

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

PDR

July 24, 1985

OFFICE OF THE  
CHAIRMAN

50-346

The Honorable Edward J. Markey, Chairman  
Subcommittee on Energy Conservation and Power  
Committee on Energy and Commerce  
United States House of Representatives  
Washington, D.C. 20515

Dear Mr. Chairman:

This is in response to your letter of June 27, 1985 regarding the Davis Besse Nuclear power plant.

In regard to your request for an internal review, the Commission will certainly review the circumstances at Davis Besse to determine if NRC needs to address its means of regulation to reduce the possibility of incidents of this nature. No formal report is anticipated, but we will advise you of the results of our consideration.

Commissioner Asselstine believes that in view of the seriousness of the questions raised by the June 9, 1985 event at Davis Besse regarding the performance of NRC, an independent investigation is warranted.

You also asked that you be provided information on the steps that NRC will require prior to authorizing a restart of Davis Besse:

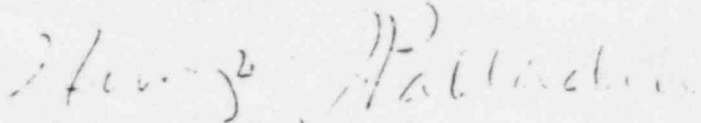
As indicated in our response to your June 17, 1985 questions, the NRC will assure that it is satisfied that the plant can be operated safely before it is restarted. This finding will be based on the effectiveness of the corrective actions and other improvements made to the existing AFW system in restoring it to the same quality standards assumed in the original licensing decision. In addition, the fact finding team sent to the Davis Besse site will develop information regarding the event and areas needing improvement. These recommendations will also be considered prior to a restart decision. The six action areas you have identified will be among the issues that will be considered by NRC prior to restart of Davis Besse.

In addition to completing the plant specific corrective actions on Davis Besse, we will determine the generic implications of the Davis Besse incident. The fact finding team is also charged with making generic recommendations which will be considered in our determinations.

Our response to your questions regarding the compliance of B&W plants with NRC regulations and standards is addressed in the answer to Question 1 herein.

The responses to the questions you appended to your letter are attached.

Sincerely,

A handwritten signature in dark ink, appearing to read "Nunzio J. Palladino". The signature is written in a cursive style with some loops and flourishes.

Nunzio J. Palladino  
Chairman

Attachment: Responses to Questions from  
June 27, 1985 letter

cc: Rep. Carlos Moorhead

QUESTION 1:      ACCORDING TO AN APRIL 17, 1984 MEMORANDUM FROM L. S. RUBENSTEIN, ASSISTANT DIRECTOR FOR CORE AND PLANT SYSTEMS, TO GUS C. LAINAS, ASSISTANT DIRECTOR FOR LICENSING, NO BABCOCK & WILCOX (B&W) REACTOR MEETS THE RELIABILITY CRITERIA OF THE STANDARD REVIEW PLANT SECTION 10.4.9. EACH OF THE B&W PLANTS WAS CITED FOR ALSO FAILING TO MEET ONE OR MORE GENERAL DESIGN CRITERIA.

- A.    WHY ARE THESE PLANTS ALLOWED TO OPERATE IF THEY FAIL TO COMPLY WITH THE GENERAL DESIGN CRITERIA AND STANDARD REVIEW PLAN?

ANSWER:

THE PLANTS DISCUSSED IN THE MEMORANDUM FROM L. S. RUBENSTEIN TO GUS C. LAINAS, DATED APRIL 17, 1984, WERE ALL LICENSED PRIOR TO THE ACCIDENT AT TMI-2. THOSE PLANTS MET THE GENERAL DESIGN CRITERIA AS THEY WERE INTERPRETED WHEN THE PLANTS RECEIVED THEIR OPERATING LICENSES AND CONTINUE TO DO SO. AFTER THE ACCIDENT, CERTAIN REQUIREMENTS WERE SET FORTH IN NUREG-0737, "CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS." ITEM II.E.1.1 OF NUREG-0737

REQUIRED A REEVALUATION OF THE AUXILIARY FEEDWATER SYSTEMS (AFWS) AGAINST THE CURRENT CRITERIA OF SRP SECTION 10.4.9 AND BTP ASB 10-1. WHILE THE OPERATING PLANTS MAY NOT MEET THE CURRENT CRITERIA, WHICH WERE DEVELOPED AFTER THEIR LICENSING, THE DESIGN OF THOSE PLANTS REGARDING THE CURRENT CRITERIA WERE IDENTIFIED FOR FUTURE CONSIDERATION BY THE STAFF AND POSSIBLE BACKFIT DETERMINATIONS. IN THE CASE OF THE DAVIS-BESSE PLANT, THE REQUIREMENT FOR A THIRD AFWS WAS CONSIDERED AS A BACKFIT.

