UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

SUPPLEMENT #4 TO PETITION FOR IMMEDIATE ACTION TO RELIEVE UNDUE RISK POSED BY NUCLEAR POWER PLANTS DESIGNED BY THE BABCOCK & WILCOX COMPANY

UNION OF CONCERNED SCIENTISTS REPLY TO RESPONSES FROM THE NRC STAFF AND B&W OWNERS GROUP

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INTRODUCTION

The purposes of this Supplement #4 to the above-captioned Petition are to reply to the responses from the NRC staff and the B&W Owners Group and to supplement the list of petitioners.

On February 10, 1987, the Union of Concerned Scientists (UCS) filed the above-captioned Petition on behalf of itself and numerous citizens, elected officials and groups seeking the agency's immediate attention to the long-standing safety problems plaguing B&W plants. Additional petitioners were added by Supplements #1, #2, and #3, dated February 10, 1987, February 20, 1987, and March 23, 1987. Additional petitioners are identified beginning on page 63 of this Supplement.

The Petition is based primarily upon A) the history of accidents and near-misses at B&W plants, particularly those which have occurred since the TMI-2 accident despite the implementation of "fixes" intended to prevent recurrence of such accidents, and B) NRC documents spanning a decade identifying the unique safety hazards inherent in the B&W design. UCS demonstrated that the B&W plants pose undue risk to public health and safety, that they are more dangerous than other pressurized water reactor designs, that the NRC's action to date in addressing that risk has been dilatory and ineffective, that NRC lacks the analytical tools and resources to understand the behavior of B&W plants during accidents, and that there is nothing being done in the reasonably foreseeable future to correct the fundamental safety problems in B&W plants.

The Petition requests the NRC to suspend the operating licenses of eight (8) B&W plants and the construction permits of two (2) B&W plants. The Petition also requests that, prior to reinstating those licenses and construction permits, NRC complete its ongoing safety "reassessment" program, identify the specific actions necessary to correct the B&W safety problems, hold adjudicatory public hearings to determine whether those actions are sufficient, and require full implementation of the needed safety improvements.

On March 6, 1987, the B&W Owners Group filed "Initial Response of B&W Owners Group to Petition Filed Under 10 C.F.R. 2.206 by the Union of Concerned Scientists," hereinafter "B&WOG Initial Response." In a letter to UCS dated March 13, 1987, the NRC Director of Nuclear Reactor Regulation provided the staff's initial response to the Petition. H. R. Denton, Director, Office of Nuclear Reactor Regulation, to E. R. Weiss, General Counsel, and R. D. Pollard, Nuclear Safety Engineer, UCS, March 13, 1987, hereinafter, "NRC Staff Response." Mr. Denton concluded that the "concerns" expressed in the Petition "do not warrant immediate suspension of the operating licenses and construction permits [for B&W plants]," and stated that the NRC staff will continue to review this Petition and "a formal decision" will be issued "in the reasonably near future." NRC Staff Response, pp. 1, 3. On April 6, 1987, the B&W Owners Group filed "Principal Response of B&W Owners Group to Petition Filed Under 10 C.F.R. 2.206 by the

Union of Concerned Scientists," hereinafter, "B&WOG Principal Response."

Based on a review of the B&WOG Initial Response, the B&WOG Principal Response and the NRC Staff Response, UCS concludes that no new information has been provided which would justify changing our original conclusion that the relief requested in the Petition is necessary to protect the health and safety of the public. As discussed in more detail later, neither the B&W Owners Group nor the NRC staff has confronted, much less refuted, the NRC staff's finding that, despite the modifications made to B&W plants as a result of the TMI-2 accident, "the number and complexity of events has not decreased as expected." V. Stello, Acting Executive Director for Operations, NRC, to Hal Tucker, Chairman, B&W Owners Group, January 24, 1986. In other words, neither the B&WOG nor the NRC staff have faced the reality that the post-TMI "lessons learned" requirements have not been effective, as evidenced by the continuing occurrence of complex accidents at B&W plants.

Nor has either the B&W Owners Group or the NRC staff confronted, much less refuted, the fact that it was the continued occurrence of accidents at B&W plants which led the NRC staff to conclude that there was a need to "reassess the overall safety of B&W plants" and to "determine whether the present set of [NRC] requirements . . . lead to a level of safety at B&W plants that is comparable to other pressurized water reactors." Id. In other words, the actual operating experience at B&W plants after the TMI-2 modifications demonstrates that neither the modified design of B&W plants nor the set of NRC requirements applicable to that unique design is adequate to provide reasonable assurance that the plants do not pose undue risk to public health and safety.

Neither the NRC staff nor the B&W owners group have provided a reasoned basis for the NRC staff's January, 1986, statement that "[w]hile we believe that this reassessment is needed, we also believe that B&W reactors can safely continue to operate in the interim." Id. Instead, they act as if that sentence, by itself, disposes of the issue. On the one hand, the staff observes that, despite the modifications made as a consequence of the TMI-2 accident, the number and complexity of accidents has not decreased as expected. Since complex accidents continue to occur at B&W plants, the staff concludes that there is a need to reassess both the unique B&W design and the basic requirements applicable to B&W reactors. On the other hand, the staff claims that the plants are safe enough to operate even though recurring complex accidents have demonstrated that the bases for allowing the B&W plants to resume operation after the TMI-2 accident, i.e., implementation of the TMI-2 lessons learned requirements, were invalid. While we understand that the recognition of a safety question does not automatically call for suspension of operation, the law requires the agency to articulate a rational basis for its position that the B&W plants do not pose undue risk.

The Petition demonstrates that the risks to the public posed by B&W plants are immediate and paragraphs 80 to 90 of the Petition specifically address the fact that NRC has provided no reasoned basis for permitting them to operate for an unlimited (and always lengthening) "interim" period while reassessments of the B&W design and the applicable requirements are conducted. The B&W Owners Group and the NRC staff responses to the Petition provide no reasoned basis for continued operation. Indeed, the staff simply repeats previous NRC statements that the "events" which led to the safety reassessment "caused no offsite consequences." The NRC staff apparently was unable to address, and therefore chose to ignore, the observation (Petition, paragraph 88) that basing continued operation of the B&W plants on the absence of "offsite consequences" is inconsistent with NRC precedent and with any reasonable interpretation of the Atomic Energy Act's mandate to prevent undue risk to the public. The Commission's responsibility is to prevent undue risk to public health and safety, not wait until there are offsite consequences before taking action. For example, in 1979, NRC suspended the operation of several plants because new information indicated that the analyses done for resistance to earthquakes might be faulty. See, e.g., Power Authority of the State of New York (James A. Fitzpatrick Nuclear Power Plant), Order to Show Cause, March 13, 1979. Clearly the occurrence of an accident with offsite consequences is not the only way the NRC may learn that its previous basis for finding a plant safe were incomplete or nonconservative.

Suspension of a plant's license is called for when its safety is not assured. In the case of the B&W-designed plants, the original bases for finding "reasonable assurance" of safety were the assumptions and analyses made at the time of their licensing. These analyses were discredited by the TMI-2 accident, which showed them to be erroneous and nonconservative. After the TMI-2 accident, NRC issued an Order to each B&W plant suspending its operating license. In the Orders, the Commission found that:

> . . . B&W-designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), Order, May 7, 1979, p. 2.

The TMI-2 accident and subsequent investigations showed that the original licensing analyses for B&W plants were inadequate to provide a basis for a safety finding, as the 1979 shutdown orders clearly establish.

The B&W plants were permitted to restart only after a set of "short-term" changes were made and a commitment made to complete a longer list of "long-term" modifications. Thus, the NRC was able to reinstate its "reasonable assurance" finding only after certain changes intended to correct the B&W safety problems were made and "reasonable progress" had been made toward completing other changes "as soon as practicable." Id.

In the eight years since the TMI-2 accident, despite the fact that these "short-term" and "long-term" modifications were declared accomplished, the same fundamental safety problems that caused the TMI-2 accident have continued to cause complex accidents in other B&W plants. The NRC staff's January 1986, letter to the B&W Owners Group announcing the current "reassessment " is explicitly based upon this recognition:

> While we recognize that utilities are now and have been making modifications to their plants, the number and complexity of events has not decreased as expected.

V. Stello, Acting Executive Director for Operations, NRC, to H. Tucker, Chairman, Babcock & Wilcox Owners Group, January 24, 1986, p. 1.

Thus, actual operating experience at B&W plants has discredited both the original pre-TMI accident basis for finding the plants safe and the post-TMI accident basis for reinstating the finding. This is the fundamental point which the NRC staff and the B&WOG seek to obfuscate: even assuming that the original licensing confers a "presumption of safety" upon a licensed plant, the presumption cannot stand after the assumptions and analyses which produced it have been shown to be in error.

Aside from the legalistic arguments based upon the purported presumption of safety, the NRC staff and the B&WOG offer two new reasons to support their claims that the plants are safe enough to operate. First, the NRC staff claims that reactor trips have been reduced by 10% over the past five years. Second, the NRC staff and the B&WOG cite many changes, some which they say have been made and some which may be made in the future, to make the plants safer. NRC Staff Response, p.2.

With regard to the 10% reduction in reactor scrams, that is clearly not a significant improvement in safety. In fact, the reduction in scram frequency occurred during exactly the same time period as the recurring accidents and near-misses at B&W plants which led to the current safety reassessment. Thus, whatever safety benefit may be associated with this marginal reduction in reactor trip frequency, it has obviously not been effective in decreasing the number and complexity of accidents at B&W plants.

With regard to the changes which have been or may be implemented as a result of the B&WOG's Safety and Performance Improvement Program, no analyses are provided to relate these modifications to the safety problems which continue to cause complex accidents in B&W plants nor are any attempts made to assess the magnitude of the alleged safety improvement. Furthermore, until a recommended change has been implemented, it provides no safety improvement and cannot be cited as a basis for claiming that continued operation is justified.

In summary, the NRC staff was correct when it concluded that the operating history of B&W-designed plants demonstrates that there is a need to reassess the overall safety of B&W plants and to reexamine the basic design requirements for B&W reactors. We turn now to addressing the B&WOG responses and the NRC Staff Response in detail and demonstrate that no reasoned basis has been provided to justify the claim that B&W plants are safe enough to remain in operation in the "interim" while their "long term" safety is reassessed.

DETAILED REPLY TO NRC STAFF RESPONSE AND B&WOG RESPONSES

The arguments advanced by the NRC staff and the B&WOG fall into the following general categories:

- UCS has not met the applicable legal standard.
- Many changes were made in B&W plants after the TMI-2 accident.
- Recent operating history, and in particular the 1985 accidents at Davis-Besse and Rancho Seco, have limited generic implications.
- Changes have been recommended and "some" have been made to B&W plants since the safety reassessment program began.
- The original safety analyses made at the time the plants were licensed were conservative and thus provide a basis for continued operation.

In addition, the B&WOG puts forth an additional argument:

- The B&W design is not less safe than other plants.

We will address these arguments *seriatim*.

Legal Standard

The NRC staff does not articulate a standard by which this Petition should be judged. Nor does NRC case law provide a coherent or inclusive set of criteria for use in such cases. The words which are used in prior cases -- that a petition must present a "substantial" or "serious safety or environmental issue" -- do little more than state the ultimate safety conclusion without illuminating what constitutes such an issue.

