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ACRS-1642

MINUTES OF THE ACRS SUBCOMMITTEE ON THREE MILE ISLAND-2 ACCIDENT IMPLICATIONS MAY 31, - JUNE 1, 1979

The TMI-2 Accident Implications Subcommittee met on May 31, - June 1, 1979, at 1717 H St., NW, Washington, DC. The main purpose of the meeting was to discuss with the NRC Staff regarding their augmented research as a result of the TMI-2 accident, and with representatives of Babcock and Wilcox about actions and research they have taken.

Notice of the meeting was published in the Federal Register on May 16, 1979. Copies of the notice, meeting attendees, and schedule are included as Attachments 1, 2 and 3, respectively. A complete set of handouts and viewgraphs is kept in the ACRS Office. The Designated Federal Employee for the meeting was R. K. Major. No written statement nor request for time to make oral comments were received from members of the public.

EXECUTIVE SESSION

Dr. Okrent, Subcommittee Chairman, convened the meeting at 8:30 a.m., introduced the ACRS members and consultants who were present and indicated that R. Major is the Designated Federal Employee.

The Subcommittee should decide on which things should be fixed, is fixable, and not just anything because there is an excuse now.

NRC STAFF PRESENTATION (8:35 a.m. - 2:30 p.m.)

Mr. R. Budnitz

Mr. R. Budnitz (Deputy Director of Research) showed graphically that accidents can be classified into three major regions: core melt accidents, which are well studied in WASH-1400, design-basis accidents, which are well studied in research and in licensing review, and accidents whose consequences are in between. The last type of accidents can lead to extensive core damage, though not core melt. Unfortunately, very little work was done on this type.

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Dr. Mark remarked that core melt accidents do not necessarily have fantastic consequences.

Research is proposing to the Commission for supplemental funds to do work on accidents which would result in extensive core damage. A total of $29.1 million would be requested, with almost half of it going to the study of transients and small LOCA events. Detailed breakdown of the resources can be found in the hand-out. There was the concern that the national laboratories have limits on man-power and computer time, but Dr. Murley replied that they have not encountered man-power problems so far. Specifically, INEL, Sandia, Oak Ridge, and Los Alamos indicated that they do not have man-power problems. Most laboratories today give NRC research activities very high priority, right after military and fusion programs. Thus, the only possible obstacle to performing the proposed supplemental research would be from the Commission or the Congress. Some of the proposed work items can begin now, without the supplemental money, since the ongoing research programs can be modified to accommodate them.

The first item on the proposed supplemental research is "transients and small LOCA events." (This would account for $13.4 million.) The goal of most of this work, as in large LOCA studies, is to develop improved computer codes that could predict the behavior of nuclear systems under small LOCA conditions. Vital concerns of small LOCAs are the coolability of the core, fission product release/transport, and coolant chemistry. Dr. Mark asked if one might direct all attention to making sure that cores are never uncovered, and thus skip the whole area of small LOCAs. Mr. Levine argued that one does not have the confidence that cores will always be covered. Dr. Okrent observed that he did not find in the supplemental program more comprehensive ways to avoid accidents that could lead to serious core damage. Dr. Okrent further remarked that if Research had studied a range of degraded accidents, one of them would have included a situation where a core was overheated but not melted, accompanied by H_2 generation. By doing a range of degraded accident studies, one would have lots more insight into what may transpire under upset conditions such as at TMI-2.

Research now has a good code, TRAC, for predicting major LOCA systems behavior, and they desire a similarly good one for small LOCA events. In principle, TRAC

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can handle small breaks but the code has not been exercised enough to prove that. Mr. Michelson asked if Research has looked at undesirable operator intervention during the course of transients. Mr. Budnitz said there was no such effort but that Michelson's suggestion is good.

Mr. Budnitz said the small LOCA work would be very similar to the large LOCA counterpart, and can be done easily by people who are doing the latter.

The second item on the proposed supplemental research is "enhanced operator capability" (accounting for $3.3 million). Work in this area consists of developing instrumentation for plant status monitoring, testing these instruments, and improving simulator capabilities for operator training. There was some discussion about NRC-sponsored research programs to develop instruments. The question was if it was proper for NRC to develop instruments and later sell it to a vendor. The question was also raised if NRC would license any instrument which it developed if an applicant proposes to use it. The Staff responded that in years past, millions of dollars were spent by NRC to develop instruments for the LOFT or SEMISCALE programs and that it would have been a waste if these instruments are not put to practical use. Staff approval of an instrument originally developed by NRC research programs would be based on the applicant's justification of the capability of the instrument, in light of what the Staff has already learned about the instrument during its development.

There would be work on level indicators and fuel thermocouples, and it would take about 1 1/2 years to produce useful results on level indicators. The TRAC code would be improved so that it can be run in real time and be used in simulators for operator training. Dr. Siess said that the impact of the TMI-2 accident was inflated by lack of communication and suggested that some social studies be done along this line. Mr. Levine said that such as in fact under way before the incident occurred.