THESE PLANTS WERE ALLOWED TO OPERATE BECAUSE THEY WERE NOT FOUND TO BE UNSAFE AND CONTINUED TO MEET THE INTERPRETATION OF THE GDC WHICH WERE IN EFFECT AT THE TIME THE PLANTS WERE LICENSED. THE APRIL 17, 1984 MEMORANDUM CITES AREAS WHERE EACH OF THE B&W PLANTS DID NOT MEET CERTAIN GENERAL DESIGN CRITERIA (GDC); HOWEVER, THIS WAS BASED ON NOT MEETING THE CURRENT CRITERIA FROM THE STANDARD REVIEW PLAN. THIS DID NOT MEAN THAT THE B&W PLANTS WERE NOT IN COMPLIANCE WITH THE GDCs APPLICABLE TO THESE PLANTS WHEN LICENSED. THE NEW STANDARD REVIEW PLAN ELEMENTS RELATED TO AFW SYSTEMS SPELL OUT THE BASIS FOR REVIEW OF A NEW PLANT NOT YET LICENSED. THE MINIMUM OF THESE REQUIREMENTS WHICH NEEDS TO BE SATISFIED ON A LICENSED PLANT, SUCH AS DAVIS-BESSE, IS SPELLED OUT IN NUREG-0737. THE IMPROVEMENTS IN SAFETY REQUIRED BY THE COMMISSION FOLLOWING THE TMI-2 ACCIDENT WERE IMPOSED ON



ALL PLANTS INCLUDING OPERATING PLANTS, IN NUREG-0737. THOSE REQUIREMENTS AND IMPLEMENTATION SCHEDULES DERIVED FROM THE NUREG-0737 REVIEW HAVE OR ARE BEING MET BY ALL THE B&W LICENSEES. WITH RESPECT TO THE AFW SYSTEMS, ALL OF THE B&W PLANTS HAVE MET THE NUREG-0737 REQUIREMENTS, EXCEPT FOR A) RANCHO SECO AND TMI-1 WHICH HAVE SCHEDULED COMPLETION AT THE NEXT REFUELING OUTAGES, AND B) ARKANSAS 1 WHICH HAS PROPOSED ADDITIONAL MODIFICATIONS BEYOND THOSE PREVIOUSLY REVIEWED AND APPROVED BY THE STAFF. IN SUMMARY, THE ORIGINAL LICENSING REQUIREMENTS PLUS THE NUREG-0737 REQUIREMENTS ARE ALL THAT ARE NEEDED TO MEET THE COMMISSION'S REGULATIONS REGARDING AUXILIARY FEEDWATER.

COMMISSIONER ASSELSTINE ADDS:

THE AUXILIARY FEEDWATER SYSTEM (AFS) OF PLANTS LICENSED BEFORE AROUND 1977 WERE NOT REQUIRED TO MEET THE PROVISIONS OF GDC-34 OR -44, BECAUSE UP UNTIL THEN THE STAFF PRACTICE WAS TO EXAMINE THE TOTALITY OF WAYS OF REMOVING DECAY HEAT (E.G., VIA THE RESIDUAL HEAT REMOVAL SYSTEM). SUBSEQUENTLY, THE STAFF MADE THE DETERMINATION THAT GDC-34 SHOULD APPLY TO THE AFS, BUT THE PLANTS THAT WERE ALREADY OPERATING OR FOR WHICH THE SAFETY REVIEWS HAD ESSENTIALLY BEEN COMPLETED WERE "GRANDFATHERED" FROM THE THEN NEW INTERPRETATION OF GDC-34. LIKEWISE, WHEN THE STANDARD REVIEW PLAN WAS MODIFIED TO INCLUDE A RELIABILITY CRITERION FOR THE AFS, ALL PLANTS ALREADY LICENSED WERE