Both the B&WOG and the NRC point to the fact that neither at Davis-Besse nor Rancho Seco, the most recent serious B&W accidents, were there offsite consequences. However, they surely would not argue that it is necessary that an accident with offsite consequences occur before the NRC can require immediate shut down and correction of substantial safety problems. As noted above, NRC required immediate shutdown of five plants when it discovered that an erroneous mathematical calculation was used in determining earthquake loads. Power Authority of the State of New York (James A. Fitzpatrick Nuclear Power Plant), Docket No. 50-333 (1979); Maine Yankee Atomic Power Co. (Maine Yankee Atomic Power Station), Docket No. 50-309 (1979); Virginia Electric and Power Co. (Surry Power Station, Units 1 & 2), Docket Nos. 50-280, 50-281 (1979); Duquesne Light Company, et al. (Beaver-Valley Power Station, Unit 1), Docket No. 50-334 (1979); See also U.S. NRC Press Release No. 79-52, "NRC Staff Orders Five Nuclear Power Plants Shut Down to Resolve Piping Questions," March 13, 1979 (reporting that plants were ordered shut down until a determination was made regarding the need for modifications in safety-related piping systems).

No earthquake had occurred (indeed, earthquakes of a magnitude sufficient to be concerned about are far less probable events than transients at B&W plants) and the NRC did not know that the as-built plants might not, in fact, be able to withstand the design basis earthquake. What caused NRC to order the shutdowns was information showing that the agency's basis for judging the plants adequate had been faulty. NRC could no longer have the requisite confidence in its earlier conclusion that operation posed no undue risk.

The evidence and arguments presented in the UCS Petition and discussed further below establish a far greater level of uncertainty, and a more palpable safety risk than was present in the case of the seismic calculation shutdowns. It may be that the NRC staff has become so enured to this uncertainty over the years that it no longer appreciates that "reasonable assurance of safety" requires a continuing rational basis for confidence in the correctness of that judgment. See Petition for Emergency and Remedial Action, CLI-78-6, 7 NRC 400; CLI-80-21, 11 NRC 707. The current lack of a rational basis for claiming that the B&W plants are safe enough to remain in operation is quite evident in the NRC staff's response to date to this Petition.

Nor is "new information" required as a condition for granting a show cause petition. See Consolidated Edison Co. of New York (Indian Point Station), DD-80-5, 11 NRC 351 (1980), granting a petition to issue a show-cause order revoking the license for Indian Point Unit 1. In the 1980 decision on UCS's Petition for Emergency and Remedial Action, CLI-80-26, 11 NRC 707, the Commission set a firm deadline for compliance by all operating plants with specific new stricter criteria for environmental qualification of safety equipment. This was done even though the NRC staff was well aware of the information presented and illustrates that unreasonable delay in resolving uncertainty concerning safety is sufficient to justify relief. In the case of the B&W plants, the delay in resolving known safety problems is much more aggravated. The fact that a safety problem is known to NRC cannot serve to defeat a petition unless the action being taken to address the problem is adequate to provide reasonable assurance of safety both in the long and short-term (i.e., now). The UCS Petition demonstrates that, for the B&W plants, this is not so.

Finally, much of the key data and perspectives presented by UCS are "new" in that they are based on B&W operating experience since implementation of the post-TMI "fixes" and analyses based thereon. For that reason, the B&WOG's repeated claims that these issues were resolved in the TMI-1 Restart and Rancho Seco hearings held after the TMI-2 accident miss the point of the Petition. The NRC may have once believed that the post-TMI fixes were adequate, but its own 1986 pronouncements make clear that it no longer believes so and the objective evidence establishes that they were not adequate.

In sum, the standard for judging this Petition does not require that a petitioner establish a certainty that a life-threatening accident has happened or will happen, nor that a petitioner bring forth some information not previously known to NRC. UCS has shown that there is not a rational basis for concluding that B&W plants currently provide reasonable assurance of the protection of public health and safety and that there is no realistic reason to expect such a basis in the reasonably foreseeable future. Under these circumstances, the relief requested by UCS is fully justified.

<u>The Post-TMI Fixes Did Not Succeed In Ameliorating The Fundamental Design-Related Safety Problems At B&W Plants</u>.

The Petition and its Appendix A contain a detailed exposition of the historical record regarding B&W plants. The purpose of this is to demonstrate that all efforts to date to address the shortcomings of the B&W design, both before and after the TMI-2 accident, have demonstrably failed. As shown in the Petition, the same basic design elements persist in both causing accidents and hindering accident mitigation. Indeed, it was precisely the failure of the post-TMI fixes to reduce either the number or complexity of accidents at B&W plants that caused the NRC to undertake a safety reassessment directed at reexamining the overall safety of B&W plants and determining what basic design criteria should be applied to B&W plants. V. Stello to H. Tucker, supra.

Under these circumstances, the NRC and B&WOG's references to the post-TMI-2 fixes is rhetorical obfuscation. Neither has directly addressed, much less disproved the key finding made by NRC Executive Director Stello in January, 1986, that despite the post-TMI fixes, "the number and complexity of events has not decreased as expected." Id.

Before the TMI-2 accident, NRC had relegated unresolved safety problems to the category of generic Unresolved Safety Issues in order to avoid the delay in licensing B&W plants which would have resulted if resolution were required prior to licensing. After the TMI-2 accident, the NRC established a category of "long term" requirements and permitted the B&W plants to resume operation on the basis that those requirements would be implemented later. However, the basic B&W safety issues were never properly addressed, instead becoming topics for endless study in the "generic issues" program or becoming lost in the cracks between various unresolved safety issue definitions.

Predictably, the same fundamental safety problems continue to cause complex accidents in the B&W plants and the NRC continues to allow the plants to operate, while postponing indefinitely effective action to correct the safety problems.

The NRC staff claims that the continuing accidents at B&W plants have "reinforced [its] concerns about these designs" and says that a safety reassessment program is needed, but turns responsibility for that program over to the B&WOG and asserts that B&W reactors can safely continue to operate in the "interim."

When confronted with the Petition, one argument that the NRC staff uses in response is that changes made after the TMI-2 accident made the B&W safe enough to operate. As discussed above, this argument is refuted by staff's own finding that "the number and complexity of events [in B&W plants] has not decreased as expected." Furthermore, numerous examples in NRC's own reports demonstrate that in some cases the changes were never made and in other cases they were ineffective. We give here three illustrative examples concerning three of the most troublesome B&W systems. The first involves the Integrated Control System (ICS), the second involves the pilot operated relief valve (PORV), and the third involves the emergency feedwater system.

Example 1: ICS

The first example demonstrates the manner in which the NRC has permitted the frustration and evasion of its post-TMI Orders purporting to address the known safety hazards posed by the nonsafety-grade Integrated Control System. There was widespread agreement after the TMI-2 accident that something needed to be done to resolve the ICS-related problems, including primarily the inability to predict the ways in which the system can fail and the effects of the many potential failures on the plants' safety. The ICS was addressed in Commission Orders, by the NRC's TMI-2 Lessons Learned Task Force, in NRC Inspection & Enforcement Bulletin 79-27, and in NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors."

In May 1979, the Commissioners ordered the B&W plants to submit a systematic analysis of the ways the ICS can fail and the effects such failures can have on plant behavior. See Petition, para. 51. When this analysis (BAW-1564) was received from B&W, it was pronounced inadequate by both the NRC staff and its contractor. See Petition, para. 54.

In July, 1979, the TMI-2 Lessons Learned Task Force Report was issued, containing a chapter entitled "Future Work by the Lessons Learned Task Force." Describing the short-term fixes as "narrow in scope," it committed the agency to a program of consideration of the "broader and more fundamental questions." NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July, 1979, p. 16.

Describing the nature of these fundamental questions, NRC further stated:

> The accident at TMI-2 demonstrated disparities between the description and evaluation of accidents in the licensing review of a Safety Analysis Report and the actual response of the plant and its operators. Events occurred that were not foreseen, analyzed, or prepared for. The differences between the actual sequence and those previously analyzed is due in part to regulatory requirements and guidelines. The system design requirements that are an important part of current regulatory requirements should, therefore, be examined in more detail for their adequacy. The system design requirements that are judged to be the most important and have been selected for further study by the Task Force are (a) the single failure criterion, (b) the division between safety grade and nonsafety-grade requirements, (c) operator interactions, and (d) post accident design requirements.

Id., at 17.

An October, 1979, report by NRC staff investigators dealing with the implications of the TMI-2 accident was more explicit.

The investigators concluded that one "problem" was that "the Integrated Control System (ICS) is not Class IE safety-related." They recommended for "near-term implementation" that the NRC establish the requirements for a safety-related ICS, evaluate the safety of continued operation of B&W plants without a safety-related ICS, and consider extra personnel and procedures to provide ICS backup during loss of power and/or loss of ICS situations. R.D. Martin, memorandum to J. M. Allan. "Operations Team Recommendations," October 10, 1979, p. 1, 18, 19.

In November, 1979, a loss of power to the ICS occurred at Oconee Unit 3, a B&W-designed plant. As a result, NRC issued Bulletin 79-27, with a requirement for response within 90 days, describing a number of actions to be carried out by licensees.

In May, 1980, NRC issued NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors," which recommended, <u>inter alia</u>, improvements in the reliability of B&W plant control systems, "particularly with regard to . . . the integrated control system itself." Eight (8) specific recommendations were made, including the following:

> <u>Prompt</u> followup actions should be taken on the recommendations contained in BAW-1564 (Integrated Control System Reliability Analysis).
>
> <u>Prompt</u> followup actions should be taken on IE Bulletin 79-27.

NUREG-0667, pp. 5-62, 5-63, 5-64, emphasis added.

To recapitulate, in 1979 and 1980, NRC at least three times specifically ordered action to be taken to address the unreliability of the ICS. However, when the Rancho Seco accident occurred in 1985 and the NRC established a special Incident Investigation Team to determine its root causes, the facts found by this team show that <u>none</u> of the 1979 and 1980 orders were met.

First, the ICS failure modes and effects analysis ordered in May, 1979, and done "generically" by B&W, "did not appear to meet the requirements of the original [Commission] Order." NUREG-1195, "Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985," p. 10-4. Moreover, although even the limited B&W analysis had "noted a number of changes that appeared to be warranted in the ICS, SMUD [the owners of Rancho Seco] concluded that no changes were necessary." Id. The NRC had never followed up by directing any of the specific recommended changes to be made. Instead, in an echo of its current stand, the NRC claimed that the plant was safe enough to operate and allowed the so-called "long-term issues" to be "considered in Unresolved Safety Issue (USI) A-47, 'Safety Implications of Control Systems.'" Id., emphasis added.

The Incident Investigation Team further found that the November, 1979, Bulletin 79-27 had also never been met.

> "Although the Bulletin raised significant concerns about the consequences of a loss of power to instrumentation and control systems, SMUD concluded that no additional design modifications were necessary . . . ."
>
> The NRC staff's initial review concluded that "SMUD's response did not contain sufficient information and did not adequately address the concerns in the Bulletin."
>
> After narrowing the scope of its review, the NRC staff "decided that the SMUD response was adequate" and concluded that "the long-term implications of Bulletin 19-27 would be addressed as part of USI A-47."

Id., pp. 10-4, 10-5, emphasis added.

Finally, the actions called for in 1980 by NUREG-0667 were likewise never done. The Rancho Seco 1985 Incident Investigation Team found:

> "it does not appear that these recommendations [in NUREG-0667] were sent to SMUD for action or that the recommendations that are relevant to the December 26, 1985 incident were implemented at Rancho Seco."

Id., p. 10-5.