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The third item is "Improved risk assessment" ($2.4 million). The focus of this program is to look at what is going to happen in the reactor (WASH-1400 looked at the risks to the public) using the techniques of event and fault tree analysis. By studying these "off-normal" transients in detail, with the aid of suitable computer codes, one may be able to interrupt event sequences in a way or manner that would reduce the consequences. Thus, a large fraction of the program is to identify those places in the event trees and fault trees where sequences can be interrupted. Dr. Theofanous pointed out that these studies should be done in thorough detail, and should not be the "input-output" type of study.

The fourth item is "improved reactor safety" ($2.2 million). This is loosely defined as that work which is not confirmatory. Included is the issue of the reliability of the decay heat removal system. There is no new work proposed but the supplemental funds would be used to accelerate ongoing work in this area.

The fifth item is "plant response under accident conditions." Research is interested in the reactor vessel's response to H_2 explosion inside the reactor. The Subcommittee remarked that it would be more interesting to find out if and how an explosion can occur inside the vessel. Oak Ridge National Laboratory has the facility to perform reactor shock experiments. Dr. Siess doubted if anything important would come out of this study, and suggested that Research should study the effects of H_2 explosion on vital features inside containment. Other work in this program include benchmarking structural and piping system analysis codes, two-phase mixture flow-through PORVs and hydrogen behavior in the reactor coolant and the containment. (The objective of the PORV work is not to develop valve reliability criteria but to study the physical behavior of the flow through the valve.)

The sixth and last item is "post mortem examination and plant recovery" ($2.7 million). While the EPRI-DOE-industrial community is thinking about recovery from the accident, NRC's main interest is to recover the information inside the reactor vessel and containment: information that can be used to benchmark a lot of computer codes, such as those that simulate fission product transport. DOE has agreed to provide resources to gather the data in the recovery operations, and EPRI is perparing a request for those resources. Dr. Lawroski suggested that Research should put part of its effort in finding better surfaces in the lower portions of the containment where fission product bearing water would accumulate. Mr. Budnitz commented that was a good point.

Dr. Okrent asked if the programs would be applicable to BWRs. Dr. Murely said that it is Resarch's desire to make the programs cover as wide an area as possible, and that the Subcommittee should give Research recommendations as to the scope of these programs. Dr. Mark asked why users' requests are not needed to initiate the six items of supplemental research. Mr. Levine said a second mechanism to start research programs is for him to write a Staff paper to the Commission (a draft of the paper was distributed in the meeting). The Commission's concurrence on the letter would satisfy the "user" requirement.

Dr. Mattson of NRR talked of the heavy NRR workload and a "verification and validation" concept which might help to solve the workload problem. This concept calls for contracting out portions of the review work to contractors such as national laboratories. However, the current NRR technical assistance of $15 million a year is barely sufficient to support current programs. NRR is developing a supplemental 1980 budget to be completed in a few weeks. He volunteered to have the "Lessons Learned" task force come to the June 14-16, 1979 full Committee meeting to provide a status report regarding where NRR stands on all the ACRS recommendations.

Dr. Murley proposed that the talk on H_2/O_2 generation, scheduled on the agenda, be omitted. Dr. Okrent accepted his proposal but the viewgraphs were handed out.

M. W. Kane

Mr. Kane said that bulletins were sent to all BWR licensees and the Staff is evaluating the responses. The Staff will meet with GE and the licensees to develop some short- and long-term actions. Some of the lessons learned can be applied to both PWRs and BWRs, these include operator qualification, health physics etc. There is, according to Dr. Mattson, no specific lessons that can be applied to BWRs. In the area of operator training, Dr. Mattson said that the TMI-2 operators are typical of other plants, but it is difficult to say if other operators would have done the same things. The Staff has not, in the past, instructed operators as to what they should not do; Mattson said this has to change. A Staff report for BWRs, comparable to the B&W feedwater transient report, will be published in July.

(The following paragraph is a brief record of a 20-minute closed session, 2:00 p.m. - 2:20 p.m.)

Mr. Novak reported that the Staff just received information that in 1974 an incident similar to TMI-2 occurred at a foreign reactor. Consequences were minimized since the operator isolated the stuck open PORV within 2-3 minutes.

(End of Closed Session)

Dr. R. Mattson

The Subcommittee decided not to ask the Staff to make presentation on the Generic Report on Feedwater Transients (NUREG-0560). Dr. Mattson stated that the original intent was to examine feedwater transients in B&W reactors. Later, as the understanding of the accident was improving, the work was broadened to include W and CE designs. Eventually, the Staff began to understand the importance of training and procedures, and concluded that their recommendations on these issues be applicable to all light water reactors. In light of that, the Lessons Learned Task Force (headed by Dr. Mattson) would issue an assemblage of short-term actions taking into account recommendations contained in NUREG-0560 and all of the ACRS letters. Thus, NUREG-0560 serves as a jumping-off point for the "Lessons Learned" work. The report's recommendations can be classified into four areas: (1) design deficiencies, (2) equipment malfunction, (3) human error, and (4) regulatory ommissions.

Following Dr. Mattson's brief introduction, the Subcommittee asked him to clarify or expand a number of points in the report. Important ones can be summarized as:

(1) Do not train operators with a simple design-basis, single-failure type of event tree. Complicate it and train operators to be effective agents for intervention in accidents.

(2) The Standard Review Plan requires all plants to have seismically qualified water source for the auxiliary feedwater system. Some older plants do not meet this requirement.