"GRANDFATHERED". THUS, UNDER THE COMMISSION'S CURRENT PRACTICES, THE STAFF WOULD HAVE TO DEMONSTRATE THAT SUBSTANTIAL ADDITIONAL OVERALL PROTECTION IS NEEDED AT THOSE OPERATING PLANTS PREVIOUSLY GRANDFATHERED BEFORE IT COULD REQUIRE THEM TO CONFORM WITH GDC-34 OR THE RELIABILITY CRITERION AND THIS WOULD ENTAIL A COST-BENEFIT ANALYSIS OF THE MERITS OF SUCH A BACKFIT. IT APPEARS TO ME THAT THERE WERE ERRORS IN JUDGMENT ON THE LICENSING OF THE PLANTS THAT WERE "GRANDFATHERED", BUT THE COMMISSION'S CURRENT BACKFITTING PRACTICES GET IN THE WAY OF CORRECTING THOSE ERRORS.

QUESTION 1.B WERE EXEMPTIONS OR AMENDMENTS GRANTED FOR EACH PLANT,  
AND IF SO ON WHAT BASIS?

ANSWER:

NO EXEMPTIONS WERE CONSIDERED NECESSARY.

QUESTION 1.c IN TERMS OF THE PROBABILITY OF A LOSS OF MAIN FEEDWATER, A LOSS OF AUXILIARY FEEDWATER OR THE LIKELIHOOD OF LOSS OF FEEDWATER LEADING TO A SEVERE CORE DAMAGE ACCIDENT, HOW DO B&W REACTORS COMPARE TO OTHER DESIGNS?

ANSWER.

THE RESPONSE TO THIS QUESTION IS DEPENDENT ON PLANT-SPECIFIC FEATURES THAT ARE MOST OFTEN INFLUENCED BY THE BALANCE OF PLANT DESIGN AND NOT THE REACTOR VENDOR. THIS WAS QUITE EVIDENT IN THE EVALUATION OF AUXILIARY FEEDWATER SYSTEMS (AFW) FOR A SINGLE REACTOR VENDOR (WESTINGHOUSE) IN NUREG-0611 WHICH SHOWED A LARGE SPREAD IN ESTIMATED RELIABILITY FOR SYSTEMS HAVING THREE PUMPS. MOST OF THE OPERATING B&W PLANTS HAVE TWO AUXILIARY FEEDWATER PUMPS AND FEED AND BLEED CAPABILITY (THE CAPABILITY TO REMOVE DECAY HEAT DIRECTLY VIA THE PRIMARY SYSTEM.

THE IMPORTANT FACTORS ARE THE CONFIGURATION OF SECONDARY HEAT REMOVAL SYSTEM (NUMBER OF TRAINS AND DIVERSITY) AND THE VIABILITY OF FEED AND BLEED. THESE FACTORS VARY WITHIN A REACTOR TYPE AND THEREFORE A COMPARISON OF B&W REACTORS TO OTHER REACTOR DESIGNS IS NOT DIRECTLY AVAILABLE. RANGES OF ESTIMATED FREQUENCIES DEVELOPED FROM VARIOUS GENERIC AS WELL AS PLANT SPECIFIC STUDIES FOR LOSS OF MAIN FEEDWATER EVENTS LEADING TO SEVERE CORE DAMAGE ARE AS FOLLOWS:

PLANTS WITH TWO PUMP AFW  
SYSTEMS AND NO FEED AND  
BLEED CAPABILITY

$2 \times 10^{-4}$  TO  $6 \times 10^{-4}$ /RY

PLANTS WITH TWO PUMP AFW  
SYSTEMS AND FEED AND BLEED  
CAPABILITY

$10^{-5}$  TO  $10^{-4}$ /RY

PLANTS WITH THREE PUMP AFW  
SYSTEMS AND FEED AND BLEED  
CAPABILITY

$10^{-6}$  TO  $10^{-5}$ /RY

IN SUMMARY, IT CAN BE SAID THAT THOSE PLANT DESIGNS WITH A TWO PUMP AFW AND LACKING FEED AND BLEED CAPABILITY COMPARE UNFAVORABLY TO THOSE REACTORS WITH A GREATER DEPTH OF DEFENSE AGAINST LOSS OF FEEDWATER SCENARIOS. THIS IS TRUE FOR ANY VENDOR'S PLANTS, NOT JUST B&W REACTORS. NOTE ALSO THAT DIFFERENCES IN EQUIPMENT RELIABILITY IN SERVICE MAY BE AS IMPORTANT AS DESIGN VARIANTS IN GOVERNING SYSTEM RELIABILITY OR ACCIDENT LIKELIHOOD.