In sum, when the Commission ordered an analysis of the ICS, B&W submitted an inadequate report (BAW-1564) and the NRC staff decided to postpone effective action by incorporating the issues into a generic Unresolved Safety Issue, A-47. When an accident at Oconee resulted in an NRC Bulletin requiring action, the NRC staff postponed resolution by incorporating the safety problems into the same generic safety issue. When a detailed study of the response of B&W plant behavior (NUREG-0667) made further recommendations, including prompt action on the previous recommendations in BAW-1564 and Bulletin 79-27, they still were not implemented.

Finally, and most remarkably, there is no prospect that they will be done in the foreseeable future because, according to the NRC Incident Investigation Team, "it appears that the analysis performed to date under USI A-47 does not address the long-term issues raised in Bulletin 79-27, BAW-1564, or NUREG-0667 . . . ." Id., pp. 10-5, 10-6, emphasis added. Thus, at the very same time that failure to immediately correct the ICS-related safety problems was being justified on the basis that the "long-term issues" would be resolved in the context of Unresolved Safety Issue A-47, the scope of that Unresolved Safety Issue was being interpreted to exclude consideration of these issues.

Both the President's Commission and the NRC's independent TMI-2 investigation team, in probing for the root causes of the TMI-2 accident, identified as a contributor the agency's practice of evading resolution of safety problems by labeling them "generic" and simply "studying" them forever.

> Because [generic] issues are regarded on a general basis and are not regarded as an impediment to individual plant licensing, little incentive exists for their resolution.

NRC Special Inquiry Group (The "Rogovin Report), Vol. II, Part 1, p. 50.

The Kemeny Commission concluded that removing generic issues from individual plant licensing might be acceptable:

> . . .if there were a strict procedure within NRC to assure timely resolution of generic problems . . . . However, <u>the evidence indicates that labeling a problem as 'generic' may provide a convenient way of postponing decision on a difficult question</u>.

Report of the President's Commission on the Accident at Three Mile Island, October 1979, p. 20, emphasis added.

Against this background, it is sadly ironic that the fundamental B&W problems should have been handled after the TMI-2 accident in the same way they were treated before -- by assignment to the bureaucratic netherworld of "generic unresolved safety issues," and subsequent evasion or, at best, "convenient postponements."

Even accepting the dubious assertion that the current safety reassessment may finally address the safety problems previously identified after the TMI-2 accident, the fact remains that neither B&W plant operating experience nor the Incident Investigation Team's report following the Rancho Seco accident supports the stated basis for claiming that the B&W plants are currently safe enough to operate. These show instead that the post-TMI Orders, requirements and recommendations applicable to the ICS have either not been implemented or they have proved to be inadequate to resolve the safety problems.

Example 2: PORV

Another example of TMI-2 lessons learned requirements that have not been implemented or, if implemented, have proved to be an inadequate resolution of the safety issues involves the pilot operated relief valve (PORV) on the pressurizer and position indication for the PORV. One specific post-TMI requirement was the seemingly straightforward order that testing be conducted to determine whether the PORV would operate reliably under the conditions of transients and accidents. See NUREG-0578, Section 2.1.2; NUREG-0660, Tasks II.D.1 and II.D.2; NUREG-0737, Item II.D.1. Another requirement was for provision of direct indication of the position of the PORV and integration of that position indication into the emergency procedures and operator training. See NUREG-0578, Section 2.1.3.a; NUREG-0660, Task II.D.3; NUREG-0737, Item II.D.3.

However, during the accident on June 9, 1985, at the Davis-Besse plant, the PORV stuck open (as it did during the TMI-2 accident), the operator relied on indication that the PORV had been electrically signaled to close (as the TMI-2 operator did), and did not look at the equipment installed after the TMI-2 accident to determine the actual position of the PORV. See NUREG-1154, "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," p. 3-10.

The NRC Incident Investigation Team for the Davis-Besse accident concluded that "[a]lthough the PORV is involved in the recovery from certain plant transients, its reliable operation has not been established by a suitable test program nor is its operational readiness verified by a periodic surveillance test." Id., p. 8-3, emphasis added. Since each B&W plant did not test its particular model of PORV, relying instead on a generic test program, the conclusion that the test program was inadequate also applies to other B&W plants.

Example 3: Emergency Feedwater

In July, 1979, the NRC concluded that "[t]he need for an emergency feedwater system of high reliability is a clear lesson learned from the TMI-2 accident." NUREG-0578, p. 10. In November, 1980, the NRC required that, "in the long term," the emergency feedwater system be upgraded to meet safety-grade requirements. The agency stated that "[t]he safety grade [emergency feedwater] system will be installed by July 1, 1981." NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.E.1.2. This deadline was later waived and, 8 years after the TMI-2 accident, there are at least two B&W plants, Rancho Seco and Three Mile Island Unit 1, which still do not have a safety grade emergency feedwater system.

In its response to this Petition, the NRC staff claims that:

> When the Sacramento Municipal Utility District finishes installing the safety grade emergency feedwater instrumentation and control system at Rancho Seco before restarting it, and when General Public Utilities finishes modifying the emergency feedwater system at Three Mile Island Unit 1 <u>during the current refueling outage</u>, the utilities will have satisfied all the NRC's requirements regarding emergency feedwater systems in B&W-designed plants.

NRC Staff Response, p.2, emphasis added.

The B&WOG goes one step beyond the NRC staff and claims that, "[c]onsistent with NRC direction and contrary to UCS' allegations, the AFW [auxiliary or emergency feedwater] systems at B&W plants <u>now</u> represent safety-grade systems." B&WOG Principal Response, p. 28, emphasis added.

With regard to Rancho Seco, it is beyond dispute that the emergency feedwater system is not safety-grade, the B&WOG's assertion notwithstanding. Whether it will be when or if the plant resumes operation remains to be seen, but NRC has demonstrated a propensity for delaying enforcement of requirements if plant operation would otherwise be prohibited. That is what has happened again and again in the case of TMI-1.

The NRC staff states in its initial response to the Petition that all requirements applicable to the emergency feedwater system would be completed "during the current refueling outage" at TMI-1. NRC Staff Response, p. 2. The refueling outage the staff referred to began in November 1986 and ended in March 1987, but contrary to the NRC staff and B&WOG's assertions, TMI-1 still does not have a safety-grade emergency feedwater system.

Extensive documentation sets forth the "requirement" that the emergency feedwater system modifications be completed prior to resuming operation after the just-completed refueling of TMI-1. For example, on February 18, 1987, the NRC informed the GPU Nuclear that "[a]ll modifications [to the emergency feedwater system] are to be completed prior to start up from your current (6R) refueling outage." J. D. Stolz, Director, PWR Project Directorate #6, NRC, to H. D. Hukill, Vice President and Director - TMI-1, GPU Nuclear, February 18, 1987, p. 1. The NRC staff also provided a Safety Evaluation Report concerning the unresolved safety issues affecting the TMI-1 emergency feedwater system.

The NRC Safety Evaluation Report concluded that a proposal put forward by GPU to block operation of the emergency feedwater system in certain circumstances "is not acceptable to the staff" and that it "significantly increases the risk" of failure of the emergency feedwater system. The staff conclusion that the upgraded TMI-1 emergency feedwater system would be acceptable, "assumes" that GPU's proposal would not be implemented and, instead, that the design will be modified to conform with NRC regulations. Id., Enclosure, p. 5.

Less than three weeks later, the staff changed its position and issued Amendment No. 124 to the TMI-1 operating license which specifically exempts TMI-1, for at least another year, from complying with the requirements which were first imposed in 1979. See, J. O. Thoma, Project Manager, NRC, to H. D. Hukill, GPU Nuclear, March 9, 1987, and enclosed Amendment No. 124. The NRC staff took this unilateral action without giving any prior notice to the parties who had litigated this issue in the TMI-1 restart hearing and in a 1984 show cause petition. Nor did the staff seek any relief from the schedule for completing the modifications which was imposed as a license condition by the Licensing Board and confirmed by the Commission during the restart proceeding.

Amendment No. 124 to the TMI-1 operating license was issued on March 9, 1987. The NRC Staff Response to the Petition is dated March 13, 1987. The B&WOG Principal Response to the Petition is dated April 6, 1987. Thus, the NRC issued a license amendment exempting TMI-1 from the requirement for a safety-grade emergency feedwater system before the filing of the NRC Staff Response and the B&WOG Principal Response, both of which claim erroneously that TMI-1 now has a safety-grade emergency feedwater system. We choose not to speculate on the reasons why both the NRC staff and the B&WOG make this inaccurate claim. In any event, TMI-1 still fails to meet the requirements imposed in 1979 after the TMI-2 accident.

In sum, the post-TMI changes do not provide a basis for finding reasonable assurance that the B&W plants are now safe and can operate during the "interim" while a safety reassessment is conducted. Despite the fact that the basic B&W design problems were known before the TMI-2 accident and were dramatically illustrated by that accident and despite the fact that it was promised that they would be resolved post-TMI, accidents continue to happen in B&W plants because these problems have not yet been resolved. Indeed there is an eerie similarity between NRC's post-TMI assessment of the B&W design problems in the Commission's 1979 Orders to shut down the B&W plants (See page 5, above) and the staff's current pronouncement:

> [I]n the course of the review of events at B&W reactors, it is apparent that for a given relatively benign loss of feedwater transient, a number of safety systems are challenged. A review of a similar initiating event on a CE [Combustion Engineering] or W [Westinghouse] plant reveals a lesser challenge to safety systems. Although there are no regulatory limits that restrict reliance on safety systems to particular scenarios, the response of B&W plant systems has become a concern in the engineering judgment of a number of analysts.

D. M. Crutchfield, Assistant Director for Technical Support, NRC, to H. Tucker, Chairman, B&WOG, March 13, 1986, Enclosure 1, "NRC Reassessment Program," p. 5.

However, despite their similarity, the NRC's 1979 assessment resulted in Commission Orders to shut down the B&W plants, but the NRC staff's 1986 assessment is accompanied by the assertion that the B&W plants can safely continue to operate despite the evidence that the post-TMI modifications have not been effective in correcting the safety problems with the B&W design.

<u>The Davis-Besse and Rancho Seco Accidents Are Evidence of the B&W Design Problems</u>.

Both NRC and the B&WOG attempt to downplay the significance of the Davis-Besse and Rancho Seco accidents by arguing that they were "plant-specific." In the case of the NRC, this argument appears to be largely a matter of belated word engineering. In the case of the B&WOG, it is part of a larger effort to deny that there is anything at all wrong with the level of safety in B&W plants.

To begin with, it is clear that the current generic "safety reassessment" was initiated by NRC as a result of the Davis-Besse and Rancho Seco accidents in 1985. Executive Director Stello's letter to the B&W owners of January 24, 1986, began:

> Following the TMI accident there was a growing realization of the sensitivity of Babcock and Wilcox (B&W) plants to operational transients. A number of recent events at B&W designed reactors have reinforced or concerns regarding these designs and lead us to conclude that there is a need to reexamine the basic design requirements for B&W reactors.

Furthermore, in response to a direct question from Congressman Matsui -- "What specific events triggered . . . the current reassessment . . . ?" -- the NRC Chairman listed the Davis-Besse and Rancho Seco "events" in June and December 1985, and other previous "less severe" events at the B&W plants. N. J. Palladino, Chairman, NRC, to Robert T. Matsui, U. S. House of Representatives, April 23, 1986, enclosure, p. 1. The NRC Chairman stated that as a result of these events, the NRC was "undertaking a broad based B&W generic reevaluation" which "will not only assess the B&W plants themselves, but will also determine the adequacy of the NRC's regulatory requirements and programs for the B&W plants." Id., p. 5, emphasis added.

