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DOLAN,J.E. Indiana & Michigan Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Requests deletion of License Condition (3)(c),per encl revised response to Question 212.40 in App Q of FSAR. Revision due to util misinterpretation of requirements re check valve leak testing.W/fee & affidavit.

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October 29, 1979
AEP:NRC:00259

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

Further review of Question 212.40 as contained in Appendix Q to the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) has led us to conclude that some of the testing described in the response is not necessary to satisfy the stated staff concerns and that the lists of valves need to be revised. The response to Question 212.40 was previously revised in our letter to Mr. Edson G. Case dated February 17, 1978. The intent of Question 212.40 is that we leak test the check valves which perform an isolation function of protecting low pressure safety systems from full reactor pressure. The staff required that each check valve which performs this isolation function be identified and classified ASME IWV-2000 category AC with the leak testing being performed to code specifications. License condition (3) (c) was included in our Unit No. 2 operating license in accordance with the commitments made in our response to Question 212.40.

Our review has indicated that in the cases where low pressure systems are isolated from full reactor pressure by check valves, the overpressure protection of the low pressure system piping is provided by ASME code safety relief valves. As such, the check valve performs an isolation function but does not protect low pressure systems from full reactor pressure. Our misinterpretation of the staff position contained in Question 212.40 resulted in the commitments made in the response which became license condition (3) (c). The results of our review are contained in a revised response to Question 212.40 which is attached for your review. We request that operating license condition (3) (c) be deleted in accordance with the attached revision to Question 212.40.

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S/1
w/check
$4,000.00

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This revision to the Question 212.40 response does not involve an unreviewed safety question or Technical Specification change, nor will it endanger the health or safety of the public. We intend to formally incorporate this revised response into the FSAR as part of a future Amendment.

Our review indicates that this revision constitutes a fee Class III Amendment to the facility license. In accordance with 10 CFR 170.22, we therefore enclose a check for $4,000.00.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

cc: R. C. Callen
G. Charnoff
D. V. Shaller-Bridgman
R. S. Hunter
R. W. Jurgensen

Response to Question 212.40

There are no check valves which protect low pressure piping from full reactor pressure. This overpressure protection is provided by safety relief valves on the low pressure piping systems as described below.

This response addresses the staff concern system by system. The design pressure of the boron injection system is higher than the design pressure of the Reactor Coolant System (RCS). Therefore the check valves in the boron injection system do not perform the function of protecting a low pressure system from full reactor pressure.

The function of protecting the Emergency Core Cooling Systems (ECCS) from full reactor pressure is performed by safety relief valves. The ECCS lines to the RCS hot legs are isolated by normally closed valves. The Residual Heat Removal normal cooldown line is isolated by normally closed valves. The check valves in the other ECCS lines perform an isolation function only to the extent that any leakage should not exceed the capacity of the associated safety valves. In each case, there are either two or three check valves in series between the RCS and the ECCS components with a lower pressure rating. These series check valves are listed in Table 212.40-1 along with the associated safety valves which protect the lower pressure systems. For each check valve, the rated capacity and pressure setting of the associated safety valve(s) are adequate to protect the low pressure piping system. The allowable leakage rate for each listed check valve was determined, very conservatively, based on the lowest relief capacity of the associated safety valve(s) and under the assumptions that all the other check valves in series are fully open and that all the other check valves in parallel leak at the maximum allowable rate.

The performance of the check valves in isolating the ECCS from full reactor pressure is tested at least once per 72 hours during operational modes 1, 2, 3 and 4 by Technical Specification surveillance requirement 4.4.6.2d. to demonstrate that unidentified leakage from the RCS is limited to 1 gpm. Because this limit is well below the allowable leakage rate through any check valve, the adequacy of these check valves to perform their isolation function is continuously verified by satisfaction of this surveillance requirement. Because of this requirement, any gradual deterioration of the check valve seats will be recognized and remedied.

These valves are located in systems that are normally maintained full of liquid, with either high pressure on the downstream side of the disc or no differential pressure across the disc. In this application, where the check valve is normally closed, any sudden, severe damage to the seating surface is very unlikely.