COMMISSIONER ASSELSTINE ADDS:

THE COMMISSION CLAIMS THAT MOST B&W PLANTS HAVE "FEED AND BLEED CAPACITY". THIS ISSUE WAS EXTENSIVELY LITIGATED IN THE TMI-1 RESTART PROCEEDING. SEE, 17 NRC 814 (1983). THE STAFF TESTIFIED IN THAT PROCEEDING THAT STAFF DID NOT RELY ON FEED AND

BLEED TO PROVIDE ADEQUATE CORE COOLING IN THE EVENT OF A SMALL BREAK LOCA OR MAIN FEEDWATER TRANSIENT. FOLLOWING IS THE APPEAL BOARD DECISION ON THIS MATTER:

"(I)T WAS LEFT TO THE STAFF TO DEMONSTRATE THE RELIABILITY OF FEED AND BLEED. WE FIND THAT THE STAFF WAS UNABLE TO MAKE SUCH A DEMONSTRATION. PLAINLY THE FEED AND BLEED PROCESS IS CONCEPTUALLY VALID. THE STAFF, HOWEVER, WAS UNABLE TO RESOLVE NUMEROUS ANALYTICAL UNCERTAINTIES WITH REGARD TO THE PROCESS. AS A CONSEQUENCE, WE ARE UNABLE ON THIS RECORD TO ENDORSE FEED AND BLEED AS A RELIABLE BACKUP SYSTEM OF DECAY HEAT REMOVAL." 17 NRC 814 (1983), PP.822-23.

THE COMMISSION DID NOT MODIFY THAT APPEAL BOARD DECISION. IT APPEARS THAT THE COMMISSION IS NOW RELYING ON THE FEED AND BLEED MODE OF CORE COOLING EVEN THOUGH ITS OWN APPEAL BOARD, WHICH CAREFULLY EXAMINED THE TECHNICAL MERITS OF THE ISSUE, WOULD NOT ENDORSE THE PROCESS, AT LEAST FOR ONE B&W PLANT.

WITH REGARD TO THE JULY 17, 1985 RESPONSES TO YOUR LETTER OF JUNE 17, 1984, QUESTION 5 ASKED FOR THE IDENTITY OF ALL PWR'S FOR WHICH FEED AND BLEED HAS BEEN DEMONSTRATED, USING SAFETY EVALUATIONS PERFORMED IN ACCORDANCE WITH ESTABLISHED NRC REQUIREMENTS, TO BE AN ADEQUATE METHOD OF CORE COOLING. THE RESPONSE IDENTIFIES EIGHT B&W PLANTS, INCLUDING TMI-1, AS HAVING ADEQUATE FEED AND BLEED CAPABILITY. BASED ON THE ABOVE APPEAL BOARD DECISION, IT APPEARS THAT FOR TMI-1 THE ADEQUACY OF THE

FEED AND BLEED MODE OF CORE COOLING HAS NOT BEEN DEMONSTRATED. FURTHER, FOR THE OTHER PLANTS LISTED IN THAT RESPONSE, IT IS NOT CLEAR WHETHER THE ANALYSES PERFORMED WERE BASED ON SAFETY EVALUATIONS PERFORMED IN ACCORDANCE WITH ESTABLISHED NRC REQUIREMENTS. I WOULD HAVE TO CONCLUDE, THEREFORE, THAT IT IS NOT CLEAR THAT THE ADEQUACY OF FEED AND BLEED HAS BEEN DEMONSTRATED FOR ANY PLANT.



QUESTION 2 GIVEN THE REPEATED RECOMMENDATIONS FOR UPGRADING THE AUXILIARY FEEDWATER SYSTEM AT DAVIS-BESSE TO INCLUDE A THIRD, DIVERSE-DRIVE AUXILIARY FEEDWATER PUMP, ON WHAT BASIS HAS THIS RECOMMENDATION NOT YET BEEN IMPLEMENTED? SPECIFICALLY, DOES THE COMMISSION BELIEVE THAT IT IS ACCEPTABLE THAT THE NRC HAS BEEN UNABLE TO RESOLVE THIS ISSUE FOR SIX YEARS?

