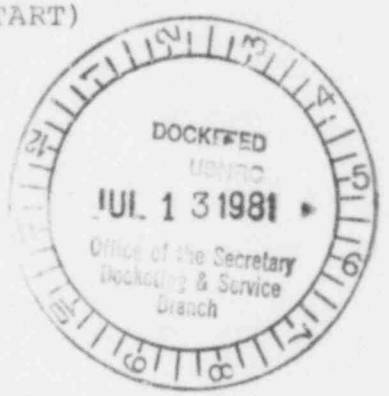
SHOLLY, 7/1/81

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(RESTART)
(Three Mile Island Nuclear Station, Unit No. 1)	)	



INTERVENOR STEVEN C. SHOLLY
REPLY TO PROPOSED FINDINGS OF FACT AND
CONCLUSIONS OF LAW ON PLANT DESIGN ISSUES

1 July 1981

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TABLE OF CONTENTS

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I. INTRODUCTION

1. Pursuant to a variety of Board Orders, parties to this proceeding were directed to file proposed findings of fact and conclusions of law and to file reply findings to the proposed findings of other parties. Intervenor Sholly herein submits his reply to the proposed findings of fact and conclusions of law filed by the NRC Staff, the Commonwealth of Pennsylvania, and the Licensee.[1]

2. Intervenor Sholly has endeavoured to utilize references to "Intervenor Steven C. Sholly Proposed Findings of Fact and Conclusions of Law on Plant Design Issues", dated 1 June 1981, were possible to avoid repetition. Where portions of that document are repeated, it is for the purpose of emphasizing a particular evidentiary point.

3. These reply findings are in the form of a point-by-point reply to the findings filed by the NRC Staff, the Commonwealth of Pennsylvania, and the Licensee in all cases

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[1] The Licensee did not file its conclusions of law when it filed its findings of fact on design issues. This may necessitate a reply to the conclusions of law when these are filed by the Licensee.

where, in the Intervenor's judgment, a response is necessary. As noted in Licensee's reply findings on management issues, the Board is not required to expressly treat in its decision each and every individual finding proposed by every party. Public Service Company of New Hampshire, et al. (Seabrook Station, Units 1 and 2), ALAB-422, 6 NRC 33, 41 (July 26, 1977). Intervenor Sholly interprets this decision to also mean that parties need not reply to each and every individual finding proposed by every party. Replies to findings are included herein only where there is substantial disagreement between Intervenor Sholly's proposed findings and those of another party.

II. REPLY TO PROPOSED FINDINGS ON THE INTEGRATED CONTROL SYSTEM

A. LICENSEE'S FINDINGS

4. Licensee asserts generally that the completion of the Integrated Control System Reliability Analysis (BAW-1564, August 1979) by Babcock and Wilcox, the submission of that report to the NRC Staff, and Licensee's reference of that report as being applicable to TMI-1 satisfies the literal language of Sholly Contention 6(a) (Licensee PF 178). Early in Mr. Sholly's cross-examination there arose a dispute among the parties as to the reach of Sholly Contention 6(a) (See generally Tr. 6960-61, 6968-70). In fact, it was clear to the Board long before this issue was litigated (i.e., when the prefiled written testimony

was filed on 15 September 1980) that both the Licensee and the NRC Staff had failed to fully respond to the issues raised by the role of the ICS in feedwater transients and other plant events. Indeed, the Licensee's prefiled testimony, in particular, is very sketchy, consisting of only 3 and one-quarter pages of text and a simple diagram of the ICS. The Licensee's written testimony generally describes the function of the ICS and summarizes the main conclusion of the B&W report (BAW-1564). (See Broughton, Sadauskas, and Joyner, ff. Tr. 6949, at 1-4 and Figure 1). As if to emphasize Licensee's lack of understanding of this issue, the summary statement of Licensee's written testimony states simply, "In summary with regard to Sholly Contention 6(a), an ICS FMEA has been completed." (Id. at 4).

5. It was at the time and continues to be inconceivable to the Board that Licensee seriously believed that the mere act of submission of a report to the NRC Staff satisfied the contention. Indeed, were this the case, the Board would have expected the Licensee (as a matter of courtesy to Mr. Sholly and to the Board) to air its view that it indeed believed the contention to have been satisfied, especially in view of the Licensee's oft-repeated desire to simplify the proceeding by reducing the number of issues to be litigated. Secondly, failing this, the Board would have expected the Licensee to have timely filed a motion for summary disposition pursuant to 10 CFR §2.749. Neither approach was followed by the Licensee. The Board notes as well that the Licensee conducted extremely little discovery

on this contention, thus essentially giving up the opportunity to further inform itself as to the basis for Mr. Sholly's contention. The Board also notes that despite the fact that the B&W reliability analysis (BAW-1564) was submitted to the NRC Staff shortly after the issuance of the Commission's August 9, 1979 Order and Notice of Hearing, Licensee failed to point this out to the parties during the Special Prehearing Conference at which contentions were discussed orally (i.e., the First Special Prehearing Conference, held in Harrisburg, Pennsylvania, on November 8, 9, 10, & 14, 1979.

6. The Board was struck by the lack of substance in Licensee's prefiled written testimony, especially since one of Licensee's witnesses, Dr. Joyner, was a co-author of BAW-1564. As Administrative Judge Jordan observed (Tr. 6950), the prefiled testimony goes into very little detail on the nature of the B&W study. As a result of the lack of detail in Licensee's prefiled testimony, the Board engaged in detailed cross-examination of Licensee's witnesses, particularly Dr. Joyner, in an effort to develop a record on the functions of the ICS and how B&W performed the ICS Reliability Analysis (See generally Tr. 6950-68, 6970-88).

7. A second surprise was in store for the Board when it became clear that Licensee did not intend to offer BAW-1564 into evidence to support its testimony. Licensee made the offer only after considerable prodding from the Board; indeed, the Board was compelled to note that Licensee was violating a basic evidentiary rule in relying extensively in its case

in chief on summarization of what is in an original document (See Tr. 6960-61, 6968-70). The Board noted later in the cross-examination that there had been considerable difficulty in trying to develop a record on this contention (Tr. 7002-03, Chairman Smith).

8. The Board concluded at the time, and reiterates its conclusion here, that the adequacy of the B&W report (BAW-1564) is subsumed by the contention and is within the scope of the contention as accepted by the Board for litigation. This should have been obvious to the parties, and the Board is at a loss to explain the apparent confusion on the part of the Licensee on this point.