The test frequency for exercising the valves identified in Table 212.40-1 is in accordance with ASME Section XI paragraph IWV-3520 of the 1974 edition with addenda through the summer of 1975. These valves are normally closed during plant operation and cannot be exercised without initiating conditions similar to a safety injection. These valves will be exercised during cold shutdowns as stated in our Inservice Inspection Program submittals dated September 29, 1977 and September 22, 1978 (the latter resubmitted September 11, 1979.)

The design pressure of the Chemical and Volume Control System (CVCS) on the discharge side of the charging pumps is higher than the design pressure of the RCS. Therefore the discharge side of the CVCS does not require protection from full reactor pressure. The suction side of the charging pumps is protected by the suction header safety relief valve. The CVCS reciprocating charging pump discharge check valve is not required to perform a pressure isolation function because the construction of a multi-piston, positive displacement pump precludes pressure propagation in the reverse direction. The centrifugal charging pump discharge valves perform an isolation function only to the extent that any leakage should not exceed the capacity of the suction header safety relief valve. These check valves are listed in Table 212.40-2 along with the associated safety valve which protects the low pressure portion of the system. The pressure setpoints and relief flow capacity ratings for the safety valves are adequate to protect the low pressure piping system. The allowable leakage rate was determined assuming that all four check valves leak at the maximum allowable rate and that there is no recirculation. However, during all modes of plant operation with the Reactor Coolant System above 220 psi, normal practice is to have one charging pump running. Therefore, any leakage through the discharge check valve of a non-operating centrifugal charging pump is recirculated by the operating pump and does not cause a significant increase in the suction side pressure.

The testing for "exercising" will be performed for the check valves in Table 212.40-2 in the same manner and at the same frequency as described above for those in Table 212.40-1.

TABLE 212.40 - 1

ECCS SERIES CHECK VALVES

Check Valve	Nomenclature	Protecting Safety Valve (s)*	Allowable Check Valve Leakage Rate (GPM)
SI151E	ECCS Low Head Safety Injection	SV-104E	400
SI151W	ECCS Low Head Safety Injection	SV-104W	400
SI152N	ECCS Safety Injection	SV-98A	20
SI152S	ECCS Safety Injection	SV-98B	20
SI161L1	SI Hot To Cold Leg Crosstie	SV-98A & SV-104E	10
SI161L2	SI Hot To Cold Leg Crosstie	SV-98B & SV-104W	10
SI161L3	SI Hot To Cold Leg Crosstie	SV-98B & SV-104W	10
SI161L4	SI Hot To Cold Leg Crosstie	SV-98A & SV-104E	10
SI166-1	Accumulator Discharge	SV-100-1	47
SI166-2	Accumulator Discharge	SV-100-2	47
SI166-3	Accumulator Discharge	SV-100-3	47
SI166-4	Accumulator Discharge	SV-100-4	47
SI170L1	ECCS Cold Leg Loop	SV-98A, SV-100-1 & SV-104E	10
SI170L2	ECCS Cold Leg Loop	SV-98B, SV-100-2 & SV-104W	10
SI170L3	ECCS Cold Leg Loop	SV-98B, SV-100-3 & SV-104W	10
SI170L4	ECCS Cold Leg Loop	SV-98A, SV-100-4 & SV-104E	10

*The Safety Valve designations are the same as those used in the Unit 2 ISI Program.

TABLE 212.40 - 2

CVCS CENTRIFUGAL CHARGING PUMPS DISCHARGE CHECK VALVES

Check Valve	Nomenclature	Protecting Safety Valve (s)	Allowable Check Valve Leakage Rate (GPM)
CS299E	Discharge	SV-56	5
CS299W	Discharge	SV-56	5
CS297E	Recirculation	SV-56	5
CS297W	Recirculation	SV-56	5

STATE OF NEW YORK)
) ss.
COUNTY OF NEW YORK)

John E. Dolan, being duly sworn, deposes and says that he is the Vice President of licensees Indiana & Michigan Electric Company and Indiana & Michigan Power Company; that he has read the foregoing request and justification for deletion of Condition (3) (c) on License No. DPR-74 and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

John E. Dolan

Subscribed and sworn to before me this 29th day of October, 1979.