Thus, the direct causal connection between the accidents at Davis-Besse and Rancho Seco and the current generic reassessment of the basic design of all B&W plants is clear. As the B&WOG states: "This program was initiated as a result of transients at Davis-Besse and Rancho Seco in 1980." B&WOG Principal Response, p. 9. As Chairman Palladino noted, the purpose of the safety reassessment is to determine the adequacy of the NRC's regulatory requirements for all B&W plants, not just for the Davis-Besse and Rancho Seco plants.

NRC now claims, in response to the Petition, that "lack of attention to detail in the care of plant equipment caused the Davis-Besse event" and while ICS "design weakness" caused the Rancho Seco accident, the other B&W plants are different and none other "would have responded in the same manner as Rancho Seco" did. NRC Staff Response, p. 2. Similar statements could have been made about the relationship of the TMI-2 accident and the other B&W plants. However, the fact that the TMI-2, Davis-Besse and Rancho Seco accidents had some plant specific aspects does not mean that they have no generic aspects. Furthermore, the NRC's internal investigations of the Davis-Besse and Rancho Seco accidents refute this effort to downplay the degree to which these accidents are manifestations of the same problems that have plagued the B&W plants for a decade.

For example, the NRC Incident Investigation Team concluded that "lack of effective engineering for determining the proper valve torque switch bypass contacts and improper implementation of specified settings" were the probable cause of some failures in the auxiliary feedwater system and that "this problem . . . could exist at other plants." NUREG-1154, p. 8-2, emphasis added.

NRC's investigation team also found that the operators at Davis-Besse did not initiate feed and bleed cooling as "required by the emergency procedures," and concluded that "operators at other plants may be reluctant to initiate [feed and bleed] or similar actions . . . ." *Id.*, pp. 8-2, 8-3, emphasis added.

Similarly, the NRC's investigation of the Rancho Seco accident reached basic conclusions that are applicable to all the B&W plants. One prominent example, as discussed above, involves the NRC Incident Investigation Team's finding that the NRC staff had decided to incorporate the long-term issues raised in B&W's

generic analysis of the ICS (BAW-1564), in the generic IE Bulletin 79-27, and in the NRC"s generic analysis of B&W plant behavior (NUREG-0667) into generic Unresolved Safety Issue A-47. See NUREG-1195, pp. 10-4, 10-5, 10-6. As detailed above, once "incorporated" into the generic unresolved safety issue, issues disappeared.

In addition, the Incident Investigation Team concluded that the "ICS performance upon restoration of power are [sic] still not fully understood" and reported "the Team's understanding that the B&W Owner's Group is planning to conduct an investigative program that will include this matter." Id., p. 10-6. Thus, even the B&WOG apparently considers the failure to understand this aspect of ICS behavior to be a generic issue stemming from the Rancho Seco accident, irrespective of the fact that the plants may not behave exactly alike upon restoration of power to the ICS.

Finally, the investigation of the Rancho Seco accident noted that, in 1983, the B&WOG reported the results of a generic B&W analysis (BAW-1791) "which predicted that an overcooling transient . . . could occur at B&W-designed reactors with a high probability . . . ." Id., p. 10-7. The Incident Investigation Team concluded that "it would appear that this analysis predicts that events comparable to the December 26, 1985 incident [at Rancho Seco] would occur approximately once every third year even if the [safety-grade] EFIC [Emergency Feedwater Initiation and Control] system were installed at all B&W-designed plants." Id. Since the B&W analysis (BAW-1791) "assumes that the EFIC system is installed," this means that even if the B&W plants have a safety-grade emergency feedwater system there is a high probability of a pressurized thermal shock accident at any of the B&W plants. Id.

In sum, the NRC's own accident investigators refute the attempts by the NRC staff and the B&WOG to downplay the significance of the Davis-Besse and Rancho Seco accidents in revealing the generic safety problems with the B&W design, as does fact that the current generic reassessment of the B&W plants and the NRC requirements applicable to the unique B&W design was initiated largely as a result of those accidents.

<u>Changes Recommended Or Implemented Since the Safety Reassessment Began Do Not Justify Continued Operation Of The B&W Plants</u>.

In its initial response to the Petition, the NRC staff presents a cursory description of the results to date of the B&WOG's reassessment program. The NRC staff states:

> Since the B&W plant reassessment began, almost one hundred (100) recommendations have been referred to the B&W plant owners, who have implemented, or are implementing, many of these recommendations. These changes have already improved plant safety by improving the ICS and the performance of the main feedwater system to reduce the number of challenges to safety systems due to feedwater transients. Because of these changes, the staff concludes that the plants are safer now than they were a year ago. The NRC found no undue risk in the utilities' interim operation a year ago, and, after taking into consideration the utilities' safety improvements, again finds no undue risk in allowing the utilities to continue to operate the plants.

NRC Staff Response, p. 2.[1]

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1 We assume that the "almost 100" recommendations to which the NRC refers are the 95 contained in BAW-1919, "B&W Owners Group Safety And Performance Improvement Program," Rev. 02, October 1986, App. J. Since NRC bases its letter to UCS of March 13, 1987, upon this version of the B&W document, UCS's instant reply does likewise. B&W has since produced Rev. 03 of BAW-1919, which it says NRC is currently reviewing and which the B&WOG claims contains 150 recommendations. We will submit our analysis of that version at a later date.

The NRC staff's response is a good example of its word engineering capabilities, but also indicates at best a grossly superficial review of the actual B&WOG recommendations. No attempt is made to discuss the substance of the recommendations, much less to show that, if implemented, the recommendations would solve the safety problems that led to the reassessment of the B&W plants. The staff does not quantify the word "many," does not distinguish between which of those "many" recommendations have actually been implemented at each B&W plant, which are formal commitments to be implemented at a later date, which are promises that may or may not be fulfilled and which are simply proposals as to which no promises or commitments have been made. In fact, the sum and substance of the NRC staff's response on this subject is that, since the staff concluded a year ago that the B&W plants could operate in the interim, the B&W plants were not required to do anything to improve safety, but whatever they did made the plants safer.

UCS's review of the "almost 100" recommendations of the B&WOG raises the question of whether the staff in fact did any substantive analysis. We have examined the substance of the recommendations and their implementation status. As discussed in detail below, the recommendations do not provide a basis for concluding that the B&W plants are adequately safe. To begin with, few of the recommendations have been implemented. Furthermore, when and if implemented, the recommendations would not resolve the fundamental safety problems which continue to cause complex accidents in B&W plants and which necessitated the current safety reassessment of both the design of B&W plants and the NRC requirements applicable to the unique B&W design.

As of October, 1986, the B&WOG reassessment program had made ninety-five (95) recommendations, of which only six (6) represented "commitments to the NRC." BAW-1919, "B&W Owners

Group Safety and Performance Improvement Program," Rev. 02, October 1986, App. J, Table 2-1. The 95 recommendations are distributed between sixteen "areas of improvement" identified by the B&WOG, as follows:

Area of Improvement	No. of Recommendations
Control Rod Drives	0
Emergency Feedwater	1
Integrated Control System (Includes the six commitments to the NRC)	22
Main Feedwater	37
Motor-Operated Valves	7
Main Steam System	5
Main Turbine System	2
NonDestructive Examination	0
NonNuclear Instrumentation	0
Operations Within Design Requirements	0
Plant Administration	7
Plant Operations	10
Reactor Coolant Pump Motor	0
Reactor Protection System	1
Steam Generators	0
Main Steam/Feedwater Isolation System	3

Id., Tables 1-1 and 2-1.

With regard to implementation of these recommendations at the B&W plants which are currently permitted to operate, i.e., Arkansas 1, Crystal River 3, Davis-Besse, Oconee 1, 2, & 3, and TMI-1, each owner has implemented only the following number of the 95 recommendations:

Arkansas Power & Light	-	1
Florida Power Corp.	-	15
Toledo Edison Company	-	11

Duke Power Company	-	24
GPU Nuclear	-	7

Id., App. J, Section 4, unnumbered 1-page table, ""Implementation Status Summary, Trip Reduction Recommendations, Total No. In System: 95." As can be seen, one utility has implemented about 25% of the recommendations and the others have done even less.

With regard to the 22 recommendations that apply to the Integrated Control System (ICS) and the 37 recommendations that apply to the main feedwater system (MFW), whose implementation the NRC staff specifically cites as having "already improved plant safety" (NRC Staff Response, p. 2), the owners of the B&W plants currently permitted to operate have implemented only the following number of those 22 and 37 recommendations:

B&W Owner	ICS	MFW
Arkansas Power & Light	1	0
Florida Power Corp.	8	0
Toledo Edison Company	8	0
Duke Power Company	4	4
GPU Nuclear	3	1

BAW-1919, App. J, Section 4, unnumbered 6-page table, "Trip Reduction Recommendations."

Since so few of the ICS and MFW recommendations have been implemented at so few of the operating plants for so short a time, the NRC staff's bald assertion that these recommendations have already improved B&W plant safety by improving the ICS and MFW is inherently unbelievable. The staff does not articulate a rational basis for this claim. Nor does the staff present any facts showing that the implemented changes have caused or contributed significantly to either the asserted "improved plant safety" or the asserted reduction in "the number of challenges to safety systems due to feedwater transients." NRC Staff Response,

p. 2. Considering the small number of changes made and the short time period that has passed, one can hardly take on faith the unsupported assertion that an effect has been documented.

Furthermore, in the case of at least 5 of the 12 ICS recommendations implemented by one or more owners of the operating B&W plants (as shown above, no single owner has implemented more than 8 of the ICS recommendations), the nature of the recommendation is such that it appears to be impossible that its implementation could improve the ICS or have an effect on either the observed or predicted frequency of challenges to safety systems due to feedwater transients. The 5 ICS recommendations of this nature are as follows:

> "Review procedures, annunciators, indicators, alarms, etc. to determine that the operator has all the necessary information" to determine whether a complete or partial loss of power to instrumentation has occurred. While the operator needs adequate information "to make timely decisions as to the nature of actions to take," the possible effects of those operator actions are either "a long outage or a quick shift to backup data and thus ensure core cooling and a safe plant shutdown," not an improvement in ICS or a reduction in challenges to safety systems due to feedwater transients. BAW-1919, App. J, p. 3-23.

> "Evaluate the restoration of ICS/NNI power and make appropriate changes to assure that the plant will remain in a safe [shutdown] state on restoration of power." Id., App. J, p. 3-63. Loss of ICS/NNI power can cause a reactor shutdown, i.e., challenge plant safety systems. The basis for this recommendation is that restoration of power can complicate plant recovery, as happened during the Rancho Seco accident. The recommendation apparently cannot improve the ICS or reduce challenges to safety systems due to feedwater transients.

> "Evaluate loss of ICS/NNI power and make appropriate changes to assure that . . . the plant will go to a known safe state without any operator action required." This recommendation may reduce the "probability of complex transients on loss of ICS/NNI power" and the "demands placed on operators during transient conditions," but apparently it cannot improve the

ICS or reduce challenges to safety systems due to feedwater transients. Id., p. 3-65, emphasis added.

"Review training records to ensure operators have had training on loss of ICS power." The basis for this recommendation is a "B&WOG . . . commitment to the NRC," but no "expected benefit" is given by the B&WOG. Id., p. 3-67. In any event, this recommendation cannot improve the ICS or reduce challenges to safety systems due to feedwater transients.