9. Turning to substantive disagreements among the parties on this issue, the Board deals first with Licensee's consistent legal position that basically holds that what is good for other B&W licensees is good for this one. There is a measure of reasonableness in this position, as noted by the Commission in its Order CLI-81-3, in response to a motion from the Licensee to expedite this proceeding. The Commission, inter alia, ruled that TMI-1 should be treated in the same manner as operating reactors, but that the Board should find to the contrary when the record so dictates (Slip. op., at 7). Even in the absence of this guidance, however, the Board has an independent obligation to review the record on its merits and reach a decision based on this record. The Board cannot rely blindly on the findings of other Atomic Safety and Licensing Boards in other proceedings, as implicitly suggested

by the Licensee (Licensee PF 211, footnote 70) in its reference to the decision of the Board in the Rancho Seco proceeding. Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station) LBP-81-12, 13 NRC ___, slip op. at 19 (May 15, 1981).

10. Apparently, the Licensee would have the Board rely blindly on the adequacy of the short- and long-term actions proposed for other B&W licensees as a demonstration of the necessity and sufficiency of those actions in this proceeding. This position is indefensible; indeed, were this position to have been accepted, it is clear that this proceeding would never have taken place in the same form which has occurred.[2] It is a clear requirement of the Commission's Order and Notice of Hearing that the Board consider the necessity and sufficiency of the recommended actions set forth in the Order. Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1) CLI-79-8, 10 NRC 141, 148 (1979).

11. Beyond these considerations, however, the Board is compelled to point out that there is no automatic transfer

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[2] The Board would have expected, if Licensee's position were to prevail, that the Commission's Order for TMI-1 would have mirrored the Orders for other B&W plants, providing for a hearing only upon request, and, most significantly, providing that requests for hearing would not have stayed restart of the facility. This, of course, is not the case in this proceeding. The reasons for this are set forth clearly in the Commission's Order and Notice of Hearing. Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1) CLI-79-8, 10 NRC 141, 142-44 (1979).

of findings from one proceeding to another, even when the findings deal with the same subject matter. The Board must of necessity rely on the record developed in this proceeding in reaching its decision. Finally, it is worth noting that the Rancho Seco ICS is a different model than the ICS at TMI-1 (Sholly PF 3), and the Board has already taken note of the many differences between the Model 820 ICS at Rancho Seco and the Model 721 ICS at TMI-1, differences in both design and performance (Sholly PF 92-93). The fact that the measures proposed by the Director of Nuclear Reactor Regulation in the Commission's Order and Notice of Hearing are identical to proposed measures for other pressurized water reactors (PWR's) or for other Babcock & Wilcox plants does not guarantee that these measures are, of necessity, all that is required for TMI-1. Licensee's blind reliance on the similarity of proposed requirements between TMI-1 and other B&W plants is clearly misplaced and inappropriate in the light of the requirements of the Commission's Order. Licensee ignores the difference between legal precedent and factual record; the latter is clearly not transferable.

12. Licensee addresses the functions of the ICS and the major subsystems of the ICS (Licensee PF 179-80). The Board notes that this narrative is similar to that proposed by Intervenor Sholly (Sholly PF 24-26).

13. Licensee then addresses the bases for the inclusion of the ICS FMEA requirement in the Commission's August 9, 1979, Order and Notice of Hearing. This matter is

addressed by Mr. Sholly in more detail (Sholly PF 4-9). The Licensee, however, attempts to belittle the significance of the five concerns expressed by the Staff in the "NRR Status Report on Feedwater Transients in B&W Plants" (April 25, 1979), by stating that the information which served as the basis for these five concerns[3] was gathered in a short time span and was incomplete and, in some instances, incorrect (Licensee PF 181). Licensee clearly misreads the record in this instance. The actual testimony reads as follows (Tr. 15,862, Ross):

> "We were working in a very accelerated atmosphere and the numbers that we have in the mid-April 1979 time span were of necessity incomplete and in some instances incorrect. Now that is the statement with respect to the reliability of the ICS." (Board emphasis)

It is clear to the Board that the witness was referring in these comments only to the reliability of the ICS, which was the first of the five concern raised by the Staff, and not,

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[3] Those five concerns were:

(a) Was the reliability of the ICS satisfactory?
(b) The failure modes and effects of the ICS had not been systematically analyzed.
(c) The ICS may initiate 10-15% of all feedwater transients.
(d) The ICS controls emergency feedwater in some plants and could contribute to a total loss of feedwater.
(e) Even when the ICS works well, there may be, in response to a feedwater transient, wide swings in reactor pressure, pressurizer level, and average reactor coolant temperature.

(Ross and Capra, ff. Tr. 15,855, at 2)

as urged by the Licensee, to the remaining four concerns raised by the Staff. In fact, it is quite clear that the Staff believed the other four concerns were well-founded. Licensee cites no other authority for concluding otherwise, and the Board is itself aware of no evidence which would support Licensee's hypothesis.

14. Licensee then moves to the the fourth concern expressed by the Staff in the "Status Report", namely, that since the ICS controls emergency feedwater it could contribute to a total loss of feedwater (Licensee PF 182). The Director of Nuclear Reactor Regulation recommended that, as a short-term action, Licensee be required to develop and implement procedures for initiating and controlling emergency feedwater (EFW) independent of the Integrated Control System. Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1) CLI-79-8, 10 NRC 141, 144 (1979). As noted by the Licensee, pursuant to this requirement, changes have been made (Licensee PF 182). Licensee relies, as does the NRC Staff (Ross and Capra, ff. Tr. 15,855, at 6; NRC Staff Ex. 1, at C1-1, C1-11), to satisfy the short-term requirement of the Commission's August 9, 1979 Order. As discussed previously (Sholly PF 96), the provision of automatic start signals for the motor-driven EFW pumps on loss of main feedwater and the provision of procedural capability to control EFW independent of the ICS do not fully alleviate the concern about this matter. The Board agrees that some degree of improvement will result from these changes, but neither the NRC Staff nor the Licensee has been able to quantify the degree of improvement. The Board recognizes that until the long-term upgrade of

EFW control is fully implemented[4], Licensee must of necessity rely on the ability of plant operators to recognize ICS failures as the possible cause of loss of EFW before the new procedures which permit control of EFW independent of the ICS are of any benefit. As observed previously (Sholly PF 96), the Board is sensitive to the fact that many ICS failures are not annunciated to plant operators (Sholly PF 51-56). Indeed, since the ICS is not a safety-grade system (Sholly PF 1), it is susceptible to multiple failures as well (Sholly PF 24, 50, and 86). The fact that the ICS FMEA did not consider off-normal initiating conditions leads to additional doubts (Sholly PF 47-48). This is particularly distressing since the ICS has as a basic purpose "to assist in increasing the unit's generating capacity by preventing reactor trips for many anticipated plant upsets (i.e., load changes, loss of a single reactor coolant pump, etc)." (Licensee PF 179) Despite the fact that a basic ICS design goal is to prevent trips under off-normal initiating conditions, the ICS FMEA did not consider such starting conditions in analyzing the effects of ICS failures (Sholly PF 47-48; Sholly Ex. 2, at 10-11; Tr. 15,896, Capra). (See