Kathleen Barry
Notary Public

KATHLEEN BARRY
NOTARY PUBLIC, State of New York
No. 41-4606792
Qualified in Queens County
Certificate filed in New York County
Commission expires March 30, 1981

Docket
~~50-317~~
50-318

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 17 1979

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All Power Reactor Licensees
All Applicants With Applications for a License

Gentlemen:

This past March, the NRC transmitted to you a copy of Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors" (ATWS) and a copy of an NRC letter that was sent this past February to each of the four nuclear reactor vendors. The letters to the vendors contained requests for information needed to perform generic analyses related to ATWS.

As we pointed out in our March letters, the generic analyses we requested were intended to confirm that the modifications proposed by the NRC staff for various classes of LWR designs would in fact accomplish the degree of ATWS prevention and mitigation described by the staff in its report. We also pointed out that we had chosen to work directly with the vendors in obtaining this information in an effort to conserve both NRC and industry resources. We requested that utilities cooperate with the vendors in performing the requested analyses.

Shortly after sending the letters to the vendors, the NRC Staff met with representatives of each of the NSSS vendors and many Utility representatives in Bethesda on March 1, 1979. The meeting was called to discuss the "early verification" approach in which we planned to use generic analyses as the basis for rulemaking. We hoped thereby to avoid costly and unnecessary repetitive analysis for individual plants. At the meeting, a tentative schedule was agreed to for generic analyses for each class of plants to be provided in three separate packages to be submitted May 1, September 1, and December 1, 1979.

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Immediately following the March 1 meeting, the NRC staff met separately with each of the NSSS vendors and agreements were made as to the minimum information to be supplied in the May 1 package. Also, as noted above, copies of the ATWS staff report and the generic analyses questions were transmitted to the Utilities.

On March 28, 1979 the Three Mile Island accident occurred. Because of the heavy expenditure of NRC resources required for Three Mile Island related activities, essentially no staff effort was applied to the ATWS issue for three months or so following the accident. There was also a substantial reduction in effort on the part of the PWR industry during that period, and some reduction for BWRs.

In June, 1979, the NRC Office of Nuclear Reactor Regulation was temporarily reorganized. Within this interim organization a group was assgned under the direction of S. Hanauer to work on the 19 Unresolved Safety Issues as designated by the Commission and reported to Congress this past January in NUREG-0510. ATWS is one of these 19 issues.

A preliminary NRR Staff review suggested that, for PWRs, the Three Mile Island accident raised new questions with regard to the appropriateness and adequacy of the resolution of ATWS as proposed by the Staff in Volume 3 of NUREG-0460. For BWRs, the staff has concluded that the technical impact of Three Mile Island was minimal and that the completion and review of the generic analyses for BWRs as specified in March should proceed as expeditiously as possible.

A meeting was held in Bethesda on July 25, 1979 to discuss, with representatives of PWR utilities and designers, considerations arising from the Three Mile Island accident that might be relevant to ATWS. For your information, a copy of the staff minutes of that meeting is attached as Enclosure 1. As can be seen from the minutes, at the meeting the staff:

a) Reiterated that ATWS is still believed by the staff to be a serious safety concern and that future protection should be provided. We stated that we are unwilling to wait another year to make progress on ATWS.

b) Expressed some general and specific technical concerns raised by the Three Mile Island accident with regard to the ATWS resolution proposed in Volume 3 of NUREG-0460.

c) Asked the industry to provide in writing, within 30 days of the meeting date, its preliminary assessment of the Three Mile Island impact on ATWS, the scope of effort now foreseen to resolve TMI issues, and a realistic schedule for providing the needed ATWS information. This would include both the March request and the TMI-related analyses.

OCT 17 1979

Subsequent to the July 25 meeting, we have met with representatives of the four NSSS vendors and of some Utility/Owners. We have met with GE to discuss the scope of the remaining generic analysis information to be supplied for BWR 4/5/6's. We have also met with representatives of the GE BWR/3 Owners, B&W, B&W ATWS Owners Group, W, W ATWS Owners Group, and CE. At all these meetings, we considered further the required information and the schedule for its submittal.