"Familiarize operators with the Rancho Seco event." Once again, the B&WOG states that this is a "commitment to NRC," but gives no "expected benefit." Id., p. 3-69. In any event, this recommendation cannot improve the ICS or reduce challenges to safety systems due to feedwater transients.

If the table appearing on page 30 above, which shows how many of the 22 ICS recommendations have been implemented at the operating B&W plants, is revised to include only those which have the actual potential to improve the ICS or reduce challenges to safety systems due to feedwater transients, the following is the result:

B&W Owner	ICS
Arkansas Power & Light	1
Florida Power Corp.	3
Toledo Edison Company	5
Duke Power Company	1
GPU Nuclear	0

In sum, UCS's review of the substance and implementation status of the B&WOG's recommendations for ICS and MFW improvements leads to the conclusion that the NRC staff's assertion that "[t]hese changes have already improved plant safety by improving the ICS and the performance of the main feedwater system to reduce the number of challenges to safety systems due to feedwater transients," is completely unsupported by any facts presented and, based upon a review of the B&WOG document, is invalid. NRC Staff Response, p. 2.

A review of the substance of the recommendations also discloses instances where the gross number of recommendations has been inflated by dividing the work needed to resolve a single problem into multiple separately-numbered recommendations. This creates the impression that more improvements are proposed than is actually the case.

For example, the B&WOG "determined that during the past six years twenty reactor trips" (i.e., reactor shutdowns) were caused by the "loss of input signals to the ICS" The B&WOG also determined that there are six ICS input signals whose loss will cause a reactor trip and proceeded to count as six separate items the recommendation intended to achieve a reduction in the frequency of reactor trips caused by a loss of ICS input signals. BAW-1919, App. J, pp. 3-1 through 3-12.

A more striking example of excessive parsing occurs in the case of the work needed to determine whether the safety-related motor-operated valves will perform as required under emergency conditions. The substance of the seven B&WOG recommendations concerning this one problem is as follows:

1) Obtain the analytical methods used by the valve and valve operator manufactures in order to understand how motor operators are sized and how limits are established;

2) Develop a procedure to determine whether the valves and their motor operators used in safety systems are the proper size;

3) Assure that the torque switch bypass limit switch in motor-operated valves is properly set;

4) Assure that the open direction torque switch is properly set in all safety related wedge seating valves;

5) Review maintenance procedures for all safety-related valves to ensure that there are proper instructions for setting the torque switches and the torque switch bypass limit switches;

6) To the extent practicable, all safety related motor operated valves should be challenged to open and close under differential pressures which simulate worst case operational and accident conditions; and

7) Institute formal motor operated valve training programs for engineers, electricians and mechanics who perform maintenance on these valves, and for reactor operators who could benefit from an understanding of valve operators when confronted with off-normal events.

Id., App. J, pp. 3-81, 3-83, 3-85, 3-87, 3-89, 3-91, 3-93.

According to the B&WOG, the "expected benefits" from implementing these seven recommendations on motor-operated valves are as follows:

- The B&W owners will obtain "a thorough understanding of how motor operators are sized and how limits are established;"

- For all safety-related motor-operated valves, it will be determined whether the valve "operator is over- or under-sized;"

- The B&W owners will "[e]nsure that the bypass limit switch serves its function by allowing the valve to unseat before the open torque switch is placed in the [motor] circuit;"

- The open direction torque switches will be set to "enhance valve operability while providing protection against damage due to overloads;"

- Proper maintenance procedures will "[e]nsure that the safety related . . . motor operated valves are capable of performing as required;"

- Testing of safety-related motor-operated valves will "[p]rovide assurance that these valves will perform as required under emergency and plant upset conditions."

Id.

In sum, the totality of the possible "benefits" to be expected when and if these seven (7) recommendations are implemented is that the motor-operated valves may be expected to operate in accord with the assumptions made during the initial licensing of the B&W plants. This does nothing to address the inherent B&W design problems or the inadequacy of NRC requirements applicable to the unique B&W design.

Perhaps the most troubling aspect of these particular recommendations is the disclosure that the B&W plant owners do not know whether the correct size valves are installed in safety systems, do not have the knowledge to determine what is the correct size, and do not know how to set properly the torque switches and bypass limit switches in motor-operated valves. In sum, neither the NRC nor the B&WOG is yet in a position to determine whether the motor-operated valves used throughout safety systems in B&W plants will perform as required during transients and accidents.

Furthermore, the implementation schedule for the B&WOG recommendations shows that they will not be implemented for years, if ever. For example, the "implementation status" of the recommendation to review the maintenance procedures for all safety-related valves to ensure that the valves are capable of performing as required shows the following:

> Duke Power Company is implementing the recommendation, but revision of the Oconee Units 1, 2 and 3 maintenance procedures is not scheduled to be completed until January 1, <u>1989</u>.
>
> GPU Nuclear was "evaluating" the recommendation and was scheduled to make a determination on whether to <u>begin</u> implementation at TMI-1 by February 1, 1987.
>
> Toledo Edison was "evaluating" whether to implement the recommendation at Davis-Besse with no schedule for a decision.

> No schedule for evaluating or implementing the recommendation was given for Arkansas Unit 1, Crystal River Unit 3, Rancho Seco and Bellefonte Units 1 and 2.

Id., App. J, p. 3-90.

Similarly, the "implementation status" of the recommendation to ensure that the torque switch bypass switches in all safety-related motor-operated valves are set properly shows the following:

> Duke Power Company was scheduled to complete implementation at Oconee Units 1, 2, & 3 by 1990.
>
> GPU Nuclear was "evaluating" the recommendation and was scheduled to decide, by February 1, 1987, whether to begin implementation at TMI-1.
>
> Toledo Edison was "evaluating" whether to begin implementation at Davis-Besse with no schedule for a decision.
>
> No schedule for evaluating or implementing the recommendation was given for Arkansas Unit 1, Crystal River Unit 3, Rancho Seco and Bellefonte Units 1 and 2.

Id., App. J, p. 3-86.

In sum, the document relied upon by the NRC staff as demonstrating safety improvement at the B&W plants reveals that only one B&W owner is implementing these two rather basic recommendations for reviewing valve maintenance procedures and ensuring the correct setting of switches that could prevent valve operation, and even that utility is not scheduled to complete the work until 1989 and 1990.

Further, three of the recommendations would negate, in whole or in part, the requirements imposed by the Commission as a result of the TMI-2 accident and one repeats an NRC-imposed requirement which has not yet been implemented.

Two recommendations which reverse a post-TMI lessons learned requirement are: 1) "Restore the high pressure reactor trip setpoint to 2355 psig . . . ." (Id., App. J, p. 3-15) and 2) "Increase the setpoint for reactor trip on high pressure from [the] current value of 2300 psig to a value of 2355 psig." Id., App. J, p. 3-61. After the TMI-2 accident, NRC required that the high pressure reactor trip setpoint be lowered in order to reduce the instances where the pressurizer pilot operated relief valve (PORV) would open and might fail to close. The PORV stuck open during the TMI-2 accident, causing a loss-of-coolant accident.[2] The B&WOG acknowledges that raising the setpoint of the high pressure reactor trip will increase "the frequency of opening the PORV during anticipated transients," and this is inconsistent with the post-TMI requirement, but claims that it will have a "negligible impact." BAW-1919, App. J, p. 3-61. The post-TMI requirement to lower the high pressure reactor trip setpoint was considered so important that the NRC ordered it to be done before the B&W plants were allowed to resume operation after the TMI-2 accident. Thus, in 1979, NRC considered this change to be necessary to provide reasonable assurance of safety. It is UCS's view that reversing this requirement constitutes a license amendment that cannot be issued legally without prior notice and opportunity for hearing.

The third recommendation which negates a post-TMI requirement is: "Raise the Anticipatory Reactor Trip or Turbine Trip arming point from its current rating of 20% power to a higher level . . . ." Id., App. J, p. 3-59. After the TMI-2 accident, as another condition to be met before the B&W plants could resume operation, the NRC required the installation of this anticipatory reactor trip in order to reduce the instances where

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2 The PORV also stuck open during the Davis-Besse accident in June, 1985. NUREG-1154, p. 3-10.

the PORV could open. Raising the setpoint would increase the chances of opening the PORV, thus diminishing the level of safety the Commission determined was necessary as a result of the TMI-2 accident.

The fact that the B&WOG now proposes to reverse these post-TMI requirements is indicative of the weakness in the general approach adopted by the B&WOG: rather than addressing the fundamental design problems reflected in the erratic and unpredictable response of the B&W plants to abnormal and accident conditions, the owners group is plowing the same ground already gone over in the immediate aftermath of the TMI-2 accident. The focus is on the symptoms rather than the disease; the effort is almost solely to reduce challenges to the unreliable safety systems rather than to make the safety systems more reliable in the event they are challenged, which they will certainly be. The fact that this effort has now resulted in some recommendations which are the direct opposite of the post-TMI recommendations illuminates the futility of the effort.

The recommendation which repeats a post-TMI requirement is: "conduct post-maintenance and surveillance PORV testing which should include an inservice functional test." The B&WOG states that "operational experience has shown that a <u>high</u> number of cycles contributes to problems such as valve damage and seat leakage" and observes that during the Davis-Besse accident, "the PORV operated automatically <u>three</u> times, and did not reseat properly the third time." <u>Id</u>., App. J, p. 3-101, emphasis added. After the TMI-2 accident, the NRC required qualification testing of the PORV to determine whether the valves could operate properly "under expected operating conditions for design basis transients and accidents." NUREG-0578, Section 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves," p. A-8. Nevertheless, as discussed above on page 19, NRC's investigation of the 1985 Davis-Besse accident showed that "reliable operation [of the PORV] has not been established by a suitable test program, nor is its operational readiness verified by a periodic surveillance test." NUREG-1154, p. 8-3. Nonetheless, although this B&WOG recommendation reiterates the need for PORV testing, the proposed testing falls short of the qualification testing program shown to be necessary by the TMI-2 accident and purportedly imposed upon the owners years ago.

Many of the B&WOG's recommendations are such that, even if implemented, they will make little or no improvement in safety at the B&W-designed plants. For example:

> "Delete from the ICS, the feedwater temperature correction to the feedwater demand function." However, since "the temperature inputs [to the ICS] do not have a troublesome history," this recommendation is unlikely to have any significant impact on the frequency of reactor trips. BAW-1919, App. J, p. 3-11.
>
> "Each utility <u>determine</u> that the [ICS] grid frequency error circuit . . . <u>is</u> inoperable." However, since the "information is that the grid frequency error circuit <u>has been [rendered inoperable] by all operating plants</u>," and "as a result <u>nuisance-type fluctuations</u> in power at full power <u>have been eliminated</u>," this recommendation is not calculated to have any effect whatsoever on improving safety or reducing the frequency of complex transients. <u>Id</u>., App. J, p. 3-21.
>
> "Install a <u>monitoring</u> system in main FW [feedwater] pump trip circuitry to <u>document</u> the primary causes of MFWP trips." By examining past history, the B&WOG determined that the cause of "25% of the total [main feedwater pump] trips" was listed as "not specified." <u>Id</u>., App. J, p. 3-27,

emphasis added. Since this recommendation only involves installing a monitoring system to try to determine the cause of MFW pump trips, it could be classified as implemented without, in fact, determining or correcting the causes of MFW pump trips.

"Ensure sufficient annunciators and trip signals are present and will seal-in to assure proper operator action and later transient evaluation in the [feedwater] water supply system. Combine this recommendation with [the FW monitoring system recommendation above]." Id., App. J, p. 3-37, emphasis added.