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[4] The long-term upgrade of EFW control is for a fully safety-grade system which is completely independent of the ICS for initiation and control of EFW following a loss of main feedwater event. (Licensee PF 182)

Licensee Ex. 18, at 4-5, for off-normal conditions for which the ICS is designed). As noted previously (Sholly PF 102), the Board has concluded that for the long-term, this issue is resolved by the safety-grade system to control EFW independently of the ICS. However, for the short-term, the Board is concerned about the ability of the operators to promptly recognize the existence of ICS failures and promptly take over EFW control from the ICS, and as a result the Board cannot conclude that the issue is resolved for the short-term. Further, neither the Staff nor the Licensee addressed the reliability of the manual EFW control process, including the reliability of the power supply required for manual control, and therefore the Board is in no position to make a judgment as to the reliability of the short-term action as a substitute for ICS control of EFW.

15. Licensee then moves on to a very brief explanation of the genesis of the requirement for the submission of the ICS FMEA (Licensee PF 183). Therein, Licensee notes the agreement of the Staff witness (Mr. Thatcher) and B&W on the definition of the ICS boundary. Licensee fails to note the strong disagreement of ORNL on this matter, which was also acknowledged by Staff witness Thatcher on the same transcript page. Nowhere in its findings does Licensee acknowledge this disagreement. As noted earlier (Sholly PF 27-28), ORNL recommended a much broader definition of the ICS boundary based on the very nature of the control system itself. Due to the impact of the definition of the ICS boundary on the framework within which ICS failures are viewed (See Sholly

PF 30-42), Licensee's failure to treat this disagreement, especially considering the recognized competance of ORNL regarding the ICS (See Sholly PF 13; Tr. 7257-58, Thatcher; Tr. 15,869, Ross), is incomprehensible to this Board. The Licensee is not alone in ignoring this disagreement, since the Staff failed to treat this matter in its findings as well. Licensee's witness, Mr. Joyner (a co-author of the B&W report, BAW-1564), attempted to address this disagreement on the definition of the ICS boundary by stating that the ICS is "a well-defined piece of equipment that is sold with the B&W product. It has a well-defined function and is well understood by us." (Tr. 6951, Joyner) It does not follow perforce that this makes B&W's definition correct. It is apparent to the Board that B&W, as well as the Staff's witness, Mr. Thatcher, based their definition on a physical basis, while ORNL based its definition on a functional basis (See Sholly PF 28). In view of ORNL's observation that the system being controlled by the ICS "includes a number of process loops that are highly interactive and which must often operate within rather narrow individual constraints" (Sholly PF 28; Sholly Ex. 2, at 6), the Board finds ORNL's definition far more appropriate than the limited definition offered by B&W. As explained previously (Sholly PF 28-42), acceptance of ORNL's definition of the ICS boundary results in a view of ICS failures that is considerably different from that proposed by the NRC Staff and the Licensee.

16. Licensee moves on in Proposed Finding 185 to assert once again that the most drastic transient obtains from analyzing only "high-scale" and "low-scale" input and output failures. As the Board has earlier observed, such assumptions do not necessarily lead to the most drastic transient (Sholly PF 53) according to NRC Staff testimony. Further, since mid-scale failures may remain undetected for some time after their occurrence, failure to detect mid-scale failures can lead ultimately to multiple failure events (Sholly PF 54; Sholly Ex. 2, at 8). ORNL found that mid-scale failures are highly credible events based on its review of operating experience (Sholly PF 53; Sholly Ex. 2, at 21). Despite B&W's insistence that "high" and "low" failures would produce the most drastic response, ORNL could find no specific evidence to support this assumption (Sholly PF 53).

17. Licensee also relied on the "high" and "low" scale assumption in its findings on the effects of failures of ICS internal modules (Licensee PF 186). In the Board's view, this suffers from the same weakness as discussed in Paragraph 16, supra.

18. Licensee moves on to briefly discuss the computer simulation (POWER TRAIN IV, or PT-IV for short) that was used by B&W in analyzing the effects of failures on the NSSS (Licensee PF 187). Licensee fails to address several significant weaknesses in the PT-IV simulation that were pointed out both by ORNL and B&W itself in BAW-1564

(Sholly PF 45-46, 59-63). These weaknesses are important since the computer simulation was utilized by B&W in about 75% of all cases to evaluate the effect of failures on the NSSS (Sholly PF 46; Sholly Ex. 2, at 22).

19. As Licensee's witness Joyner testified, failures involving the ICS are "very dependent" on the time in core life at which the failure occurs, the initial power level at which the failure occurs, the response of teh plant operators to the failures, and other unspecified factors (Sholly PF 45; Tr. 6967, Joyner). A basic assumption made in the PT-IV computer simulation is that the reactor core is at its mid-point in core life for all evaluations (Sholly PF 46; Licensee Ex. 18, at 4-2). Other weaknesses, discussed in detail previously (Sholly PF 59-63) are that the computer simulation is based on the Rancho Seco plant which has a different ICS Model than does TMI-1, the computer models the ICS (which is an analog system) as a digital system based on functional blocks (See Sholly PF 57-58 for a discussion of the problems with the functional block approach), temperature and pressure limitations on various functions related to the primary, secondary, and feedwater systems, the lack of detail in the computer modelling of the feedwater system, and the fact that the computer model is not valid at low power.

20. Licensee's findings (Licensee PF 187) also expresses the opinion, unsupported by citation to expert testimony or other evidence, that the proportion of postulated

failures identified as "Category 3"[5] failures is "small." Licensee PF 187) Licensee cites the number of failures in Category 3 as being 15 out of 115 possible failures. Even if the Board accepts Licensee's tabulation as correct (which we do not; See below), this represents 13% of all postulated ICS failures. The Board, in the exercise of its independent judgment, does not regard this as a "small" proportion. Furthermore, the Board does not regard Licensee's tabulation as being accurate. A brief examination of Table 4-6 (Licensee Ex. 18, at 4-61 to 4-64) reveals that there are many more failures than are accounted for in Licensee's tabulation. The Board notes that this is so because quite a number of the 115 failures tabulated by the Licensee have two or more separate functional blocks, inputs, or outputs whose functions are the same, but which apply to multiple components in the plant (such as four reactor coolant pumps, listed on page 4-61 as Output numbers 3-38 through 3-41). Since each of these blocks, inputs, and outputs can fail individually, the Board believes they should be regarded individually. Consequently, the Board arrives at 28 out of 179 failures at being in Category 3, or roughly 15.6%,

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5 "Category 3" failures, as defined by B&W, are those failures that might cause a reactor trip, and following the trip could require HPI or EFW to control the effect of the failure unless the operator intervenes. (Licensee PF 187)

again, not an insignificant proportion in the Board's view (looked at differently, this means that one out of six to one out of seven failures that were evaluated by B&W result in consequences which place them in Category 3 in terms of impact on the plant).