We have now received letters (see the list in Enclosure 2, attached) from the various groups describing the information to be furnished and projected schedules. On the basis of our review of these letters and meetings with the industry representatives, we perceive that the projected responses in several cases would not address several questions in our February 15 letter. In particular, several items are lacking that we will need to justify acceptance of the hardware approaches of NUREG 0460 Vol 3 rather than using the design basis accident approach.

I am determined to submit a proposed ATWS rule to the Commission for both PWRs and BWRs early in 1980. The type and content of the rule we will propose will depend critically upon the types and content of the information available to the staff. This will, of course, include whatever responses are actually provided by the industry in response to the questions attached to the February 15 staff letter, the March meetings, and the Three Mile Island related concerns as discussed in the July 25 and subsequent meetings.

I still believe that it is possible for the early verification generic analysis program to provide an acceptable resolution of the ATWS issue and that this is the way to achieve resolution with the least possible expenditure of NRC and industry resources. However, I want to reiterate that the success of this approach depends on whether or not all of the information necessary for the staff to confirm that its proposed ATWS modifications provide an acceptable level of protection, for all plants, is provided by the industry.

I strongly encourage you to join or form Utility/Owners Groups, if you have not already done so, and provide the resources necessary to supply the needed technical information pertaining to your plants, either operating or under construction. It would further reduce the impact on the industry as well as the staff resources if the ATWS effort coordination and the review role is performed by one industry group.

If you have additional questions on the generic analysis early verification program discussed in this letter, please contact Mr. Ashok Thadani, (301-492-7341).

Sincerely,

H. R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC-Industry ATWS Meeting Summary dtd 7/25/79
2. List of letters from Industry on Content of Report Submittals

ENCLOSURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 27 1979

Task Action Plan A-9

MEMORANDUM FOR: S. H. Hanauer

FROM: A. Thadani

SUBJECT: NRC-INDUSTRY ATWS MEETING SUMMARY

The staff met with the PWR vendors, the Atomic Industrial Forum (AIF) and several utility representatives to discuss the impact of TMI-2 events on the ATWS resolution plan described in Volume 3 of NUREG-0460.

The staff made the following initial remarks:

1) ATWS is still a safety concern and protection from these events must be provided. Although plants need not be shutdown immediately because of relatively low likelihood of a severe ATWS in a PWR in the next couple of years, ATWS resolution with suitable speed is necessary to permit an implementation plan which would assure an acceptably low risk from ATWS over the life of nuclear plants.

2) The staff would like to receive industry views on the impact of TMI-2 on ATWS and how to proceed from now on to resolve ATWS. The staff noted that they intend to propose an ATWS solution to the Commission preferably with but if necessary without the industry input.

3) In view of TMI-2 accident, the staff expressed the following general concerns with the Vol. 3 proposed resolution and asked for industry comments.

   a) What assurance do we have that the excessive calculated pressures for some designs modified per Alternative #3 would not result in loss of integrity of reactor coolant pressure boundary. (Note - Some designs may experience peak pressures - 4000 psi).

   b) Would increasing the number of safety valves as per Alternative #4 result in insufficient overall risk reduction? Would the primary system integrity be maintained? Would it be better to have larger capacity valves?

c) In view of questions a and b above, the pressurizer relief and safety valves must be qualified for water relief to assure that the nozzles, the valve body and the support structure integrity will be maintained and to estimate discharge flow rate and the likelihood and effects of valve chatter.

d) In view of significant plant differences in the designs of auxiliary feedwater system, Emergency Core Cooling Systems and other systems, how would the industry provide assurances that plant specific features have been adequately addressed in the "Early Verification" approach for resolving ATWS as described in NUREG-0460, Vol. 3.

e) Other Lessons Learned from TMI-2.

Following preliminary comments from the NRC staff members, G. Sorensen of WPPS who is also the Chairman of the AIF ATWS committee, made the following comments.

1) ATWS is not a safety issue but rather it is a licensing issue which needs resolution.

2) AIF in concert with the industry had reviewed ATWS in light of TMI-2 and had concluded that the Alternative #4 fix (mitigation) in Vol. 3 of NUREG-0460 is not the correct solution to ATWS. The industry believes that the alternative #2 fix (Prevention - Electrical Portion of RPS) is the appropriate ATWS solution.