(3) Some licensees proposed to block off the PORV permanently but Mr. Denton recommended to the Commission not to require such.

(4) NRR is working on the requirement for close-in technical support. Operators trained on one unit of a multi-unit plant may not be able to function effectively in other units of the same plant. Close-in technical support is a better solution during accidents.

BABCOCK & WILCOX PRESENTATION (4:00 p.m. - 7:48 p.m.)

Mr. J. McMillan

Mr. J. McMillan indicated that B&W has prepared a full day-and-a-half presentation, in an attempt to answer the questions raised in the April meeting.

Mr. Taylor

Dr. Carbon extended an invitation to B&W to make a presentation to the full ACRS in the June meeting.

On March 28, 1979, B&W's major effort was directed to TMI. At about mid-April, its effort was split 50-50 between TMI and other customers, while today, its main effort is being spent on the latter in helping them to make design changes, re-train operators, etc. B&W is trying to develop a tighter coupling between the designer, analyst, trainer, and operator.

The B&W on-site representative was notified of the accident at about 6:00 a.m., and he arrived at the site at 7:00 a.m. By 9:00 a.m., B&W has formed a task force to be dispatched to Harrisburg. Only the site representative got into the plant, the rest of the people arrived after site emergency was declared and were not admitted until the next day. There was much communications problem either way between TMI and B&W, and Mr. Taylor did not comment on what the lone represenative was doing during the first day. It was until the second day, after the B&W team entered the site, did they start getting some hard data. B&W eventually had about 25 people at TMI and 220 people in Lynchburg dedicated to support TMI-2.

About two weeks after the accident, 170 operators have gone through special training at Lynchburg. At the site, however, neither B&W nor anybody else, was ever in full control.

Mr. N. Elliott

There are normally four sources of individuals who could become operators: high school graduates, U.S. nuclear navy enlisted men, U.S. nuclear navy officer, and engineering graduates from colleges. The NRC considers navy nuclear experience equivalent to large power reactor experience. A high school graduate would get some (not specified) training in thermodynamics and thermalhydraulics from the licnesee. B&W does not provide any basic training. Anybody with a high school diploma can apply for operator training but the basic training, and subsequent stages of training would eliminate some people (Mr. Elliott did not give any fail-out rate). The basic training course includes atomic theory, reactor theory, heat transfer, fluid flow, nuclear instruments, health physics, shielding, and radiation. Dr. Okrent wondered how all these can be taught in a period of 12 weeks.

For the navy officer, he can obtain a Senior Operator License in 6 months without ever having operated the reactor. Mr. Elliott assured the Subcommittee that the officer has had a lot of experience at the control panel.

Training on the simulator includes all DBAs described in SRP Chapter 15, but does not currently go beyond design-basis accidents.

(A videotype film on operator training for the TMI-2 event was shown).

Dr. Okrent said he thought there was a commitment made by B&W to make available to the Subcommittee reports describing the equations used in the simulator. Mr. Taylor said he could not remember such commitment, and that there are no such reports. The Subcommittee showed a general desire to see the equations used. Mr. Elliott said that most of these equations can be found in Gladstone, and that the B&W simulator meet the requirements in ANSI, ANS 3.5, 1979, "Nuclear Power Plant Simulators for Use in Operator Training." Dr. Kerr expressed doubt if the similator can predict new phenomena that it was not programmed to do. Mr. Michelson similarly doubted if it can keep computing into degraded conditions such as core melt. Mr. Elliott showed slides comparing P, T and water level plots produced by the simulator and actual data from TMI; even before modification of the computer program, the simulation of the transient was qualitatively accurate.

B&W does not provide simulator trainees with any written material since procedures from the trainees' plants are used.

Mr. R. Kosiba

Mr. Kosiba addressed the issue of how operating experience is processed and used. The Site Problem Reports are originated at the sites during construction and startup testing by B&W people. These SPRs document equipment failures, system abnormalities, difficulties in maintenance, etc. The next stage is evaluating and resolving the problem, and the last is the action stage, wherein the resolution is delivered to the site. Occasionally, a letter advising all customers is sent. Routine service bulletins are also sent advising on minor items.

Dr. Okrent asked if, out of this program of review process, B&W has arrived at any conclusions concerning the integrated control system. Mr. Kosiba said there was no question of safety, but B&W did note that it is necessary to tune up and adjust the ICS in the start-up phase.

Mr. C. England

Mr. England demonstrated a device, newly installed on the B&W simulator, which could take as input temperature and pressure, and read out how much margin there is to saturation.

The meeting was adjourned at 7:48 p.m., to be reconvened at 8:30 a.m. on the following day.

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Executive Session (8:30 a.m. - 8:35 a.m.)

Dr. Okrent called the meeting to order and stated that the Subcommittee should think of possible recommendations for the full ACRS. It was not likely that the Subcommittee can finish all the items on the agenda and there will be at least one more meeting in the future. Mr. Etherington would prepare a tentative list of topics that the Subcommittee may present to the full ACRS.

B&W Presentation (8:35 a.m. - 7:00 p.m.)

Mr. Taylor briefed the Subcommittee on the planned order and content of the B&W topics, and asked for guidance on the level of detail on the 2- to 3-hour presentation to be made to the full ACRS.