21. Licensee also cites a conclusion by witness Joyner that no failures were identified that affected the operation of plant safety systems (Licensee PF 187). Given the lack of the scope of failures which B&W included within its evaluation (Sholly PF 1, 24, 27-28, 46-68, 86-88, and 95), this conclusion is hardly surprising. Similar to an earlier Board finding (Sholly PF 87), at best B&W and the Licensee have demonstrated that for failures as they were defined and limited in B&W's study, no failures were identified that affected the operation of plant safety systems. Given the numerous cited weaknesses and limitations on B&W's study, the Board regards this a very minimal demonstration which is not greatly significant. The Board agrees with ORNL's analysis--B&W simply did not go far enough in its consideration of failures, and therefore the results of B&W's analysis are necessarily of limited value (Sholly PF 87; Sholly Ex. 2, at 4). The Board notes that Mr. Joyner's conclusion tells us absolutely nothing about the potential failures that were not studied.

22. In Proposed Finding 188, Licensee briefly highlights some of the results of the B&W review of operating experience. In doing so, Licensee trots out the same tendentitous

statistic upon which both the Staff and the Licensee have "hung" themselves--namely that only 6 out of 310 ICS internal component hardware failures have caused reactor trips. Later (in Licensee PF 189) Licensee makes this statistic even more misleading by stating, "ICS hardware performance has not led to a significant number of reactor trips (6 trips out of a total of 310 analyzed)". In the first instance, the use of this statistic is misleading, while in the second instance it is simply dead wrong. The operating statistics have been discussed at some length previously (Sholly PF 31-42), but the Board will repeat some of those statistics here in order to make the distinction between Licensee's use of the statistics and the manner in which the Board views those same statistics. In the first instance above, it is literally correct that failure of hardware internal to the ICS cabinets themselves has caused only 6 trips. This is certainly not the only involvement of the ICS in reactor trips, however, and that is what makes the second use of the statistic as cited above wrong, rather than simply misleading. In examining the history of operation of ICS systems, the B&W analysis tabulated 162 "findings" related to ICS hardware; 47 failures were noted, 71 instances of calibration required, and 44 instances of tuning required (Licensee Ex. 18, Table 5-8 at 5-14). Given that 6 trips were caused by ICS failures, this gives 6 out of 47 failures leading to trips, or 12.8%.

23. Furthermore, there are other relevant statistics, as the Board has already discussed (Sholly PF 31-42). In summary form they are:

a. The 71 instances of calibration problems mentioned in Paragraph 22, supra, may have additional significance since instrument drift not normally associated with a failure might be sufficient to initiate an ICS-induced trip (Sholly PF 31; Sholly Ex. 2, at 5).

b. Power supply failures to the ICS have caused 11 additional trips. This does not include an additional 6 trips due to human actions involved with ICS power supplies (Sholly PF 33).

c. Power supply failures have led not only to reactor trips, but to overcooling incidents as well (Sholly PF 34).

d. Failures of ICS inputs other than power supply have caused 11 trips, five from loss of RC flow signals, three from loss of RCS temperature signals, two from loss of neutron flux signals, and one from loss of feedwater flow signals (Sholly PF 37).

e. ICS participation in or cause of feedwater oscillations have caused 11 trips, and have also resulted in actuations of ESF and loss of main feedwater (Sholly PF 38).

f. The control response of the ICS has caused 16 trips; twelve were caused by feedwater/power mismatches and four were caused during switching modes of ICS control (Sholly PF 40).

g. Operator error could have caused additional trips resulting from switching control modes of the ICS, misunderstanding an instruction, or misunderstanding a procedure, although no statistics are available on these trips, but a bounding number of 35 additional trips has been fixed from such causes (Sholly PF 41-42).

24. When the Board examines these statistics in the light of ORNL's suggested definition of the ICS boundary (See Paragraphs 15 and 23, supra), if ICS internal failures, ICS control response, and ICS input failures are considered (the latter is reasonable since it is the control response of the ICS to the failure that causes the trip), the ICS is found to have caused a total of 56 trips out of the 310 studied by B&W, slightly over 18% of all trips. It should be understood that the total of 56 trips does not include any trips that may have been caused by operator error in taking over manual control of the ICS, nor does it include any trips which may have resulted from operator error in misunderstanding an instruction or procedure related to the ICS, even though the Staff has testified that additional trips may indeed have resulted from such causes. Therefore, the Board regards the total of 56 trips caused by the ICS as a minimum bounding number, not the maximum possible (Sholly PF 89).

25. In addition, as noted previously (Sholly PF 90), in addition to causing reactor trips, ICS power failures have led to six overcooling transients in excess of permissible cooldown limits and four actuations of HPI.

26. Licensee next moves to its response to the recommendations made by B&W as a result of the FMEA (Licensee PF 190-96). Licensee's response to these recommendations is fairly straightforward, and the Board will not deal at length with these items, other than to make a few pertinent observations as follows:

a. Licensee references the ATOG program is providing procedural guidance to operators on dealing with stuck open main feedwater startup and turbine bypass valves (Licensee PF 196). Since the ATOG program for TMI-1 will not be completed, according to Licensee's estimates, until September 1981, and is not actually required under the provisions of NUREG-0737 until January 1, 1982, and since the Board has obviously not seen the ATOG program for TMI-1, the Board is in no position to judge the acceptability of its results in terms of providing assurance that these procedures will alleviate the concerns associated with stuck open main feedwater startup and turbine bypass valves.

b. As noted earlier (Sholly PF 101), neither the Staff nor the Licensee has been able to quantify the degree of improvement that will result from ICS/NNI power supply modifications. As a result, the Board knows neither where we started nor where we ended in terms of the reliability of ICS/NNI power supplies. As a consequence, the Board cannot reach a determination as to the sufficiency of the modifications; this is significant since by Licensee's admission loss of ICS/NNI power can result in total loss of feedwater (Licensee PF 189).

c. Finally, given the lack of scope of the B&W analysis which the Board has previously addressed, and given the numerous recommendations to which neither the Staff nor the Licensee have addressed themselves (Sholly PF 105-106), the Board is left with signficant questions about whether enough has been done regarding the ICS. As a result, the Board has proposed additional measures (Sholly PF 169, a through f, and 171).