3) Industry recognizes the TMI-2 impact on the role of the operator, his training aids and other lessons learned from this event. The industry believes that there is no need to rush to resolve ATWS because of the low probability of ATWS and because some of the anticipated changes to plants as a result of TMI-2 accident review would direct resources to other issues.

Following the AIF presentation, the staff raised their concerns that the ATWS resolution (not yet achieved) has been anything but hasty, that the NUREG documents on ATWS have been out for sufficiently long time period, that protection from ATWS is necessary, that TMI-2 event has raised concerns with the analyses assumptions and therefore the staff needs industry technical assessment of the TMI-2 impact on ATWS. The staff suggested that the TMI-2 event indicates a need to answer at least the following specific questions.

1) Analyses indicate the sensitivity of peak pressure to AFWS design and actuation time for some plants.

   Why should auxiliary feedwater actuation not be delayed beyond technical specification values? What bases are available to assume AFWS actuation earlier than the technical specification value? How do the analyses take into consideration the limits on AFWS injection rate due to water hammer considerations? How is the impact of flow restrictors on some AFWS designs considered in the ATWS analyses? How are the significant plant specific features of AFWS treated in the analyses?

2) As in question 1 above how are the differences in ECCS designs evaluated? For example, for some ATWS events, the pressure and the pressurizer level remain high enough such that either the HPSI cannot be actuated (because of shut off head considerations) or the operator may fail to actuate HPSI because of insufficient available information.

3) Would single failure cause all PORVs to fail to open? If so, then analyses must be based on all PORVs failing to open. Further, several plants are operating today with PORVs isolated. For these plants credit cannot be taken for relieving capability of these valves.

4) What assurance do we have that the ATWS events with a stuck open safety valve have been correctly analyzed? What is the potential for core uncovering under this scenario? What is the importance of ECCS actuation, reactor coolant pumps operation, and the pressurizer safety/relief valve discharge model on the potential for uncovering of the core? Further, why should more valves not be assumed to stick open following discharge of subcooled water.

5) For long term shutdown, discuss the following:

   a) available equipment, instrumentation and their qualification. (Must consider the effect of water discharged to the containment via ruptured quench tank).

   b) impact of loss of offsite power

   c) continued operation of reactor coolant pumps. Also consider tripping of reactor coolant pumps.

   d) Describe natural circulation, including effects of non-condensables. Is reflux boiling mode of operation anticipated? If so, justify.

e) Would one anticipate Boron precipitation problem? Also consider TMI-2 type problems with possible letdown line plugging from Boron precipitation.

f) How are leakage problems from equipment outside containment considered?

6) Why should credit be given for operator action even after ten minutes following an ATWS event initiation? TMI-2 experience does not provide enough confidence in the ability of the operator to perform correct actions only in this short time period under high stress conditions.

In response to the staff concerns the industry made the following comments.

AIF

1) The industry is frustrated because the staff concerns imply consideration of multiple failures and small LOCA which are beyond the credible events to be considered under ATWS. (Note - safety valve stuck open (small LOCA) is considered an anticipated transient).

2) Industry would like to wait for approximately six months before considering ATWS evaluations to minimize duplicate expenditures.

W

1) W has submitted responses to the 2/15/79 Mattson letter.

2) Calculated peak pressure of 2800 ∿ 2900 psi (for Alt. #3) and proposed modifications in turbine trip and auxiliary feedwater system actuation circuitry.

3) EPRI expects to issue a request for proposal to conduct tests on PORVs and safety valves and some results should be available by end of CY 79.

4) Recommended that "Early Verification" approach should be continued.

CE - Ed Shearer speaking for himself

1) TMI raises few questions like the behavior of S/R valves and the operator action. Further, prevention is better than mitigation and that mitigation would mean more and more analyses.

2) Continue with early verification.

B&W

1) Basically agrees with the staff concerns. Industry has longer list of items that could impact ATWS.

2) Stress analyses should be completed.

3) Likelihood of additional failures beyond ATWS should be considered.

4) Prevention is better than mitigation.

B&W Owners Group

1) ATWS is not a safety problem.

2) Even if ATWS occurs, no significant risk to public health and safety.