D. LaBelle

Immediately after the TMI accident, an analytical team was established under Mr. LaBelle with the objective to develop a factual event scenario and to demonstrate that B&W techniques can predict the profiles of the accident. He said that the comparison between the site data and the simulator-calculated system response is excellent. In reconstructing the TMI-2 accident, B&W used information from three different areas: 1) reactimeter (a data logger) and strip charts from the site, 2) NRC event sequence and 3) B&W analytical simulations.

Dr. Mark pointed out that the simulator does not "predict" the accident; it merely, if given the right input, reproduces the broad features. If the simulator could really predict the accident, the accident could have been stopped.

The CADDS code was used to simulate the first 6 minutes of the accident. This code can simulate a single loop with a series of control volumes and has been bench marked against other computer codes. It cannot, however, take into account two-phase flow. To model the steam generators, it has two options. It can either use a heat-demand model and thus becomes the NRC-approved CADD code, or it can use detailed steam generator model with up to 52 modes. The former option was used. The code was not modified to do the simulation. Mr. Michelson remarked that it was possible that there was some void formation in the reactor coolant system during the first 300 seconds, but CADD has not predicted any voids during that period.

Mr. LaBelle felt that based on the close match between actual and simulated data, B&W has provided good bench-marking. He said that the code can be effectively used in design and safety studies up until the onset of two-phase conditions. Dr. Theofanous asked if B&W would provide a document which will be sufficiently complete to allow one to conclude in the same way as B&W did. The answer provided by Mr. Taylor is "no."

Dr. Okrent said that by now each reactor vendor must have some incentive to improve his ability to simulate transients before their occurrence.

The CADD code is described in Topical Report 10098. Mr. Womack explained that the "heat demand" option simply means that instead of having a model for the secondary side, one inputs time-dependent heat removal rates. As such, feedwater water flow rates cannot be an input. B&W iterated on the heat demand to produce reasonable match-up, and then ran the steam generator model (option 2 of the code) to verify that the heat demand is is indeed what they have iterated. Dr. Theofanous expressed suspicion at this iterative technique to obtain good match.

Mr. B. Dunn

Simulation of the accident from 6 to about 130 minutes was done with the CRAFT-2 code, which is a LOCA-ECCS code capable of accounting for void formations. The primary system was treated as a homogeneous system, assuming effective stirring. Calculation shows that there were conditions of high void, as high as 67%. Mr. Michelson questioned the capability of the RCS pumps to maintain circulation under such conditions. Mr. Dunn said that the TMI pumps have been tested in steam and can function as blowers for steam. Dr. Theofanous challenged the assumption of homogeneous system, noting that the average void fraction in the coolant system was 50%. Mr. Dunn said that the homogeneous assumptions corresponded rather well to the measure flow. Even the pump (electric) current flow showed that the void passed through the pump, although it was not accurate enough to pinpoint when the void actually passed. Calculation showed that if auxiliary feedwater was available, with the RCS pump running and one high pressure injection train available, the void fraction would have been small and the system could return to solid water conditions in a short time.

Mr. Dunn concluded by saying that the B&W small break evaluations remain valid and the available tools are good, and that the safety instructions designed for proper recognition of a small break can prevent recurrence of the TMI-2 accident. Dr. Okrent said it was premature to draw such conclusions.

Dr. A. Womack

Dr. Womack discussed the event path of the loss-of-main-feedwater transient, if it occurred prior to April 21. B&W has taken actions to reduce the TMI sequence of events, and similar sequences. Some actions were taken in response to I&E Bulletin 70-05B. These actions are: changing PORV setpoint to 2450 psi (above the high-pressure reactor trip setpoint), reduce the latter to 2300 psig and require an anticipatory reactor trip on loss of feedwater and turbine trip. With these modifications, loss of FW would trip the reactor and there would be no pressure transient. Even if the anticipatory trip signal does not work (it is control-grade), the protection system would trip the reactor at 2300 psig, and a pressure pulse may open the PORV. He feels that B&W has greatly reduced the probability of PORV opening.

Dr. Okrent asked if water hammer is expected with delayed auxiliary feedwater activation. Dr. Womack said that he has not seen water hammer under these conditions and does not anticipate it. Early in the design of the Oconee plant, B&W has established some very specific criteria with regard to feed piping configuration. The NRC staff has evaluated these criteria and concluded there is no problem with water hammer because of those criteria.

Mr. B. Dunn

He discussed the small break management guidelines prepared by B&W. "Small break" is defined as those breaks which result in a series of quasi-steady states (no rapid alterations in system conditions), in which the injection system(s) plays catch-up with the coolant loss at the break for a period of time. In this type of analysis, the steam-water distribution is important.

For small break with loss of RC pumps, the reactor coolant will eventually come through natural circulation, to the "boiling pot mo[illegible]" when the primary cooling mechanism would be high pressure injection. The worst case for the Babcock 177 plant is found to be the 0.07 ft^2 break, where some core uncovery occurs. Even with this case, the system would eventually refill.

For small breaks with RC pumps available, the steam generators will be available for cooling. The primary pressure may level off at steam generator pressure, or, if the break is large enough, the pressure will continue decreasing. The system pressure eventually stabilizes at or below steam generator pressure, and the system is considered homogeneous.