27. Licensee moves to a discussion of the original five concerns which led the NRC Staff to require the performance of the FMEA on the ICS (Licensee PF 198, 207-210) and some of the criticisms set forth by ORNL of that FMEA (Licensee PF 199-206). Licensee begins this process with the observation that the ICS

performed as designed during the TMI-2 accident (Licensee PF 198). Licensee makes this observation without making a connection between it and any other fact or consideration. Given this naked observation by the Licensee, the Board wonders what the Licensee would have us do with it. The significance of this point is, however, addressed earlier by the Board (Sholly PF 4-5). Licensee proposed no conclusion to be drawn from this observation, and the Board draws none.

28. Licensee then moves to the conclusions drawn by ORNL and agreed to by the Staff (Licensee PF 199). It is clear to the Board that these conclusions[6] are not as unequivocal as Licensee would have the parties believe; furthermore, the Staff concurred with only one of the cited conclusions. The Board will address each conclusion asserted by the Licensee individually.

29. First, Licensee cites the conclusion that the ICS itself has a low failure rate and does not instigate a significant number of plant upsets (Licensee PF 199). Initially the Board notes that this is the only one of the four cited

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[6] The conclusions cited by the Licensee are: the ICS itself has a low failure rate and does not instigate a significant number of plant upsets; failures of and within the ICS are adequately mitigated by the RPS; many potential ICS failures would be mitigated by the cross-checking features of the system without challenging the RPS; and, that the ICS is failure tolerant to a significant degree (Licensee PF 199).

conclusions with which the Staff concurred. In neither of the Licensee's citations to the record does the Staff even mention the other conclusions, much less indicate the Staff's concurrence with them (Compare Licensee PF 199 with Thatcher, ff. Tr. 7122, at 6, and Ross and Capra, ff. Tr. 15,855, at 3). It is the citation to Sholly Ex. 2 at 14-15 which contains all four conclusions. The conclusion that the ICS has a low failure rate and does not instigate a significant number of plant upsets is rather sweeping when the statistics themselves are examined. In the exercise of its judgment, the Board has done so. It is apparent from examining the statistics in Table 5-9 of BAW-1564 (Licensee Ex. 18, at 5-14) and the text explaining these statistics (Licensee Ex. 18, at 5-8 to 5-10) that the failure rates of the Model 820 and Model 721 ICS differ dramatically. B&W places the mean time between failure (MTBF) for the 721 ICS (the model in place at TMI-1) at between 2754 hours and 3660 hours, whereas the MTBF for the Model 820 ICS (which is used at Rancho Seco, the plant upon which the FMEA was conducted; Licensee Ex. 18, at 4-1) is between 33,000 and 49,000 hours (Sholly PF 93). This represents a difference of more than an order of magnitude in the failure rates between the two ICS models. The failure of ORNL and all of the Staff and Licensee witnesses to point this out is inexplicable to the Board. Basing a comparison on the failure rate per calendar hours, these MTBF's would lead to about 2 failures per year for the Model 721 ICS, whereas the 820 ICS

would be expected, based on these statistics, to experience a failure about once every 8 years (this comparison, as noted, is based on a calendar hour figure, assuming roughly 6,000 calendar hours per year of operation--about a 70% capacity factor; using a different capacity factor or basing the comparison on reactor hours rather than calendar hours would not substantially alter the disparity in failure rates). The Board considers the difference in failure rates between the Model 721 ICS and the Model 820 ICS to be quite substantial. The reasons for the large difference in failure rates has been previously addressed (Sholly PF 92-93). B&W concluded as well that the Model 820 ICS has improved reliability when compared with the Model 721 ICS (Sholly PF 92).

30. The second conclusion cited by the Licensee, that failures of and within the ICS are adequately mitigated by the RPS, is also a rather sweeping statement, especially in the manner in which Licensee has subtly altered the actual language used by ORNL. Licensee's finding states, "ORNL has made the following conclusions, in which the Staff has concurred, on the basis of its own review: . . . failures of and within the ICS are adequately mitigated by the RPS . . ." (Licensee PF 199). The ORNL review stated, however (emphasis added), "The analysis further shows that anticipated failures of and within the ICS are adequately mitigated by the PPS . . ." (Sholly Ex. 2, at 14-15). The clear reference in the ORNL language is to anticipated failures, and it is clear to the Board that "anticipated" refers to those failures which were antici-

pated by the B&W analysis. As ORNL made abundantly clear, and as the Board has previously observed (Sholly PF 1, 24, 27-28, 46-68, 86-88, and 95), the scoep of the "anticipated" failures in the B&W analysis was very limited. Therefore, while it may be true that for the failures actually considered the B&W analysis showed that the RPS adequately mitigates such failures, there is no basis whatsoever upon which to conclude that all other failures will also fall within this same category. Indeed, to the contrary, ORNL concluded that the B&W analysis does not answer in a meaningful way the question of whether the ICS can cause the plant to malfunction in a credible way so that the protective systems cannot handle the problem (Sholly PF 88).

31. Licensee also points to the conclusion that the ICS is failure tolerant to a significant degree (Licensee PF 199). ORNL also concluded, however, that the cross-limiting features of the ICS, while useful, are not infallible (Sholly PF 95). Further, there is no evidence in the record which suggests that limiting devices are more or less reliable than any other component of the ICS (Sholly PF 95). Indeed, ORNL observed that signal limiters and auctioneers can fail without those failure being annunciated; ORNL pointed in particular to the example of a failed auctioneer which might have no effect on ICS performance until called upon to implement a cross-limit required by another ICS failure (Sholly PF 51; Sholly Ex. 2, at 8).

32. Licensee also cited an ORNL conclusion that the

the ICS prevents or mitigates many more upsets than it creates (Licensee PF 199). Licensee neglects other statements by ORNL on this issue wherein ORNL concluded that this assertion is not pertinent and is probably true, but that the data presented by B&W in BAW-1564 do not substantiate this assertion (Compare Sholly Ex. 2, at 11 with Sholly Ex. 2, at 15). The Board agrees with ORNL's observation that this statement is not pertinent, since the issue is not necessarily how often the ICS creates plant upsets (although this is clearly part of the issue) so much as the issue is what happens when the ICS does cause such upsets.

