications of the license and that monthly inspections of these features be conducted. The Order further required a meeting with NRC management prior to resumption of operation.

IE Information Notice 79-28 was issued on November 5, 1979 to all holders of operating licenses and construction permits to provide them with the details of this occurrence. On November 16, 1979 the Director, Office of Inspection and Enforcement, sent a letter to chief executives of all utilities with operating licenses and construction permits informing them of the enforcement action against Consumers Power Company and stating the intention to take similar action in any future instances where ineffective management leads to a serious breach of safety.

Dated at Washington, D.C., this 13th day of December 1979.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 79-38803 Filed 12-18-79; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Ad Hoc Subcommittee on Three Mile Island, Unit 2, Accident Bulletins and Orders and the Emergency Core Cooling System Subcommittee; Meeting

The ACRS Ad Hoc Subcommittee on the Three Mile Island, Unit 2 Accident Bulletins and Orders and the Emergency Core Cooling System Subcommittee will hold a joint meeting in January 3-4, 1980 at the Airport Park Hotel, 600 Avenue of Champions, Inglewood, CA 90301 to consider the NRC Office of Inspection and Enforcement Bulletins and NRC Orders pertaining to the TMI-2 Accident, including information related to the small-break loss of coolant accident. Notice of this meeting was published December 20, 1979.

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 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:

Thursday, January 3, and Friday, January 4, 1980

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

The Subcommittees may meet in Executive Session, with any of their 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 Subcommittees will hear presentations by and hold discussions with representatives of the NRC Staff, the nuclear industry, various utilities, and their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittees 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 or 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, Dr. Andrew L. Bates (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m. EST.

Background information concerning items to be discussed at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555 and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, PA 17126.

Dated: December 13, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-38804 Filed 12-18-79; 8:45 am]

BILLING CODE 7590-01-M

[Dockets Nos. 50-315 and 50-316]

Indiana & Michigan Electric Co.; Notice of Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 34 to Facility Operating License No. DPR-58, and Amendment No. 15 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

The amendments require the maximum allowable purge isolation valve closure times to be reduced from 10 to 5 seconds.

The application for the amendments complies 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 these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1), the application for amendments dated February 3, 1978 and March 7, 1979, (2) Amendment Nos. 34 and 15 to License Nos. DPR-58 and DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Maude Reston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon

ATTENDEES LIST

NAME	AFFILIATION
M. Plesset, Chairman	ACRS Member
W. Mathis	ACRS Member
H. Etherington	ACRS Member
D. Okrent	ACRS Member
A. Acosta	ACRS Consultant
T. Wu	ACRS Consultant
V. Schrock	ACRS Consultant
Z. Zudans	ACRS Consultant
W. Lipinski	ACRS Consultant
C. Michelson	ACRS Consultant
P. Matthews	NRC
D. Ross	NRC
Z. Rosztoczy	NRC
W. Hodges	NRC
B. Wilson	NRC
W. Kane	NRC
H. Sullivan	NRC
D. Verrelli	NRC
Tolbert Young, Jr.	NRC Region RV
R. Macduff	Exxon Nuclear
J. Owsley	Exxon Nuclear
C. Domeck	Toledo Edison
G. O. Geissler	B&W
R. Leyse	EPRI
R. Skwarek	W
D. Waters	Carolina Power & Light
D. Ward	IXUS (Representing GPU)
P.W. Marriott	GE
D. Hill	The Daily Breeze
T. Rogers	Pacific Gas & Electric
S. Stakr	GE
J. Vorees	Boston Electric
J. Holderness	CE
G. Liebler	Florida Power & Light
P. Morrill	NRC Region V I&E

A. Bates, Designated Federal Employee

Attachment B

ECCS/B&O SUBCOMMITTEE MEETING
TENTATIVE SCHEDULE
JANUARY 3-4, 1980

	PRESENTATION TIME	ACTUAL TIME
JANUARY 3, 1980		
I. Introduction		
W. Mathis - M. Plesset	5 min	8:15 am
II. NRC Bulletins & Orders (B&O) Task Force Efforts		
A. Overview of Efforts and Future Plans D. Ross - Chairman, B&O Task Force (NRC)	15 min	8:20 am
B. Status of Task Force Efforts RE: Bulletins & NRC Orders - W. Kane (NRC)		
1. Response to I&E Bulletins	15 min	8:50 am
2. Status of B&W Plant Responses to NRC Orders (Long-term Requirements)	30 min	9:20 am
- BREAK -	10 min	10:05 am
C. Review of Plant Auxiliary Feedwater Systems, and Analysis of Design and Off-Normal Transients & Accidents - P. Matthews (NRC)		
1. Auxiliary Feedwater System Upgrade for PWRs (Incl Lessons Learned Items 2.1.7.a and 2.1.7.b) o Program Status (W, CE & BW) o Implementation of Requirements o Problem Areas	60 min	10:15 am
- LUNCH -	60 min	12:00 noon
2. Analysis of Design and Off-Normal Transients and Accidents (Lessons Learned Item 2.1.9) - Z. Rosztoczy (NRC)		
o Small Break LOCA Analyses - W & CE - B&W (NUREG-0565) - GE	30 min	1:00 pm
o Inadequate Core Cooling (Incl Lessons Learned Item 2.1.3.b.1)	30 Min	2:00 pm
o RCP Trip (NUREG-0623) and HPI Termination Criteria	15 min	2:45 pm
- BREAK -	10 min	3:05 pm

ATTACHMENT #C

	PRESENTATION TIME	ACTUAL TIME
o LOFT, Semiscale and Two-Loop Test Apparatus (TLTA) Prepredictions	15 min	3:15 pm
D. Vendor's Emergency Guidelines, Plant Emergency Procedures and Operator Training - B. Wilson (NRC)	15 min	3:40 pm
1. Approval of Vendor Emergency Guidelines		
2. Plant Emergency Procedures and Operator Training		
o Transient & Accident Scenarios - Incl Operator Actions Not Previously Analyzed - Small Break LOCA - Inadequate Core Cooling - Expanded Chapter 15 Analyses	45 Min	4:00 pm

- RECESS -

JANUARY 4, 1980

W.Hodges (NRC)

	PRESENTATION TIME	ACTUAL TIME
o NRR/I&E Training Interface	15 min	8:30 am
o Results of B&O Task Force Plant Audits of Implementation of Plant Emergency Procedures and Operator Training	30 min	8:50 am
o ACRS Comments on Operator Action	10 min	9:30 am

- BREAK -

	PRESENTATION TIME	ACTUAL TIME
E. B&O Task Force Vendor Generic Reports		
1. Recommendations in Generic Reports		
o BWRs - System Review - W. Kane (NRC)	45 min	9:55 am
o PWRs - All Recommendations not covered above - W. Kane	15 min	11:00 am
o Implementation of Recommendations	15 min	11:20 am
2. PORV/Reactor Trip Setpoint Recommendation - D. Ross (NRC)		
o Status of W, B&W and CE Plants	10 min	11:40 am

- LUNCH -

		PRESENTATION TIME	ACTUAL TIME
III.	Plant Owners' Group Presentations		
	A. GE Owner's Group - S. T. Rogers (PG&E)	30 min	1:00 pm
	B. B&W Owner's Group - C. Domeck (Toledo Ed)	30 Min	2:00 pm
	- BREAK -	10 min	2:45 pm
	C. CE Owner's Group - G. Liebler (Florida Power and Light)	30 min	2:55 pm
	D. W Owner's Group - D. Waters (Carolina Power and Light)	30 min	3:30 pm
IV.	Concluding Remarks	15 min	4:15 pm
V.	Adjourn		4:30 pm