B&W has written new operator guidelines for small breaks. ("Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plants," May 7, 1979) The general principle of these guidelines is that nothing other than the automatic system actions has to take place immediately. The operator should allow the system to stabilize, make his observations based on multiple instrument readings, and then do certain specific things. None of the further action described in the guidelines is harmful. The Subcommittee asked if automatic systems can stabilize the system, why bother to have the operator take any actions. Mr. Dunn said there is a difference between being "ok" and "preferred." Actions in the guidelines are preferred.

Mr. Michelson asked if single failures, including inadvertent bumping of the RC pump at the wrong time, have been taken into account in developing these guidelines. Mr. Taylor said "yes," but B&W did not consider the starting of a pump to be a failure of any consequence.

The guidelines were developed based on results of B&W's small-break analysis, other experienced engineers' knowledge of small breaks and input from senior operators.

Regarding RC pumps, the guidelines recommend leaving them running, even to the point when they are just blowing steam, and eventually burn out. Upon loss of all feedwater, the operator is to initiate high pressure injection until the system is sufficiently sub-cooled or the feedwater is resumed.

Mr. Etherington pointed out that B&W has not addressed valve relieving capacities. Mr. Taylor agreed to discuss this in the full ACRS meeting.

Mr. D. Roy

B&W has a continuing effort to gather data, such as that at TMI, so to bring its computer codes up to the state of the art. In the past few weeks, such effort has resulted in the set of guidelines described above. It is for sure that lessons will be learned from TMI, and B&W hopes that the ACRS and the industry would provide ideas as to where it can focus its effort and refine the ongoing process.

Dr. Okrent asked members of the Subcommittee to send suggestions and recommendations (on safety research, staff feedwater transient report and B&W presentations) to him, and copies to R. Fraley and R. Major.

Mr. R. Jones

He presented a much truncated version of his intended presentation due to shortage of time.

During the first few minutes of a small break accident with loss of the pump, B&W has calculated that the primary coolant would naturally circulate as a homogeneous mixture. Core uncovery would not happen for most breaks and be brief for some. Mr. Michelson commented that experiments would be needed to verify these claims.

For very small breaks, after the initial natural circulation phase, the primary system would function as a reflux boiler: water is turned to steam in the core and returned to the vessel as water via the steam generator where condensation occurs at some level inside the tubes. This level of condensation is quite constant for very small breaks but decreases with time with larger small breaks. Mr. Michelson did not agree completely with the scenario and on Dr. Mattson's suggestion, Dr. Okrent asked him to write down his opinion in the form of a few questions for the staff. (This request was withdrawn by Dr. Okrent later in the day as the difference in opinion was resolved.)

Mr. B. Karrasch

The once-through steam generator uses super-heated steam without using the drying technique required for a recirculating boiler. Steam temperature and throttle pressure are maintained constant over the entire load range by adjusting control rod position and turbine throttle valve position. The upper part of the steam generator is where super-heating takes place and the inventory in the shell side is not specifically controlled as a level. Thus "level" in a once-through steam generator is different from "level" in a recirculating counterpart. The integrated control system (ICS) is specifically designed in conjunction with the OTSG. The ICS will atuomatically adjust the turbine throttle valve positons and feedwater flow rate to maintain constant stream pressure. The ICS also automatically positions the control rods to adjust reactor power to maintain a constant temperature. It was designed with the objective ro redcue the reactor power in a controlled fashion for several moderate frequency abnormal events (turbine trip, partial or full load rejection, loss of a feedwater pump, loss of a RC pump). The ICS runs the reactor within a region allowed by the reactor trip setpoints prescribed by the reactor protection system.

The design basis of the pressurizer was discussed. The pressurizer is not designed to maintain pressure after reactor trips since its heating rate cannot accommodate rapid pressure drops.

Dr. Okrent said he would like to have B&W address any safety problems the ICS might have. (At this point, Dr. Okrent asked Mr. Etherington to recommend to B&W what to present to the full ACRS.)

Mr. Etherington

Recommendations for B&W regarding items of interest to be presented to full ACRS on June 14 and 15:

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- Mr. Taylor would make introductions and describe chronology of events. Should attempt to answer Dr. Carbon's question on the communication problem between Lynchburg and the B&W site representative.
- For operator training, describe only use of simulator for operator training.
- For response to NRC bulletins, keep to the specific responses and not background material.
- Skip the description of simulation of the TMI-2 accident.
- Describe actions taken to reduce the probability of future TMI-2 events.
- Describe the small break phenomenology and operating guidelines.
- Provide a very much shortened version of Dr. Karrasch's presentation.
- The staff will respond to Mr. Michelson's concern relating to the level in the upper candy cane area (see earlier under "Mr. Jones").

Dr. Okrent indicated that the total B&W presentation, including questions and answers, would last two hours.

(Dr. Okrent announced that he would interrupt Dr. Karrasch's presentation further to have Dr. Mattson summarize what the staff would have presented if there was enough time.)