33. The Board next deals with Licensee's treatment of the ORNL criticisms of the B&W report (Licensee PF 200-206). Licensee begins by asserting generally that the Staff failed to give ORNL appropriate guidance in evaluating BAW-1564 (Licensee PF 200). The Board recognizes that ORNL drew significantly from the recommendations and comments made by the Staff in NUREG-0560 (Sholly PF 11). However, it is also apparent to the Board that ORNL did not rely solely on NUREG-0560. As the Staff itself noted, ORNL had, previous to its review of BAW-1564, reviewed the ICS in other work which ORNL had done (Sholly PF 13). Further, it is clear that in addition to addressing the issues raised in NUREG-0560, ORNL judged the methodology and the results of the FMEA and the operating history review in BAW-1564 on their merits. Any assertion that ORNL relied exclusively on NUREG-0560 in its review of BAW-1564 is without a basis.

Moreover, the Board notes that even if it were the case that ORNL had relied exclusively on NUREG-0560, it is clear that ORNL has addressed each of the five issues expressed by the Staff as being the basis for its concern about the ICS (Sholly PF 6, 99-104). While ORNL may have reviewed the B&W report from a broader perspective, this does not mean (and in fact the record demonstrates to the contrary) that the observations and recommendations made by ORNL are somehow invalid. As the Board earlier observed, the B&W report and the ORNL review of that report raised far more questions than they answered (Sholly PF 104). Furthermore, beyond any other standards that may be applied to the B&W report, the Board observes that it should of necessity be able to stand on its own merits as a technical report; it is clear from ORNL's criticisms of BAW-1564 that it fails to do so.

34. Licensee next addresses ORNL's criticism of B&W's treatment of multiple failures (Licensee PF 202). Licensee, in so doing, makes the bald statement, unaccompanied by any evidence whatsoever other than a conclusionary statement made by witness Joyner, that multiple failures are considered in plant safety analyses and, further, that the effects of multiple failures would be bounded by the events analyzed in the FSAR.[7] The Board is unaware of any instance involving an FSAR that involves multiple failures, and Licensee has given the Board not a single example to support its testimony. Furthermore, the Board is itself aware of at least two instances of multiple

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[7] The witness acknowledged, however, that these analyses do not assume improper operation of non-safety systems (Tr. 7045, Joyner).

failures which actually involve the ICS itself, and which involve an event which is outside the TMI-1 design basis. The event is total loss of feedwater, which can result either from a chain of failures evolving from an NNI/ICS power failure (Licensee PF 189; Sholly PF 56) or from a steam generator isolation on low pressure (caused by the steam line rupture detection system) followed by a single ICS failure causing loss of feedwater flow to the remaining steam generator (Sholly PF 56; Tr. 5730-31, Lanese).

35. Further, NRC Staff witness Thatcher addressed the matter of how multiple failures are addressed in an FSAR. Based on his familiarity with the contents of an FSAR on the ICS and his secondary review responsibilities with respect to the FSAR, Mr. Thatcher testified that Chapter 7.7 of the FSAR would cover the ICS generally, and Chapter 15 would have to include a determination that the accident analyses therein did not take credit for the performance of the ICS (Tr. 7231, Thatcher). Mr Thatcher testified subsequently that not taking credit for the performance of the ICS is not the same as assuming that the ICS (which is not a safety-grade system) fails in the worst possible manner, and he also testified that he would not expect to see an analysis of such failures in an FSAR (Tr. 7232, Thatcher). The witness also testified the Staff, in analyzing a new FSAR for a B&W plant, would not extend its review to include multiple failures, with the possible exception of so-called "cascaded"

failures, and that he would not expect that prior FSAR analyses would have included consideration of multiple ICS failures (Tr. 7235-36). Finally, Mr. Thatcher testified that he could not identify any reports, studies, or other type of analyses concerned with multiple failures in the ICS and their impact on the plant (Tr. 7240, Thatcher).

36. Mr. Thatcher expressed the view held by Licensee witness Joyner that the Chapter 15 analysis would bound any event that could be caused by the ICS (Licensee PF 202; Tr. 7236, Thatcher). The Board cannot discern any basis for this view because the Staff explicitly limits its analysis to single failures, and could not identify any other type of analysis of the ICS which dealt with multiple failures. This view is apparently a "belief" which has no substantive basis, and the Board, therefore, accords it no evidentiary weight. There is no evidence in the record to support such a proposition, and, other than the naked conclusion, Licensee cited no such evidence.

37. Licensee also relies (in Licensee PF 202) on the ATOG program to develop procedures to allow plant operators to assess the significance of multiple ICS failures. The Board has been presented with no evidence that suggests, however, that ICS multiple failures will be specifically addressed in the ATOG program.

38. Licensee concludes (Licensee PF 202) that further analysis of the ICS to assess multiple failures may not be economically justifiable, and expresses the view that failures

within the ICS do not constitute a significant threat to plant safety. In view of the lack of scope of the B&W analysis (Sholly PF 1, 24, 27-28, 46-68, 86-88, and 95), the Board cannot reach the same conclusions. Further, the Board believes that the Licensee is misinterpreting ORNL's comments; the Board understands ORNL to be suggesting only that further studies of the type done by B&W might not be economically justifiable. This same view was expressed by Staff witness Thatcher, who testified that the comments by ORNL were related only to studies of single failures of ICS internal components, and not to additional studies of interactions of ICS with actuated components and other systems (Tr. 7275-76, Thatcher).

39. Licensee next addresses mid-scale failures (Licensee PF 203) and repeats previous testimony that the most severe plant response is obtained from "high" and "low" scale failures. The Board has addressed this issue at Paragraphs 16-17, supra, as well as in the decision itself (Sholly PF 53-54). Contrary to Licensee's finding (Licensee PF 203), the Board finds no merit to the suggestion that ORNL had concluded that mid-scale failures are mot likely to result in mid-scale failures; indeed, ORNL addresses multiple-input signal failures, not mid-scale failures (Sholly Ex. 2, at 20-21). Moreover, there is no evidence to suggest that if Licensee's finding were correct, such power failures represent the only cause of mid-scale failures. Studies of power failures only, therefore, would not eliminate mid-scale failures as a problem even if Licensee were correct.

40. Further, contrary to Licensee's citation to Mr. Joyner's testimony, Mr. Joyner said nothing about whether Licensee's studies will specifically include mid-scale failures. Moreover, if these studies exist, the Board has not been presented with them, nor has the Licensee attempted to introduce them into evidence. As a result, the Board can reach no determination as to the usefulness of these studies in resolving the mid-scale failure problem. The Board cannot depend in its decision on the results of undefined future studies which do not specifically address the issues at hand.