"Ensure that TAP [Transient Assessment Program] reports include specific information regarding events where human errors occur, such as what the error was, title of person making it and why did it occur." The basis given for this recommendation is that "insufficient information on human error events makes it difficult to develop recommendations to prevent their occurrence." Id., App. J, p. 3-57. Thus, the essence of this "recommendation" is that reports to be used in the B&WOG's safety reassessment program should contain the information deemed necessary to develop recommendations which might improve B&W plant safety.[3]

"Use the [B&WOG] Transient Assessment Committee's Trip Investigation/ Root Cause Determination Program." If implemented (only one B&W owner has done so), this "recommendation" may result in "improved root cause

3 This recommendation is an example which should be recalled when evaluating the significance of the NRC staff's statement that "the B&W owners . . . have implemented, or are implementing, many of these recommendations." NRC Staff Response, p. 2. In fact, this is one of the few recommendations that has been implemented by three B&W owners. BAW-1919, App. J, p. 3-58.

determination and effectiveness of correction to reduce the number of plant trips, thereby reducing challenges to safety systems." Id., App. J, p. 3-79. This is another example of a recommendation which, if implemented, may enable the B&WOG to develop recommendations.

"Personnel who may make emergency notifications should receive training to assure that they are familiar with the type of information which must be provided." The "expected benefit" of the recommendation, which has been implemented by only one B&W owner, is an "improved ability to provide the governmental agencies with required information." Id., pp. 3-117, 3-118. This recommendation has no potential to resolve the inherent safety problems of B&W plants.

To summarize, three recommendations would negate a post-TMI requirement, one repeats (but in weakened form) a post-TMI requirement, and seven are essentially record-keeping measures which may or may not make it possible to come up with recommendations sometime in the future.

Finally, our review shows that the B&WOG recommendations are primarily directed to improving the economic performance of the B&W plants, not to improving the overall safety of the B&W design. The Advisory Committee on Reactor Safeguards reached a similar conclusion in July 1986. See Petition, p. 21. The October, 1986, revision of the B&WOG program has not corrected this problem. Therefore, the B&WOG program appears incapable of fulfilling the second of its two stated objectives:

> Reduce the number of trips and complex transients on B&W Owners Group plants and ensure acceptable plant response during those trips and transients which do occur.

BAW-1919, p. II-4, emphasis added.

In this regard, UCS has reviewed the 95 recommendations and the B&WOG's statements of the "expected benefit" to be achieved by implementing the recommendations. As can be seen from the tabulation below, most of the recommendations are directed toward reducing the number of trips and transients, not to improving plant behavior following those trips and transients which do occur.

> Out of 22 recommendations for improvement to the Integrated Control System, 15 recommendations (Nos. 1 through 11, 13, 21, 38 and 39) are predicted to reduce the number of trips and transients, 4 recommendations (Nos. 12, 32, 33 and 37) are intended to improve plant response to trips and transients, and 3 recommendations (Nos. 34, 35 and 36) have no identified expected benefit.
>
> Out of 37 recommendations for improvement to the main feedwater system, 27 recommendations (Nos. 15 through 18, 20, 66, 67, 68, 72 through 77, 79, 80, 82 through 85, 87 through 90, 93, 94, 95) are directed to reducing the number of trips and transients, 7 recommendations (Nos. 69, 70, 71, 78, 81, 86, and 91) are directed to improving plant response and 3 recommendations (Nos. 14, 19 and 92) will have no effect on the number of trips and transients or the plant response.
>
> All seven recommendations (Nos. 41 through 47) for improvements to motor-operated valves can be said to improve plant response by assuring the valves will work properly.
>
> Out of 10 recommendations for improvement to plant operations, 9 recommendations (Nos. 26, 51, 58 and 60 through 65) are directed to improving plant response and one (recommendation 59) will affect neither the number of transients nor plant response.

Out of 7 recommendations to improve plant administration, two (Nos. 27 and 28) are directed to reducing transients, three (Nos. 55, 56 and 57) are directed to improving plant response, and two (Nos. 29 and 40) will affect neither.

Out of the remaining 12 recommendations for improvement to the emergency feedwater, main steam, main turbine, reactor protection and main steam/feedwater isolation systems, eight recommendations (Nos. 22, 23, 25, 30, 31, 52, 53 and 54) are predicted to reduce trips and transients, and four (Nos. 24, 48, 49 and 50) are directed at improving plant response. BAW-1919, App. J, Section 3, Trip Reduction Recommendations.

To summarize, out of 95 recommendations, 52 are directed toward reducing the number of trips and transients, 34 are directed toward improving plant response to trips and transients that do occur, and nine will have no effect on either the number of trips or plant response. However, these raw numbers may give the unjustified impression that substantial work is being done to improve plant response to transients and accidents. There is no substitute for evaluating the substance of each of the recommendations, something the NRC staff apparently did not do when preparing its response to this Petition. For example, five recommendations which are classified above as improving plant response to transients are:

> "The activities of plant . . . personnel should be coordinated to facilitate timely access to critical equipment [during an accident or transient]. <u>Consideration</u> should be given to: a. dispatching security . . . personnel to the control room to assist in gaining access to secured areas; b. minimizing the number of keys required to gain access to locked valves, doors, panels . . . ; c. providing access to bolt cutters as a backup measure." <u>Id</u>., App. J, p. 3-109, emphasis added.

> "Move chain link fences as necessary to provide better access to critical components." <u>Id</u>., App. J, p. 3-111.

> "Where problems have been identified with gaining access to critical components because of . . . fire barriers, <u>consider</u> ways to improve access to these components." <u>Id</u>., App. J, p. 3-113, emphasis added.
>
> "Stress in operator training that emergency procedures are to be followed explicitly, even when such procedures are considered as drastic actions." <u>Id</u>., App. J, p. 3-119.
>
> "Establish a means of systematically identifying high priority operator tasks" that are required in high stress, emergency conditions and develop short-term and "long-term requalification training programs for these tasks." <u>Id</u>., App. J, p. 3-121.

The above examples show that when one goes beyond the bare, inflated, numbers of recommendations into the content of the work done by the B&W Owners Group, and relied upon by the NRC staff, it becomes apparent that the B&W safety reassessment, 1987 version, exhibits the same mindset that has doomed previous B&W safety assessments to failure and ultimately led to abandonment of the post-TMI promises for long-term fundamental change. Basic changes to the B&W design have been ruled out <u>a priori</u>.

Note, for example, that no consideration whatever is given to upgrading the Integrated Control System, Non-Nuclear Instrumentation and their power supplies to safety-grade, as was recommended and believed to be "already in progress" in October, 1979. R. D. Martin, to J. M. Allan, "Operations Team Recommendations, October 10, 1979, pp. 1, 18, 19. Similarly, the B&W owners are silent on "the advantages of incorporating sprays at the top of the hot legs (and vessel head) to condense trapped steam" which blocks water circulation through and threatens to uncover the core during a small break loss-of-coolant accident. H. R. Denton, to R. B. Minogue, December 30, 1981, p. 3. There are no recommendations to perform or incorporate the results of "sensitivity studies of possible modifications which would reduce the response of the OTSG to secondary coolant flow perturbations" which might "mitigate overcooling and undercooling events." NUREG-0667, p. 10. Nor is there any indication that either the B&WOG or the NRC staff intends to resolve the "broader and more fundamental questions in the design and operation of nuclear power plants" which the NRC claimed, in 1980, would be addressed, such as: A) the "disparities between the description and evaluation of accidents in the licensing review of a Safety Analysis Report and the actual response of the plant and its operators;" and B) whether accident analyses should "include multiple equipment failures and more explicit consideration of operator actions and inaction, rather that employing the conventional single failure criterion."[4] NUREG-0578, p. 17.

In sum, even should the B&WOG recommendations be implemented (which they are not yet), the result would be at best to reduce to some unknown degree of unknown significance the number of transients and accidents in B&W plants. However, transients and accidents will surely still continue to occur, indeed transients are expected to occur, and NRC will be no closer to knowing either how to predict B&W plant response or how to ensure that the response is acceptable. Considering that NRC's original goals for the safety reassessment of B&W plants were to "reassess the overall safety of B&W plants" and "determine whether the present set of [NRC] requirements" for B&W plants provides an adequate level of safety (V. Steel, to Hal Tucker, January 24, 1986), it is quite apparent that the B&WOG program, as least as developed and implemented to date, cannot be expected to achieve these objectives.

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4 Considering that both the Davis-Besse and the Rancho Seco accidents involved multiple failures, common-mode failures, and operator errors, this subject is particularly important for the safety reassessment of B&W-designed plants. However, it is not included in the B&WOG's program and, for its part, the NRC staff appears content to do nothing.

Indeed, as the B&WOG reply to this Petition makes clear in its totality, the B&WOG has begun with the unswerving premises that overall B&W plant safety is adequate and that current regulatory requirements are adequate, as were the regulatory requirements in effect when the plants were originally licensed. This is the only conclusion to be drawn from the B&WOG's repeated citation as evidence for the proposition that the plants are safe the fact that they were deemed so when originally licensed and again after the post-TMI fixes. As noted above, the B&WOG is plowing the same ground covered in the immediate aftermath of TMI-2 and is no more likely now than it was then to achieve changes that will ensure that B&W plants are made safe.

<u>The Original Basis for Licensing the B&W Plants Was Not Conservative and Without Integral Testing Cannot Be Made So</u>.

The NRC staff asserts that operation of the B&W plants does not pose undue risk because the Commission's original decisions to grant operating licenses to the B&W plants were based on analyses in each plant's Final Safety Analysis Report (FSAR) which are sufficiently conservative to provide assurance of safety. However, the TMI-2 accident and the 1985 accidents at the Davis-Besse and Rancho Seco plants demonstrate that those assertions are without merit.

In its response to the Petition, the NRC staff states that the analyses of design basis accidents presented by the B&W owners in each plant's FSAR were based on "conservative computer codes" which were used to calculate the "worst-case values of such system parameters as pressure, [fuel] cladding temperature, and departure from nucleate boiling ratio." The NRC staff concludes that:

> "Therefore, while the NRC continues to perform experimental and computer work in order to understand

> and predict more accurately the behavior of B&W-designed plants under accident conditions, the plant safety analyses currently on file predict conservative results sufficient to conclude that continued plant operation poses no undue risk to the public health and safety. The NRC, therefore need not shut down or limit the operating power level of B&W-designed plants."

NRC Staff Response, p. 3.

This response is utterly devoid of details, substance and validity. It is contrary to the operating experience of the B&W plants, the agency's own documents and the stated reasons why the staff concluded that a reassessment of the overall safety of B&W plants was needed.

The Final Safety Analysis Report (FSAR) is the principle part of an application for an operating license. The FSAR contains analyses of certain so-called Design Basis Accidents, such as a loss of main feedwater (but not loss of main <u>and</u> emergency feedwater, as occurred at TMI-2), and a rupture of the reactor coolant system piping (but not a rupture of the reactor vessel, as could have happened during the Rancho Seco accident, particularly if the plant had been older). The FSAR purports to show that in the event of any one design basis accident (combinations of accidents, such a loss-of-coolant accident and a total loss of electrical power, are not considered), the consequences will be acceptable. For example, it is claimed in the FSARs that, in the event of a design basis accident, the amount of fuel cladding that will react with steam to form hydrogen will be less than 5% (during the TMI-2 accident, between 30% and 50% of the cladding reacted to formed hydrogen) and that temperature of the fuel will remain well below its melting point (which it did not at TMI-2).