Dr. Mattson

All the topics listed in the agenda for 2:00 p.m., June 1, are being considered by one or more experts in the Lessons Learned Task Force. A brief discussion on each of these topics, a thorough n (see agenda for these items), ensued:

Dr. Mattson asked the author of item l to explain his intent. This item was written by former ACRS member N. J. Palladino and maybe a few others. No direct answer was given but Dr. Lipinski indicated that the Germans have such a full pressure system which may shed some light on this item.

On item m, Dr. Mattson asked if the intent was protection against sabotage. This item arose from the fact that a TMI-2 operator event to the auxiliary building to try to get the condenser back to operation, found the valves locked out, and simply broke the locks. This incident can be looked at both ways: does one want locks that can break easily for emergency, or does one care more about security? The Subcommittee did not clarify this point. Dr. Mark, as Chairman of the Safeguard and Security Subcommittee, said that it was unlikely that the TMI-2 accident resulted from sabotage. The "Lessons Learned Task Force" would not divert its attention to sabotage protection, but would spend some time to evaluate the TMI-2 security in the sense of emergency preparedness or crisis management (people who needed to get into the site could not get in). Dr. Mark thereupon proposed to take up the subject of TMI-2 security in the next Safeguards and Security Subcommittee meeting.

On item a, Dr. Mattson asked if the Subcommittee wanted to have undue risk quantified, or to have quantitative criteria on some individual safety issues. Dr. Siess suggested that there should be two quantifications; one is a definition of risk in terms of man-rem, curies released etc., the other one is an acceptable level of probability for it. At present, decisions

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are made (such as the shutdown of five plants for fear of seismic events) without these quantitative criteria. Dr. Mark said that one Senate building would give an occupant 250 millirem per year, much higher than what one would get at the TMI-2 site boundary. He suggested that Mattson could collect some such real life data so that one can easily compare consequences of any future accidents with these data. Dr. Mattson committed to have his task force to develop some policy along Dr. Mark's comment.

Item b., the task force will look at it "in the context of does it do what it was intended to do? Is it sufficient?"

Item c., the task force will examine facilities both from the sense of separation criteria, and the sense that facilities from the other unit may help to cope with the accident.

Item f., standardization does inhibit changes in design. NRR is currently looking at possible policy implications in this regard.

Item g., suggested the possibility of introducing Class IIE for classifying equipment (now there is only Class IE and all others).

Item i., Dr. Mark said according to the interviews, people at TMI-2 have done their best based on whatever information they had. He pointed at the equipment that B&W brought (see minutes on May 31) to the meeting and stated that it is a very good example of improved assistance to the operator.

Dr. Kerr remarked that maybe there are just too many warning lights and alarms causing confusion.

General Discussion

Mr. Michelson said that he and B&W have resolved their differences on the issue of water/steam in the hot leg. He requested and Dr. Okrent agreed that there was no need for him to write questions for the NRC staff (see under "Mr. Jones").

Mr. Taylor invited the ACRS Consultants to discuss with his staff details of B&W's computer codes. Dr. Okrent declined the offer.

Dr. Karrasch (continued)

There are two elements which contribute to natural circulation in a B&W plant: water temperature gradient along the axis of the core, and the gradual increase in density inside the steam generator tubes. Thus if a certain level is maintained inside the tubes, natural circulation will proceed and continue. The higher the water level in the tube, the better it is.

Mr. Carlton

Early in the development of the ICS, B&W has aimed at desigining it to have the capability to accept load change at certain rates that utilities were interested in. The best way to keep from challenging the protection system or safety system is to keep the unit on line. B&W has designed the turbine to be the fastest reacting component, followed by the steam generator and then the reactor.

Since the TMI-2 accident, B&W has started a failure mode and effects analysis of the ICS. This analysis will concentrate on ICS failure modes that could affect the feedwater system, emergency feedwater system, pressurizer level and reactor coolant pressure. Plant data for tnis analysis is being collected and analyzed.

Dr. Kerr asked why there are so many feedwater loss events. Mr. Carlton answered that the feedwater system comprises the flash tanks, moisture separator drains, demineralizers, etc., a very complicated system. There are many ways that these could fail and affect the whole system. Dr. Kerr commented that because of the large number of failures, more effort should be spent on improving its reliability. A reliable system is more likely to be safe.

Dr. Okrent said he was interested in knowing what the unique features of the ICS are, and what kind of things can happen during transients. Mr. Ray pointed out that loss of the low voltage power supply to the control system will result in loss of automatic operation of the ICS as well as the ability to manually override it. Mr. Carlton agreed that more work should be done in the area of power availability. Dr. Okrent stated that in the ICS/FMEA, B&W should see if there are combination of ways in which the system could fail and lead to some awkard situations, and if B&W just sticks to the single-failure approach, these situations may not show up. Mr. Ray added that electromagnetic induction caused by short circuit could induce into the ICS circuits abnormal conditions.

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Dr. D. Roy

He summarized B&W responses to ACRS recommendations. In the area of accident prevention, a lot has been said in the past 15 hours of presentation. The responses are summarized as:

1) Additional analysis of transients or small break accidents - reports in response to this was submitted, titled "Evaluation of Transient Behavior and Samll Reactor Coolant System Breaks in the 177 Fuel Assembly Plant." B&W will continue benchmarking the TMI-2 accident with its computer codes.