ANSWER

THE LONG EXCHANGE WITH THE LICENSEE, AND AMONG THE STAFF REGARDING THE THIRD AUXILIARY FEEDWATER PUMP ISSUE IS A RESULT OF DIFFERING JUDGEMENTS ON THE MOST EFFECTIVE WAY OF IMPROVING AUXILIARY FEEDWATER RELIABILITY. THE YEARS THAT IT HAS TAKEN TO RESOLVE THE ISSUES SURROUNDING THE UPGRADE OF THE AUXILIARY FEEDWATER SYSTEM AT DAVIS BESSE IS UNUSUAL AND UNDESIRABLE. THE LICENSEE PROVIDED STUDIES THAT PROPOSED MORE COST EFFECTIVE ALTERNATES (BASED ON THE LICENSEE'S ANALYSIS) TO ACHIEVE ACCEPTABLE RELIABILITY GOALS. THE DETAILS REGARDING THE LICENSEE STUDIES, STAFF REVIEWS INCLUDING A NRC-SPONSORED BROOKHAVEN NATIONAL LABORATORY

STUDY AND EFFORT NEEDED TO JUSTIFY GENERIC BACKFIT FOR A THIRD PUMP REQUIREMENT ARE PROVIDED HEREIN. EVEN AT THIS TIME, THE LICENSEE'S COMMITMENT FOR THE INSTALLATION OF A THIRD PUMP IS NOT BASED ON THE RESULTS OF A RELIABILITY ANALYSIS BUT THE NEED TO RELOCATE THE STARTUP FEEDWATER PUMP TO RESOLVE SAFETY CONCERNS ABOUT HIGH AND MODERATE ENERGY LINE BREAKS IN THE AUXILIARY BUILDING THAT COULD EFFECT THE AUXILIARY FEEDWATER SYSTEM.

THE EARLIEST REFERENCE TO IMPOSING A REQUIREMENT FOR DIVERSE-DRIVE FOR THE AFW PUMPS AT DAVIS-BESSE APPEARS IN THE STAFF SAFETY EVALUATION WHICH WAS TRANSMITTED WITH NRC LETTER DATED JULY 6, 1979. IN THAT SAFETY EVALUATION, THE STAFF TOOK NOTE OF TOLEDO EDISON COMPANY'S AGREEMENT TO CONTINUE TO REVIEW AFW SYSTEM PERFORMANCE TO ASSURE RELIABILITY, AND CONSISTENT WITH THAT LONG TERM AGREEMENT, WOULD REQUIRE THE ADDITION OF A 100% CAPACITY ELECTRIC-MOTOR DRIVEN AFW PUMP. THE STAFF SPECIFICALLY HELD OPEN THE OPTION FOR OTHER ALTERNATIVES TO THE 100% CAPACITY DIVERSE-DRIVE PUMP.

SUBSEQUENTLY, THE B&W REACTORS TRANSIENT TASK FORCE RECOMMENDED THE EXPEDITED INSTALLATION OF THE 100% CAPACITY DIVERSE DRIVE PUMP AT DAVIS-BESSE. THE BASIS FOR THE RECOMMENDATION WAS THAT AN INTERRUPTION OF FEEDWATER FLOW CAN RESULT IN STEAM GENERATOR

DRY-OUT WHICH WOULD RESULT IN LOSS OF STEAM PRESSURE TO DRIVE THE AFW PUMP TURBINES. DIVERSITY OF AFW PUMP POWER WAS NOT AN EXPLICIT REQUIREMENT FOR DAVIS-BESSE WHEN IT WAS LICENSED. THE SRP REQUIREMENTS FOR DIVERSITY IN BTP-10-1 WAS ADOPTED SUBSEQUENT TO DAVIS-BESSE BEING LICENSED. NRR INITIALLY CONSIDERED ISSUING AN ORDER REQUIRING THE INSTALLATION OF THE ELECTRIC MOTOR DRIVEN PUMP. HOWEVER THE DECISION WAS MADE TO AWAIT THE RESULTS OF A RELIABILITY STUDY BEING PERFORMED BY THE LICENSEE AND NOT TO ISSUE THE ORDER. IT WAS FELT, AT THE TIME, THAT A RELIABILITY STUDY WOULD DEMONSTRATE THE RELATIVE IMPROVEMENT IN RELIABILITY TO BE GAINED THROUGH THE INSTALLATION OF THE ADDITIONAL PUMP VS. OTHER SYSTEM MODIFICATIONS WHICH COULD ELIMINATE DOMINANT FAILURE CONTRIBUTORS.