The FSAR analyses use computer models or "codes" to calculate the consequences of the design basis accidents. It is the NRC staff's current claim that the computer codes used to analyze B&W plant were "conservative." NRC Staff Response, p. 3. However, in order to make a claim that an analysis is conservative, it is necessary to have a factual base of information onto which a conservative factor can be added. When there is no factual base, or an inadequate factual base, then as much as one might wish to be conservative, there is no basis for knowing that has been accomplished. For example, samples of structural steel are destructively tested to determine the "test" strength of the steel. Then, in designing a structure, a lower "design" strength is used to offset possible reductions in the strength of the steel resulting from fabrication or construction variations, thus making the design conservative. However, if the steel was never tested, there is no basis for assurance that the design strength is in fact conservative. This is the situation that exists with regard to the computer codes used to analyze accidents in B&W plants.

Integral test facilities were built for the design used in Westinghouse and Combustion Engineering pressurized water reactors, but none were built for the design used in B&W plants. Integral test facilities represent the entire nuclear steam supply system, e.g., the reactor, reactor coolant piping, and steam generators, although the test facility may include only one steam generator whereas the actual plant contains two or more. An "integral" test facility is designed to test the behavior of the plant's systems and components in combination. In contrast, "separate effects" test facilities can investigate particular phenomena of interest, e.g., the rate of temperature rise in the fuel rod during a simulated loss of coolant accident.

The computer codes which were used to analyze the consequences of accidents in B&W plants were developed from tests done in integral test facilities built for the Westinghouse and

Combustion Engineering design. However, the B&W design is different than those designs in important, fundamental ways that have a significant impact on the behavior of B&W plants during accidents. The computer codes used to analyze transients and accidents in B&W plants "are not configured for the once through steam generators used in B&W reactors" and there are important "system volume and piping arrangement differences from the C.E. and Westinghouse design for which the [computer codes] were configured . . . ." Even with the use of "correction factors," the B&W "safety analysis results are characterized with a greater degree of uncertainty than desired." V. Stello, to S. Chilk, Secretary, NRC, April 30, 1986, enclosure, p. 8. Thus, there is no basis for claiming that use of computer models derived from the C.E. and Westinghouse tests to analyze the unique B&W design results in a conservative analysis, even if the attempt was made to do so by applying "correction factors," which likewise have little or no factual basis. One might as easily claim that a conservative design strength for plastic was determined by applying a correction factor to the test strength of steel.

A similar situation currently exists in the analysis of accidents in B&W plants. For example, the emergency feedwater system is designed to spray water onto the bundle of 15,000 tubes in the B&W once through steam generator in order to remove heat from the reactor coolant system during an accident. However, it is not known how many of the tubes the emergency feedwater actually reaches. The NRC's description of this current lack of knowledge and its effects on accident analyses is as follows:

> Auxiliary feedwater in B&W Once Through Steam Generators (OTSG) is injected as a spray at the top of the tube bundle. <u>Flow characteristics of this spray</u>, e.g., depth of penetration into the tube bundle and counter current flow with upflowing steam at the tube support plates <u>is [sic] not well understood or modeled in existing [computer] codes</u>. <u>Consequently, the</u>

> primary-to-secondary coolant system heat transfer, a very important characteristic during both overcooling and undercooling transients [i.e., during design basis accidents], e.g., steam line break or feedwater line break, is also not well understood. NRC has initiated an OTSG experimental program which consists of full-scale single tube and multitube sector models of the OTSG. The data from this program will provide information for improving the code modeling of OTSG for B&W plants.
>
> The adiabatic single tube OTSG test fixture is in operation at the INEL. Characterization data has been completed and has verified that the air flow capacity is sufficient to obtain complete flooding at the tube support plate under maximum expected auxiliary feedwater flow rates. Operation of the test facility to obtain flooding data is currently in progress.

T. A. Rehm, Assistant for Operations, Office of the EDO, to the Commissioners, "Weekly Information Report - Week Ending February 20, 1987," February 25, 1987, enclosure E, p. 2, emphasis added. UCS seriously questions whether a test facility with a single tube is an adequate representation of the 15,000 tube steam generator, but it is certainly not an integral test facility. In any event, since basic information such as heat transfer from the reactor coolant system during design basis accidents is not known, it is clear that the NRC staff has no rational basis for claiming that the accident analyses for B&W plants, whether performed when the plants were originally licensed or in 1987, are conservative.

In addition to using computer codes that do not model the unique B&W design, the FSAR analyses of design basis accidents incorporate a number of assumptions, some of which are inappropriate and others dubious. It is assumed, for example, that no operator action is required for 10 or 20 minutes, which the NRC said "is misleading and may be nonconservative." NUREG-0578, p. 18. It is also assumed that, in addition to the accident and the failures caused by the accident, no more than one single failure will occur. It is assumed that emergency procedures exist for design basis events such as loss of power to the ICS, that the operators will follow the emergency procedures, that motor-operated valves are correctly-sized and that their torques switches and torque switch bypass switches are correctly set. In sum, it is assumed that there will not be "disparities between the description and evaluation of accidents [in the FSAR] and the actual response of the plant and its operators," and that "events" which are "not foreseen, analyzed, or prepared for" will not occur NUREG-0578, p. 17. Such assumptions were shown to be not even realistic, much less conservative, by the accidents at TMI-2, Davis-Besse and Rancho Seco. Accident analyses based on these assumptions cannot be considered conservative.

The TMI-2 accident was a complex accident involving multiple equipment failures and operator errors. The destruction of the reactor core demonstrated that the allegedly "conservative" analyses used as a basis for licensing the plant were unrealistic. Both the sequence of events during the accident and the consequent fuel melting were events which the FSAR analyses deemed incredible. Nothing since then has altered the fact that the original analyses performed when the B&W plants were originally licensed were not conservative.

The post-TMI modifications were ineffective in resolving the safety problems inherent in the B&W design largely because many of the NRC's own determinations about what needed to be done to understand B&W plant behavior were never translated into requirements applicable to the B&W plants. As long ago as December, 1981, the NRC identified major deficiencies in the computer analyses used to license B&W plants, the same computer analyses which the staff calls "conservative" in its response to this Petition:

> "Recent analyses have shown the system pressure to behave in a cyclic manner that could be confusing to the operator during certain small break conditions.
>
> "Prior to [the] TMI-2 [accident], the repressurization phenomenon had not been identified and was only predicted to occur after a close model examination and modeling change was made. We do not know if the unique oscillatory phenomenon is real or an artificiality of the [computer] analyses. * * * We believe the predicted phenomenon could produce false symptoms of other events . . . and lead to incorrect operator actions which could result in more severe consequences than now predicted for SBLOCA [small break loss of coolant accident].
>
> "The phenomenon described is unique to B&W-designed NSSSs [nuclear steam supply systems] because of the hot leg and Once-Through-Steam-Generator (OTSG) configuration.
>
> "At present, we have no confirmatory integral systems data with which to verify the acceptability of the predicted behavior of transients and accidents including small break LOCAs in B&W-designed reactors. Also, the long-term hydraulic stability of the plant following a SBLOCA has never been analytically or experimentally confirmed."

H. R Denton, Director, Office of Nuclear Reactor Regulation, to R. B. Minogue, Director, Office of Nuclear Regulatory Research, December 30, 1981. Thus, not only is there insufficient information about plant behavior to "verify" that the computer analyses accurately reflect that behavior, there is information affirmatively indicating that accidents more severe than predicted can occur.

Interestingly, both the above comments on the deficiencies of the computer analyses of the B&W design and the claim in the response to this Petition that these analyses are conservative were made by the same NRC staff official, Harold Denton, then the Director of Nuclear Regulation. It appears that the NRC staff has one position for public consumption and the opposite position internally on the merits of the issues.

In any event, the facts that the computer analyses of the B&W design and the Final Safety Analysis Reports currently on file for B&W plants are not conservative were confirmed again by the NRC's investigations of the Davis-Besse and Rancho Seco accidents in 1985.

For example, the David-Besse accident involved "multiple equipment failures" which resulted "in a transient beyond the design basis of the plant," including "several common-mode failures affecting redundant safety-related equipment." NUREG-1154, p. 8-1, emphasis added. In addition, NRC's Incident Investigation Team concluded that "neither the SFRCS [steam and feedwater rupture control system] nor the auxiliary feedwater system . . . meet the single failure criterion for all design basis accidents." Id., p. 8-2.

Similarly, NRC investigators found evidence that the Rancho Seco Final Safety Analysis Report was not conservative. "It is not clear that the overcooling transient was within the FSAR analysis of the Rancho Seco plant. * * * In addition, the Rancho Seco FSAR analysis of main steam line breaks appears to be flawed and nonconservative in that it assumes that the non-safety related ICS operates successfully to mitigate the consequences of the accident." NUREG-1195, p. 10-3.

There is also remarkable consistency over a period of years in the NRC staff's determination that integral system testing is essential to gaining an understanding of how B&W plant will behave during transients and accidents. The NRC staff has reiterated this need several times: in 1980, in the staff's generic evaluation B&W plant behavior following a small break loss-of-coolant accident (NUREG-0565); in 1981, in Mr. Denton's request to the Office of Nuclear Regulatory Research on the need for a test facility to model the unique B&W design (H. Denton, to

R. Minogue, December 30, 1981); in 1986, in the NRC's investigation of the Davis-Besse accident (NUREG-1154); and again in 1986, in Mr. Stello report to the Commissioners on the impact of budget cuts on NRC's ability to assure safety (V. Stello, to S. Chilk, April 30, 1986).

> "In order to fully understand response to small reactor coolant system breaks, it is necessary to verify the calculational model used to predict the small break response. Many of the individual models within the overall B&W evaluation model have previously undergone comparisons against experimental data as well as other methods of verification. However, the accident at TMI-2 has emphasized the importance of certain phenomena which are expected to occur during a small break LOCA. From this, the staff has identified certain models, methods, or features of the evaluation codes which require more extensive verification.
>
> "In addition to verification of individual models, it is also necessary to assure proper interaction of these models within the overall systems evaluation model. <u>This is accomplished through verification by comparison to integral systems tests</u>."

NUREG-0565, p. 4-7, emphasis added.

> The analysis methods used by B&W "are satisfactory <u>for the purpose of predicting trends</u> in plant behavior following a small LOCA. However, several concerns regarding the small break model have been identified . . . and should be evaluated before the B&W methods can be considered for NRC approval under 10 CFR 50.46. The comparison of the total analysis method with available small break integral data has indicated <u>large uncertainties</u> in the calculations. Accordingly, <u>integral verification of the methods should be included as part of the approval under 10 CFR 50.46</u>."

<u>Id.</u>, p. 4-10, emphasis added.

> "A final aspect of our <u>data needs</u> regarding the capability of B&W plants to accommodate small breaks is (a) the potential benefits of high point vents restoring natural circulation and (b) the advantages of incorporating sprays at the top of the hot legs (and vessel head) to condense trapped steam. <u>We believe</u>

> <u>that the benefits of these modifications need to be explored and confirmed experimentally</u>.
>
> "Another aspect of the B&W-designed NSSS is its relatively rapid response to secondary side upsets, particularly in the feedwater system. Methods are being studied which could alleviate this behavior. * * * Presently, any design changes or modifications proposed to reduce this sensitivity can be justified by analysis only, without the benefit of experimental verification . . . . <u>An experimental facility would be of significant benefit in this area of concern</u>."