2) Natural circulation analysis, criteria - report submitted (see above item). New operating guidelines submitted to customers on controlled transition to natural circulation.

3) Safety system status monitoring - would recommend to customers means for display of status of key systems.

4) Instrumentation to diagnose and follow accident - have recommended use of wide-range T_h indication, and use of T_{sat} - P_{sat} indicator (demonstrated at end of yesterday's meeting). Will continue to investigate use of reactor water level instrumentation.

5) Operator training - supplementary training offered. Currently developing emergency procedure guideline.

6) Expanded safety research - no action has been taken but would support TMI-2 recovery. B&W R&D plan is being reviewed to assess impact of TMI-2.

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7) RCS venting - no action has been taken but is still being investigated.

At the end of Dr. Roy's presentation, Dr. Okrent adjourned the meeting at 7:01 p.m.

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Duvall as Presiding Officer in this investigation.

The Secretary shall serve a copy of this Order upon all parties of record and shall publish it in the Federal Register.

Issued: May 10, 1979.

Donald K. Duvall,

Chief Administrative Law Judge.

[Investigation No. 337-TA-60]

[FR Doc. 79-15271 Filed 5-15-79; 8:45 am]

BILLING CODE 7020-02-M

NATIONAL FOUNDATION ON THE ARTS AND THE HUMANITIES

Theatre Advisory Panel Meeting

Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92-463), as amended, notice is hereby given that a meeting of the Threatre Advisory Panel to the National Council on the Arts, will be held on June 2, 1979 from 9:00 a.m.—5:30 p.m. in Room 1422 of the Columbia Plaza Office Building, 2401 E Street, N.W., Washington, D.C. 20506.

A portion of this meeting will be open to the public on June 2, 1979, from 9:00 a.m.—10:00 a.m. for Questions and Answers pertaining to the Theatre for Youth Category and for a policy discussion.

The remaining sessions of this meeting on June 2, 1979, from 10:00 a.m.—5:30 p.m. are for the purpose of Panel review, discussion, evaluation, and recommendation on applications for financial assistance under the National Foundation on the Arts and the Humanities Act of 1965, as amended, including discussion of information given in confidence to the agency by grant applicants. In accordance with the determination of the Chairman published in the Federal Register March 17, 1977, these sessions will be closed to the public pursuant to subsections (c)(4), (6) and 9(b) of section 552b of Title 5, United States Code.

Further information with reference to this meeting can be obtained from Mr. John H. Clark, Advisory Committee Management Officer, National Endowment for the Arts, Washington, D.C. 20506, or call (202) 634-6070.

John H. Clark,

Director, Office of Council and Panel Operations, National Endowment for the Arts.

May 8, 1979.

[FR Doc. 79-15220 Filed 5-15-79; 8:45 am]

BILLING CODE 7537-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Ad Hoc Subcommittee on Implications of Three Mile Island, Unit 2 Accident; Meeting

The ACRS Ad Hoc Subcommittee on Implications of the Three Mile Island, Unit 2 Accident will hold a meeting on May 31 and June 1, 1979 in Room 1046, 1717 H St. N.W., Washington, D.C. 20555.

In accordance with the procedures outlined in the Federal Register on October 4, 1978 (43 FR 45926), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: *Thursday and Friday, May 31 and June 1, 1979, 8:30 a.m. until the conclusion of business each day.*

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

At the conclusion of the Executive Session, the Subcommittee will discuss with representatives of the NRC Staff, the nuclear industry, various utilities, and their consultants, state and local officials, and other interested persons, the implications of the Three Mile Island, Unit 2 Accident.

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

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the Designated Federal Employee for this meeting, Mr. Richard K. Major (telephone 202/634-1414), between 8:15 a.m. and 5:00 p.m., EDT.

Background information concerning this nuclear station 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 Streets, Harrisburg, PA 17126.

Dated: May 10, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-15122 Filed 5-15-79; 8:45 am]

BILLING CODE 7590-01-M

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

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 24 and 48 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments for BSEP, Units 1 and 2 provide Technical Specifications for the protective instrumentation associated with the Anticipated Transients Without Scram Recirculation Pump Trip. These specifications were inadvertently omitted when Amendment No. 12 to DPR-71 and Amendment No. 39 to DPR-62 were issued on November 23, 1977.

The amendment for BSEP Unit 2 also changes the Technical Specifications to establish revised safety and operating limits for operation in Cycle 3 with 7x7, 8x8, and 8x8R fuel.

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

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

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ATTACHMENT 2

Meeting Attendees

ACRS

D. Okrent	
M. Carbon	
H. Etherington	
W. Kerr	
S. Lawroski	
C. Mark	
J. Ray	
C. Siess	
I. Catton,	Consultant
W. Lipinski	"
C. Michelson	"
T. Theofanous	"
P. Tam	ACRS Staff
R. Major	Designated Federal Employee
J. Griesmeyer	ACRS Fellow

NRC Staff

R. Budnitz
R. Capra
J. Conran
T. Speis
T. Cox
D. Kirkpatrick
A. Marchese
R. Mattson
L. Tong
S. Levine
T. Murley
T. Richardson
T. Heltemes
T. Long
M. Taylor
C. Kelber
T. Larkins
G. Edison
W. Johnston
W. Kane
D. Hoatson
R. Sherry