THE RELIABILITY STUDY WAS SUBMITTED DECEMBER 1981. THE LICENSEE'S CONCLUSION WAS THAT THE ADDITION OF A THIRD TRAIN OF AFW WOULD NOT IMPROVE RELIABILITY OF AFW AS MUCH AS WOULD OTHER MODIFICATIONS TO THE EXISTING SYSTEM TO ELIMINATE THE DOMINANT FAILURE CONTRIBUTORS.

DURING THIS TIME, NO SPECIFIC ANALYSES HAD BEEN PERFORMED BY OR FOR THE NRC THAT COULD JUSTIFY, IN A QUANTITATIVE MANNER, AN ORDER FOR THE THIRD PUMP. THE NRC PROCEEDED TO REVIEW THE LICENSEE'S RELIABILITY STUDY AND HAD BROOKHAVEN NATIONAL LABORATORY PREPARE AN INDEPENDENT EVALUATION. CONCURRENTLY, WORK HAD

STARTED ON THE PREPARATION OF A PACKAGE FOR CRGR CONSIDERATION WHICH, IF APPROVED BY THE CRGR, WOULD REQUIRE THE INSTALLATION OF A DIVERSE-DRIVE PUMP ON ALL PLANTS WHICH DO NOT HAVE ONE. PRESENTATION OF THIS PACKAGE TO CRGR HAS BEEN DELAYED, PRIMARILY TO ENSURE THE STAFF HAS AN ADEQUATE JUSTIFICATION FOR RECOMMENDATION.

COMMISSIONER ASSELSTINE ADDS:

I BELIEVE IT IS UNACCEPTABLE THAT THIS AGENCY HAS BEEN UNABLE TO RESOLVE THE ISSUE OF WHETHER REQUIRING DIVERSE MEANS FOR REMOVING DECAY HEAT FROM A REACTOR IS REASONABLE. THE STAFF'S INITIAL INSTINCTS ON THIS ISSUE, THAT DIVERSE MEANS SHOULD BE REQUIRED, APPEAR TO BE RIGHT ON THE MARK. THE FACT THAT THE MATTER HAS REMAINED MIRED IN THE MORASS OF "COST-BENEFIT ANALYSES" FOR THE LAST SEVERAL YEARS IS ILLUSTRATIVE OF THE PROBLEMS ASSOCIATED WITH THE COMMISSION'S POLICIES, PRACTICES AND DECISIONS ON BACKFITTING.

QUESTION 3.

1  
ACCORDING TO A SEPTEMBER 28, 1984 MEMORANDUM FROM HAROLD R. DENTON, DIRECTOR OF THE OFFICE OF NUCLEAR REACTOR REGULATION TO WILLIAM J. DIRCKS, EXECUTIVE DIRECTOR FOR OPERATIONS, THE NUREG-0667 RECOMMENDATION FOR A DIVERSE DRIVE AUXILIARY FEEDWATER PUMP AT DAVIS-BESSE WAS NEVER IMPLEMENTED BECAUSE THE AUXILIARY SYSTEMS BRANCH DETERMINED THAT IT WAS NOT NECESSARY. PLEASE PROVIDE ANY SUCH ANALYSIS OR RECOMMENDATION BY THE AUXILIARY SYSTEMS BRANCH AND ALL RELATED DOCUMENTS. ADDITIONALLY, PLEASE RECONCILE THIS STATEMENT WITH A MARCH 12, 1985 MEMORANDUM FROM OLAN D. PARR, CHIEF OF THE AUXILIARY SYSTEMS BRANCH TO JOHN F. STOLZ, CHIEF OF THE OPERATING REACTORS BRANCH NO. 4, THAT STATES: "WE ARE DELIGHTED THAT THE LICENSEE [HAS DECIDED TO ADD] THIS THIRD, MOTOR DRIVEN PUMP."

ANSWER.