41. Licensee moves on to address the issue of B&W's failure to consider the effects of ICS failures on related systems (Licensee PF 204). Therein, Licensee draws the unfounded conclusion that ORNL's criticism in this regard resulted from ORNL's use of NUREG-0560. It is clear to the Board, however, that ORNL's criticism flows directly from ORNL's disagreement with the definition of the ICS boundary (Sholly PF 27-28; Sholly Ex. 2, at 6). Additionally, there is no citation presented by the Licensee to support its hypothesis (presented for the first time in Licensee's findings) that the ICS FMEA considered failures of related systems in the analysis of failures of ICS inputs and outputs. Indeed, it is the Board's observation that it is quite a different matter to assume an ICS output fails, than assume that the output is retained but that the actuated equipment itself fails. The Board can discern no basis for Licensee's conclusion, and Licensee cites none.

42. Licensee again misinterprets ORNL's statements on the economic justifiability of additional studies (Licensee PF 205) to justify its elimination of consideration of additional studies using the equipment block diagram technique, rather than the functional block technique used by B&W. As the Board observed at Paragraph 38, supra, this is an incorrect inference.

43. Lastly, Licensee postulates that the Staff considered all of the concerns expressed by ORNL (Licensee PF 206). The Board does not believe this to be true, and could find no evidence that the Staff has done so. If it is true that the Staff has considered the ORNL criticisms, it is clear from the Staff's recommendations that the Staff simply did not take ORNL's work very seriously. The Board has closely examined ORNL's views on the adequacy of the B&W ICS study, and has made additional recommendations as set forth earlier (Sholly PF 169 a-f, and 171). In addition, it is quite clear to the Board that neither the Staff nor the Licensee have ever responded to a number of recommendations made by ORNL for additional work that should be undertaken (See Sholly PF 105-106).

44. The Licensee moves on in the final portion of its proposed findings to address the five issues which gave rise to the ICS FMEA (See Paragraph 13, supra, footnote 3). Licensee's conclusion that the FMEA served the purposes for which it was performed mirror the conclusions of Staff witnesses Ross and Capra (Licensee PF 204-210). The Board

has addressed these issues at length (Sholly PF 99-104) and will not belabor them here with a few exceptions. The Board notes that the Licensee compares feedwater transients and reactor trips (Licensee PF 209). The Board does not view the comparison as accurate; the Board has been told nothing with regards to the accuracy of the Staff's original observation that the ICS may initiate 10-15% of all feedwater transients. Indeed, as noted earlier (Sholly PF 101), ICS/NNI power failures alone have been found by the Staff to have caused 18% of all feedwater transients.

45. In addition, the Board can find no support for Licensee's proposition that ORNL was suggesting that a complete systems interaction analysis be performed (Licensee PF 204). ORNL was pointing out only that B&W's analysis refused to consider this issue at all, and that some consideration was necessary considering ORNL's definition of the ICS boundary.

46. Regarding Licensee's citation to Staff testimony that there is no evidence that the ICS causes more frequent or more severe challenges to the protection system than other control systems, the Board has previously addressed itself to this matter (Sholly PF 99). It is fundamental that if the PORV is [illegible] challenged less frequently, the RPS will be challenged more frequently. Further, the Board was presented with not one shred of evidence about the challenge rate for safety systems caused by other designs for control systems. The Staff's comparison, therefore, is an empty one absent some

evidence about other vendors' plant control systems.

B. NRC STAFF'S FINDINGS

47. The Board notes initially, before addressing directly the Staff's findings, that the Staff's findings on this matter are very brief, and in the Board's view, superficial. The Staff ignores, for all intents and purposes, all of the disagreements between ORNL and B&W, attempting to sweep them away by referring to a few limited cites in the record. It is obvious to the Board that the Staff does not take this issue seriously, and the Board is at a loss to understand why.

48. Substantively, the Staff took the same initial position, namely that the submission of the FMEA satisfied this contention. The Board dealt with this absurb postulate at Paragraphs 4-5, and 8, supra. (See NRC Staff PF 216).

49. The Staff then moves on to attempt to sweep under a rug any hint of disagreement between ORNL's criticisms and the findings of the Staff (NRC Staff PF 222). Neither the Licensee's witness, Mr. Joyner, nor the Staff witness, Mr. Thatcher, testified in detail that the B&W report was adequate, rather they simply concluded that this was so. Moreover, Mr. Thatcher was quite specific in his answer, and would only state that the B&W report met the requirement to submit an FMEA as stated in the Commission's Order (Tr. 7298, Thatcher). Both witnesses concluded that the "bottom line" as section 6 of the ORNL

review, rather than the first five sections of the ORNL report which all witnesses acknowledged was critical of the B&W report (Tr. 7045, Joyner; Tr. 7126-27, Thatcher). Beyond these conclusionary statements, the Board itself must independently determine whether the report submitted by B&W adequately addressed the concerns expressed by the Staff, and we have done so at length (Sholly PF generally, specifically 85-106).

50. As the Staff has noted in its findings, the Board, after considering a request from Mr. Sholly to produce the ORNL authors, gave Mr. Sholly the opportunity to pose additional interrogatories to be answered by the authors of the ORNL review report (See Tr. 7335-41). Shortly thereafter, Mr. Sholly, in a memorandum to the Board and parties, notified the Board that he was withdrawing the remainder of his contentions and that his participation in the remainder of the proceeding would be limited due to his employment with the Union of Concerned Scientists. As a result, Mr. Sholly did not have the opportunity to complete his efforts at inquiring into the ORNL authors' criticisms in the form of interrogatories, although Mr. Sholly did return and participate in cross-examination of NRC Staff witnesses Ross and Capra whose testimony was intended to supplement Mr. Thatcher's.

51. The Staff's findings then briefly summarize the conclusions of the Ross and Capra testimony (NRC Staff PF 224). The Board has addressed these concerns previously (Sholly PF 94-104). Moreover, the Board notes that the Staff's supplemental testimony provided no additional information on the extent to

which the Staff relied on the ORNL review, whether that reliance was reasonable, what exactly the ORNL review team did in performing their review (See Tr. 7335-41). In large measure, other than clarifying the process by which the Staff came to be concerned about the ICS, the supplemental testimony of Ross and Capra merely reiterated prior testimony by Mr. Thatcher. It seemed to the Board that Ross and Capra were even more removed from the ORNL review team in terms of contact than was Mr. Thatcher, and it was Mr. Thatcher who was the Staff's technical monitor for the ORNL review project.