B&W

D. LaBelle
J. McMillan
E. Womack
E. Kane
T. Taylor
D. Roy
R. Davis
R. Winks
T. Carlton
K. Suhrke
B. Dunn
B. Karrasch
R. Jones
R. Kosiba
N. Elliott
C. England

Public

T. Tyler, TVA
R. Leyse, EPRI
S. Lewis, Duke Power
A. Friedman, United Engineers
G. Overbeck, " "
A. Hosler, WPPSS
E. Wallace, GPUSC
N. Salerno, Consumers Power Co.
M. Weinstein, American Nuclear Insurers
W. House, Bechtel
B. Washburn, DOE
K. Trickett, DOE
W. Stewart, Florida Power Corp.
T. Brandt, Houston Light & Power
J. Wroblewski, PSE&G
W. Williams, S. C. Public Service Authority
S. Quennoz, Toledo Edison
D. Call, Westinghouse
R. Bruce, Offshore Power Systems
D. Mardis, Arkansas Power & Light
N. Shirley, GE
C. Brinkman, GE
L. Davis, Stang & Svendson Law Firm
R. Schaffstall, KMC Inc.
S. Sarisohn, AWEC
K. Ota, Kansai Power
T. Martin, WTECH
E. Losen, American Public Power Assoc.
J. Coombe, Stone & Webster
H. Hamada, Tokyo Electric Power
P. Smith, Newhouse News Service
R. Hollis, Public Service Commission of Md.
H. Yocom, Gilbert Associates
O. Dixon, S. C. Electric & Gas
W. Smith, Bechtel
T. Myers, Toledo Edison
B. Whitlock, Ace-Federal
T. Barham, Ace-Federal
S. Sarisohn, American Nuclear Energy Council
T. Martin, NUTECH

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Attachment 3

ACRS SUBCOMMITTEE
ON
THREE MILE ISLAND-2 ACCIDENT IMPLICATIONS
May 31 - June 1, 1979

8:30 A.M. DISCUSSION WITH NRC STAFF

. Introduction

(1) Discussion of Staff proposal for augmented research and technical assistance (STAFF: NRR, RES, EDO)

(2) Research on TMI during recovery process (update of previous discussions) - (STAFF)

(3) H_2/O_2 - generation under Accident Conditions (STAFF: NRR & RES)

***** BREAK *****

(4) Preliminary discussion on implications of TMI-2 for a Boiling Water Reactor (STAFF)

(5) Generic Report on Feedwater Transients (NUREG-0560) - Other Questions Arising from Specific design considerations for various B&W plants:

Examples:

- Seismic qualifications of auxiliary feedwater piping and water supply
- Auxiliary feedwater piping arrangement at Oconee
- Others

1:00 P. M. **** LUNCH *****

2:00 P. M. Discussions with B&W

. Introduction

(1) Chronology of Events - When was B&W first notified of incident? When did B&W start to give technical assistance to Met. Ed.? What assistance during 1st 24 hours? Other customers? Response team formed?

(2) Response to I&E Bulletins 79-05, 79-05A, 79-05B

(3) Proposal for Adequate Interim Operator Training and Adequate Instrumentation

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5:00 P. M. (4) Simulation Capability

(5) How is Operating Experience processed/used? (B&W)

7:00 P.M. ADJOURN

JUNE 1, 1979

8:30 A.M. RESULTS OF STUDIES: (B&W)

1. Specific studies on TMI-2
2. Small Break Studies
3. Hardware changes
4. Guidelines for operators
5. Natural Circulation Cooling (to include a discussion of transition phase i.e., from a water solid natural circulation mode to a boiling/condensing mode of natural circulation cooling.)
6. Feedwater Transients
7. Combined Transients
8. Others

(Copies of draft or final studies on the above topics would aid the Subcommittee)

10:30 A.M. ***** BREAK *****

9. (NUREG: 0560. Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company by NRR)

10. Effects of Loop Seals and OTSG on Reactor Behavior

11. Results of Failure Mode and Effects Analysis on I.C.S. (B&W)

12. Summary (Highlights responses to ACRS reports.)

1:00 P.M. ***** LUNCH ****

2:00 P.M. Other Topics as Time Permits - (STAFF)

a. Frequency of challenges of safety system
b. Single failure criterion
c. Use of shared facilities
d. Other design features to reduce the probability of accidents (e.g., plant simplification, improved materials, separation of control and safety systems)
e. Other design features to reduce the consequence of accidents (e.g., improved separation of barriers)

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f. Do standardized plants inhibit safety improvements?

g. List of equipment which must operate

h. Use of probabilistic vs. deterministic criteria/ evaluation (consider multiple human errors, etc.)

i. Use of computer assist to operator

short term - ?
long term monitor status/positon of important comporents/valves

j. Added emphasis on small/very small breaks

k. Research related to path of molten core release

l. Design all ECC systems to take full primary system pressure so that pressure interlocks are not needed on valves and operator doesn't need to take so many actions depending on his estimate of the progress of an accident

m. Effectiveness of valve locks (at TMI-2 the operators broke the locks)

h. Provide more RHR loops

4:45 P.M. Discuss Future Subcommittee Actions

5:00 P. M. ADJOURN

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