3) TMI-2 suggests a desirability for realistic analyses. TMI-2 suggests a need to assure that analyses bound the facilities.

4) Wait until "Lessons Learned" and "Bulletins and Orders" issues are resolved before pushing ahead with ATWS.

After the above industry comments, the staff made the following concluding remarks.

1) We don't intend to go too fast on ATWS.

2) If Early Verification is to be pursued then there is a need to assure that earlier ATWS analyses are correct and review the industry TMI-2 related list. In this regard the industry was invited to meet with the staff to discuss the technical issues which impact ATWS. The staff asked the industry to provide their assessment of TMI-2 impact on ATWS, the scope of effort to resolve these issues, and the schedule for performing this effort within 30 days.

3) We cannot wait another year to make progress in ATWS.

The list of attendees is in the enclosure.

A. Thadani

Enclosure:
As stated

cc: See next page

cc: Meeting Attendees
ATWS Distribution
PDR
RSB Files
T. Speis

ENCLOSURE

ATWS Meeting with Vendors & AIF

July 25, 1979

Ashok Thadani	NRC/DSS
Arthur McBride	B&W
Alan Hosler	WPPSS
Samir K. Sarkar	FP&L
Alan E. Ladieu	YAEC
Fred T. Stetson	AIF
Richard G. Rateick	DECO
Andrew J. Rushnok	OEC
M. Srinivasan	NRC/DSS
F. Akstulewicz	NRC/DSE
G. Sorensen	WPPSS/AIF
T. Speis	NRC/DSS
F. C. Cherny	NRC/DSS
J. A. Norberg	NRC/OSD
Stuart Thickman	TVA - EN DES
Karl O. Layer	BBR
J. Ted Enos	AP&L
Ted Myers	TECo
Robert Dieterick	SMUD
Michael J. Salerno	CPCo
S. Hardy Duerson	B&W
Bob Steither	W
Gary Augustine	W
P. M. Abraham	Duke Power
Mark Wisenburg	USTVA - Office of Power
Michael Tokar	NRC/DSS
Paul Boehnert	NRC/ACRS
David Bessette	NRC/ACRS
Steven Traisman	Pacific Gas & Electric
Sam Miranda	W
Pat Loftus	W
Fred Mosby	Wyle Laboratory
Roger Newton	Wisconsin Electric Power
Craig Grochmal	Stone & Webster
Charles A. Daverid	Long Island Lighting Co.
Robert L. Stright	SNUPPS
Joseph M. Weiss	GE
Joseph A. Gonyeau	Northern States Power

Seth M. Coplan	NRC/DSE
Clayton L. Pittiglio	NRC/DSE
Kulin D. Desai	NRC/DSS
Fuat Odar	NRC/DSS
Kris Parczewski	NRC/DOR
Roy Woods	NRC/DOR
Harold Vander Molen	NRC/DOR
Gururajarao Rangarao	PASNY
Frank McPhatter	B&W
Steve Banwarth	B&W
William R. Murray	Virginia Electric & Power Co.
Ben Rodell	VEPCO
Don Swanson	PGE Co.
Paul V. Holton	Bechtel
Tommy Errington	Mississippi Power & Light
Ron Clauson	Florida Power Corporation
Charles B. Brinkman	CE
C. L. Kling	CE
William Benjamin	Commonwealth Edison Co.
Denny Kreps	CE
William E. Burchill	CE
A. E. Scherer	CE
Richard C. L. Olson	Baltimore Gas & Electric Co.

ENCLOSURE 2

Letter from R. H. Bucholz (GE) to S. Hanauer, "ATWS Generic Analyses - Content of December 1979 Submittal", dated September 5, 1979.

Letter from J. H. Taylor (B&W) to S. Hanauer, "B&W Commitments for ATWS", dated September 13, 1979.

Letter A. E. Scherer (CE) to S. Hanauer, "NRC Request for Generic ATWS Information", dated August 31, 1979.

Letter L. O. DelGeorge (BWR 3 Owners representative) to S. Hanauer, "ATWS BWR/3 Plants and Vermont Yankee - Generic Analysis Supplement", dated August 28, 1979.

Letter T. M. Anderson (W) to S. Hanauer, "ATWS", dated August 24, 1979.

cc:
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