III. REPLY TO PROPOSED FINDINGS ON CONTAINMENT ISOLATION

A. LICENSEE'S FINDINGS

52. Licensee addressed the requirements of Standard Review Plan Section 6.2.4 by installing an isolation upon receipt of a reactor trip signal, and asserts in its findings that this signal (in concert with pre-existing isolation signals) will assure automatic isolation prior to release of radioactivity from the reactor building for all postulated accident conditions (Licensee PF 213). Licensee's own witness, however, testified that this assumes that there are no spurious PORV openings (Sholly PF 110). Further, it is possible to have the purge line open on an operating or maintenance bypass and have the line fail to close on a reactor trip signal (Sholly PF 111).

53. As the Board earlier concluded (Sholly PF 113),

a safety-grade high radiation isolation signal is clearly preferable.

B. NRC STAFF'S FINDINGS

54. The Board notes that the Staff's findings acknowledge that it is possible to have a containment purge line open and have it fail to isolate on a reactor trip (NRC Staff PF 261). The additional assurance provided by a safety-grade high radiation isolation signal is believed by the Board to be very desirable in order to prevent what would clearly be very large releases of radioactivity to the environment if the containment purge line remained open during a LOCA, and the Board has required the installation of such a signal as soon as practicable (Sholly PF 113).

IV. REPLY TO PROPOSED FINDINGS ON THE PLANT COMPUTER

A. LICENSEE'S FINDINGS

55. Licensee's findings repeatedly emphasize that plant operators make very little use of the plant computer, especially in the very first stage of a transient/accident situation (See Licensee PF 237-243, 250). These findings ignore a very substantial record which demonstrates quite clearly that not only do operators and senior shift personnel (including the Shift Supervisor and Shift Foreman) utilize the

plant computer during transients and accidents (including the TMI-2 accident), but that such utilization of the computer is expected behavior from reactor operators (See Sholly PF 120-126).

56. Licensee also glosses lightly over the concerns raised by Staff witnesses on the accuracy and speed of the computer and its printout hardware (Licensee PF 246-250). It is clear, however, that even if the computer is 100% accurate (for which there is no guarantee), there are two situations in which the computer becomes of concern. First of all, it is not always immediately recognizable when the computer begins to have operational problems, unless the computer fails completely (Sholly PF 131). Secondly, Staff witness Price, a human factors consultant, testified that if the plant system status and the computer got out of synchronization, the use of computer information could present a problem (Tr. 10,545, Price; Sholly PF 127). Licensee, based on its findings, would apparently have this Board ignore these problems, despite clear evidence that the plant operators must be expected to rely on the plant computer during all types of situations (Sholly PF 124-126). Even the Licensee's witness, Mr. Keaten, agreed that if the computer is present, operators will attempt to utilize it under all conditions (Sholly PF 125; Tr. 10,547, Keaten).

57. Although it is clear that the Licensee does not rely on the plant computer to satisfy GDC 13 (Licensee PF 243, 251; Sholly PF 114; NRC Staff PF 303), it is also clear that

because of its very presence and its usefulness, the computer has become an integral part of the control room and an operational aid upon which operators rely for information under all conditions, including transients (See, generally, Sholly PF 120-128). As a result of the role of the plant computer and the concerns about the computer which were raised during the litigation of this contention, the Board has set forth additional requirements to be accomplished (Sholly PF 135-138).

B. NRC STAFF'S FINDINGS

58. Even more so than the Licensee's findings, the NRC Staff's findings ignore a very substantial record on the plant computer that was made during the litigation of the human factors engineering contention. The Staff's proposed findings are devoid of any reference to this record.

59. As a result, the Staff's findings suffer in general from the same weaknesses set forth in Paragraphs 55-57, supra, and perhaps to a greater extent since the Staff's findings simply ignore those issues completely, rather than at least attempt to set forth a partisan viewpoint as did the Licensee. The Staff concentrated in its findings solely on the issue of whether the computer was needed to comply with GDC 13. As the Board has observed, this is certainly not the sum total of the issues raised by this contention (See, generally, Sholly PF 114-134).

V. REPLY TO PROPOSED FINDINGS ON HUMAN FACTORS ENGINEERING REVIEW OF CONTROL ROOM DESIGN

A. LICENSEE'S FINDINGS

60. Licensee's findings rest generally, as do Mr. Sholly's, on the fact that extensive control room reviews have been conducted by both the Licensee (with assistance from human factors and control room design consultants) and the NRC Staff. As the Board has observed previously (Sholly PF 144, 162-163), the key to the full implementation of the improvements committed to by the Licensee will be the "followup" review by the NRC Staff to ensure that these commitments are kept on an expeditious basis.

61. One specific observation is in order, however, with respect to Licensee's findings. Licensee proposes, at Paragraph 302, essentially that adequate human factors considerations were involved in the original design of the TMI-1 control room. The Board cannot agree; for one, the number and degree of proposed improvements to the control room are a clear indication of the lack of human factors consideration in the original design. This was not unexpected, however, since no one involved with the original design had any academic training in human factors engineering, but rather the design team members relied on their experience in designing other control rooms and their general engineering backgrounds (Tr. 10,239-242, Meek).

62. The Board has, however, unlike the Licensee, found reason for a few additional requirements. These requirements, and the bases therefore, are set forth previously (Sholly PF 156-160).

B. NRC STAFF'S FINDINGS

63. The Staff's findings do not differ substantially from the Licensee's, and do not therefore require any additional response, other than to note the Board's prior emphasis on the inspection and enforcement aspects of the "followup" review on Licensee's implementation of its commitments on human factors improvements to the TMI-1 control room.

DATED: 1 July 1981

RESPECTFULLY SUBMITTED,

Steven C. Sholly

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(RESTART)
(Three Mile Island Nuclear Station, Unit No. 1)	)	
	)	

CERTIFICATE OF SERVICE

I hereby certify that single copies of INTERVENOR STEVEN C. SHOLLY REPLY TO PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON PLANT DESIGN ISSUES, dated 1 July 1981, were served upon those persons on the attached service list by deposit in the United States Mail, postage prepaid, first class, this 1st day of July 1981, with the only exceptions clearly noted on the service list.

Steven C. Sholly

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* Indicates hand delivery.

** Indicates delivery by hand on June 22, 1981, pursuant to Board Order authorizing early service of proposed findings and reply findings to Board members and timely service to other parties.

*** Indicates service on June 22, 1981, as explained immeidately above.

AMENDED CERTIFICATE OF SERVICE

Intervenor Sholly's Reply Findings were served on 13 July 1981, rather than on 1 July 1981, due to a change in the service date authorized by the Board. This change was not known when the original certificate of service was typed. The Board was served with advance copies prior to this date.

Steven C. Sholly