H. Denton, to R. Minogue, December 30, 1981, emphasis added.

> "<u>Thorough integrated system testing</u> under various system configurations and plant conditions as near as practicable to those for which the system is required to function during an accident <u>is essential for timely detection and correction of common mode design deficiencies</u>."

NUREG-1154, p. 8-3, emphasis added.

> "System transients and accidents in B&W reactors are now analyzed using the RELAP and TRAC [computer] codes <u>which are not configured for the once through steam generators used in B&W reactors</u>. Consequently, appropriate correction factors must be used. However, because of system volume and piping arrangement differences from the C.E. and Westinghouse designs for which TRAC and RELAP were configured, <u>safety analysis results [for B&W plants] are characterized with a greater degree of uncertainty than desired</u>."

V. Stello, to S. Chilk, Secretary, April 30, 1986, enclosure, p. 8, emphasis added. <u>See also</u>, Petition, pp. 36, 37.

The relative frankness and clarity of the above analyses of the uncertainties surrounding the behavior of B&W plants is in stark contrast with the testimony routinely presented by the NRC staff during the licensing hearings for those plants. In these hearings, where the licensing of the plants was at stake, the NRC staff routinely represented these analyses as conservative. It is therefore not surprising that the plants were licensed. However, since those earlier decisions were based on incomplete information, they cannot be used to bootstrap a continuing finding that the plants are safe now that the facts are known.

The B&WOG Claim That the B&W Design Is Not Less Safe Than Other PWRs Is Without Merit.

Shortly after NRC's January 24, 1986, announcement of the need to reassess the overall safety of B&W plants and the adequacy of NRC requirements applicable to the B&W design, "John MacMillian, Senior Vice President for Nuclear Power and Advanced Technology at Babcock & Wilcox, said that the company is 'very confident' that its reactors are 'perfectly safe' as designed." Wall Street Journal, February 19, 1986, as cited in NRC's response to Congressman Edward Markey, April 15, 1986, Question 20. In its response to this Petition, the B&WOG has picked up the melody and added a short refrain: "The B&W plants are safe." B&WOG Principal Response, p. 11.

The B&WOG devotes twenty pages of its response to an ultimately failing attempt to prove this proposition. See, Id., pp. 11 through 40. Two principal techniques are used. First, the B&W owners cite old reports without explaining why the cited portions of those reports should be considered valid (or why the changes made as a result of those reports should be considered adequate) considering the subsequent operating history of the B&W plants as detailed in the UCS petition.

Second, the B&WOG relies on reports that were either recently submitted to NRC (and apparently not yet reviewed by the agency) or were not even ready to be submitted to NRC:
1) "Currently, the NRC is reviewing SPIP [Safety and Performance Improvement Program], Rev. 03. The final conclusions and recommendations of the SPIP are expected to be available by mid-1987," and 2) a "Sensitivity study commissioned by the B&W Owners

Group" which "is in the process of being documented for submittal to the NRC." Id., p. 10, n. 8. While it may serve the B&WOG's public relations purposes to cite as "facts" (Id., p. 18) the conclusions of the next version of its reassessment program and some "sensitivity study" now being conducted, but not yet produced to NRC or UCS, these are nothing more than unsupported assertions and promises. Since the NRC staff did not base its March 13, 1987, response to this Petition on those documents, we do not reply to the B&WOG's unsupported assertions in this Supplement.

It is worth noting, however, the remarkable consistency of the B&WOG's response with the pattern described in the UCS Petition: study after study being used as a means to forestall effective action on the B&W design problems. The operating history of B&W-designed plants establishes the dangerous futility of hoping that the next study will make such action unnecessary.

The B&WOG also claims that "B&W plants contain a combination of other features not found in other pressurized water reactors which enhance their safety." B&WOG Principal Response, p. 13. While it is beyond dispute that the "features" of B&W plants are unique among pressurized water reactors, the four examples cited by the B&WOG do not support, as discussed below, the assertion that safety is thereby enhanced.

1. "High pressure emergency core cooling pumps which permit direct core cooling under the full range of operating conditions." Id.

   This is a simple description of so-called "feed and bleed" cooling, which is discussed at pages 33 through 35 of the Petition. The B&WOG points to recent tests and analyses which, it is once again claimed, "confirm" the adequacy of feed and bleed cooling. B&WOG Principal Response, p. 73.

The Appeal Board's decision in the TMI-1 restart case that "we are unprepared to state conclusively that feed and bleed will successfully provide core cooling . . ." (ALAB-729, 17 NRC at 852) followed a similar series of claims. First NRC and B&W claimed feed and bleed cooling was adequate and tests would be performed to confirm that conclusion. When these tests of feed and bleed were performed, they had to be prematurely terminated because there was a continuing net loss of water and the simulated reactor core was in danger of melting. Next, NRC claimed that the tests were not adequate to assess feed and bleed and should be disregarded. Finally, UCS was permitted to cross examine the NRC and B&W witnesses and the Appeal Board decision cited above was the result. Part of the relief requested by this Petition is an adjudicatory hearing; the reason for that request is that our experience has shown that even the most fervent assertions by NRC and the B&W owners do not always withstand the process of discovery and cross-examination.

Furthermore, even assuming, arguendo, that feed and bleed might be adequate in some plants under special circumstances, it possible and perhaps likely that the reactor operators will not use it because they are reluctant to contaminate the plant. This is, in fact, what happened during the Davis-Besse accident. See NUREG-1154, p. 8-2.

2. "Special features in the reactor internals to enhance core cooling (such as reactor vessel vent valves)." B&WOG Principal Response, p. 14.

The B&WOG does not specify what other "special features" beyond the internal vent valve it claims, but it is quite clear that the reactor vessel internal vent valve is an absolute necessity in a B&W-designed plant, not some "special feature" that enhances safety above that required. During a small break loss-of-coolant accident in which the break flow exceeds the high pressure emergency core cooling flow, "[w]ithout the aid of the vessel vent valves to pass steam into the upper downcomer annulus and hydraulically disconnect the vessel liquid from the steam generator liquid, a complete uncovering of the core for B&W plants would probably be predicted prior to loop seal clearing." H. Denton to R. Minogue, December 30, 1981, p. 3, emphasis added. While the addition of the vent valves is predicted to result in a B&W core remaining covered for a wider spectrum of break sizes than other designs, its inclusion is a necessity and the predicted partial uncovery of the core for a relatively short time in both the B&W and other designs is not calculated to cause a significant problem. The same could not be said for a B&W plant without internal vent valves.

3. "No potential for noncondensible gases to collect in steam generator U-tubes and interrupt natural circulation. (The U-bends in the two B&W hot legs have high-point vents.) B&WOG Principal Response, p. 14, citation omitted.

It is not clear what "special feature" which enhances safety the B&WOG has in mind. The reason there is no potential for gases to collect in the steam generator U-tubes is that B&W plants do not use U-tube steam generators. The fact that there are vent valves on the "candy cane" of the reactor coolant outlet piping is the result of a post-TMI requirement imposed by the NRC on B&W plants because of their unique piping configuration. Unless the B&WOG is arguing that the specific NRC regulation cited by the B&WOG

(stating that vent valves are <u>not</u> needed in U-tube steam generators) makes Westinghouse and C.E. plants dangerous, then this example is not a special feature enhancing the safety of B&W plants.

4. "Superior steam generator tube integrity." <u>Id</u>.

The B&WOG cites pages 2-2 and 5-18 of NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors," as the basis for its claim that this is an example of a special feature which enhances safety. However, an examination of the pages cited in this 1980 NRC report reveals that the steam generators used in B&W plants have a number of safety disadvantages that outweigh these and that most of their advantages are <u>economic</u> rather than <u>safety</u> advantages.

For example, the report states that "It is clear that the OTSG [Once-Through-Steam-Generator] is unique in terms of its capability to affect either rapid cooldown or heatup of the reactor coolant system. However, replacement of the OTSG does not appear to be practical or a necessary action for operating plants, especially when weighed against certain other safety advantages of the OTSG." NUREG-0667, p. 2. The report goes on to discuss safety disadvantages of the B&W design and its steam generators. The "general findings" are as follows: "(1) Confirmation that B&W-designed plants are more responsive to secondary side perturbations than other pressurized-water reactors; (2) The once-through steam generator design is technically sound; however, it requires a highly interactive and responsive control system (i.e., the integrated control system); (3) A high degree of overall plant interaction is inherent in the integrated control system and the once-through steam generator; (4) Based on the design features and the faster response of B&W plants during transients and upset conditions, the operators may be required to take more rapid action and have a better understanding of instrument response than operators on plants having other designs." Id., pp. 2-2, 2-3.

The "safety advantages" referred to by the NRC report and cited by the B&WOG are contained in a paragraph entitled "Operational Advantages Associated with the OTSG Design." The NRC designates the B&W steam generator's capability to produce superheated steam, which results in lower moisture in the main turbine "and longer turbine life, an obvious economic advantage." The NRC also notes that "this superheat produces a slight increase in plant efficiency, also a desirable economic advantage. Finally, the NRC states that "[o]perational experience has also indicated favorable tube integrity in the OTSG design compared to inverted U-tube design . . . ." Id., p. 5-18.

It should be noted that the NRC's statement about favorable steam generator tube integrity was written before essentially every tube in the TMI-1 steam generators was corroded and had to be either removed from service or repaired. But more importantly, it must also be recognized that whatever the degree of favorable operating experience with tube integrity in B&W steam generators, it is more than offset by the increased safety hazards of a steam generator tube rupture in B&W plants. Unlike the plants with U-tube steam generators, the emergency procedures for a steam generator tube rupture in B&W plants call for the deliberate release of radioactive steam outside the plant. The difficulties of dealing with this accident in a B&W plant are illustrated by the fact that the NRC has still not been able to determine whether the technical bases for the tube rupture emergency procedures in B&W plants is adequate.

In January, 1986, the B&WOG told NRC that its response to NRC's "request for additional information on the Steam Generator Tube Rupture Chapter of the B&W Owners Group Emergency Operating Procedures Technical Bases Document can be provided by June 27, 1986." M. A. Linn, B&WOG, to W. Paulson, NRC, January 7, 1986. However, in May, 1987, Three Mile Island Alert inquired whether this matter had been resolved and was informed that it was not. NRC was asked: "Have you or GPU [the operator of TMI-1] resolved with B&W that there is an adequate technical base for emergency procedures of steam generator tube rupture accidents?" The answer was: "No. Latest submittal and justification will be in SER due out by the end of the year." Memorandum of meeting between Eric Epstein, Three Mile Island Alert, and Rich Conte, NRC, May 14, 1987. In other words, the NRC is permitting the B&W plants to operate despite the fact that it is not known whether the emergency procedures for a steam generator tube rupture, a design basis accident, have a valid technical basis.

In sum, the B&WOG's claim that its plants are "safe" is based on "special features" that are either required to compensate for the safety hazards inherent in the unique B&W design or are, in reality, economic rather than safety advantages. The B&WOG also relies on the next version of its own reassessment program containing recommendations that have neither been reviewed by the NRC nor implemented in the B&W plants. Finally, the B&WOG relies on its own "sensitivity study" which was not submitted to the NRC and which, to the extent quoted in its response, appears to contradict numerous reports issued by the NRC over a period of years, many of which were prepared as a result of accidents in B&W plants.

CONCLUSION

For the above stated reasons and those contained in the original Petition, UCS urges the Commission to grant the relief requested.

ADDITIONAL PETITIONERS

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The number in the street address for Avery Schlagheck given in Supplement #3 of the Petition was incorrect. The correct address is: 4352 Lyman Rd.

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