THE PURPOSE OF THE SEPTEMBER 28, 1984 MEMORANDUM WAS TO CLOSEOUT TMI ISSUE II.E.5.2 DUE TO ISSUANCE OF NUREG-0667 IN MAY 1980. IN ADDITION A STATUS REPORT WAS PROVIDED ON THE IMPLEMENTATION OF THE 22 NUREG-0667 RECOMMENDATIONS. THE INFORMATION IN THE SEPTEMBER 28, 1984 MEMORANDUM FROM HAROLD R. DENTON TO WILLIAM J. DIRCKS PERTAINING TO DAVIS-BESSE DIVERSE DRIVE AUXILIARY FEEDWATER

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QUESTION 3. (CONTINUED) - 2 -

PUMPS WAS BASED ON AN ERRONEOUS INTERPRETATION OF STATUS INFORMATION PROVIDED BY AN INDIVIDUAL IN AUXILIARY SYSTEMS BRANCH (ASB) IN JULY 1982 ON THIS ISSUE. THIS STATUS INFORMATION RELATED TO A LICENSEE SUBMITTAL CONCLUDING THAT THE THIRD PUMP WAS NOT NEEDED. THE SEPTEMBER 28, 1984 MEMORANDUM WAS PREPARED BY A NEWLY APPOINTED LEAD PROJECT MANAGER FOR THE TMI ACTION PLAN ITEM II.E.5.2. IN THE COURSE OF TRANSFERRING THE RESPONSIBILITY FOR THIS TASK, SEVERAL DISCUSSIONS WERE HELD AND NUMEROUS DOCUMENTS CHANGED HANDS ON THE 22 RECOMMENDATIONS INVOLVED. THE STATUS FOR DAVIS BESSE WAS BASED ON THESE DISCUSSIONS AND A STATUS NOTE ON THE RECOMMENDATION FOR A THIRD AFW PUMP SAYING "LICENSEE SUBMITTED INFO AND INDICATED THAT DIVERSE-DRIVE AFW IS NOT NEEDED." SINCE THEIR STATUS WAS PROVIDED BY ASB, THE NEW LEAD PROJECT MANAGER CONCLUDED THAT THE ISSUE HAD ALREADY BEEN RESOLVED AND THAT THE AUXILIARY SYSTEMS BRANCH HAD AGREED THAT THE DIVERSE-DRIVE PUMP WAS NOT NEEDED. WHILE ASB WAS SENT COPIES OF THIS MEMORANDUM, NO ONE IN ASB NOTED THE INCORRECT STATUS.

IN FACT, A REVIEW OF THE AUXILIARY SYSTEMS BRANCH ACTIVITIES DURING 1984, AS DOCUMENTED IN THE MEMORANDA ALREADY PROVIDED TO THE SUBCOMMITTEE, CLEARLY INDICATES A CONSISTENT DESIRE BY THE AUXILIARY SYSTEMS BRANCH TO REQUIRE THE LICENSEE TO ADD THE THIRD PUMP. A MEMO CORRECTING THIS STATUS HAS BEEN SENT TO THE EDO.

MARKEY/NRR  
REVISED 7/16/85

QUESTION 3. (CONTINUED) - 3 -

THE AUXILIARY SYSTEMS BRANCH WAS NOT AWARE OF THE CLOSEOUT OF TMI ACTION PLAN ITEM II.E.5.2, AND CONSEQUENTLY IT DID NOT INFLUENCE THEIR THEN ONGOING ACTIONS REGARDING THE NEED FOR ADDING THE THIRD PUMP.



QUESTION 4.

ACCORDING TO A JANUARY 16, 1984 MEMORANDUM FROM ASHOK C. THADANI, CHIEF OF THE RELIABILITY AND RISK ASSESSMENT BRANCH, TO OLAN PARR, CHIEF OF THE AUXILIARY SYSTEMS BRANCH, THE AVERAGE PROBABILITY OF A CORE MELTDOWN INITIATED BY A LOSS OF MAIN FEEDWATER IS APPROXIMATELY 1 IN 2,000 PER REACTOR YEAR ( $5.4 \times 10^{-4}$ ). GIVEN CURRENT PLANT CONFIGURATIONS, IS THE COMMISSION AWARE OF ANY SINGLE ACCIDENT SEQUENCE AT ANY PLANT THAT POSES AS HIGH A PROBABILITY OF RESULTING IN CORE DAMAGE?

ANSWER.

NO, THE COMMISSION IS NOT AWARE OF ANY OTHER SINGLE ACCIDENT SEQUENCE AT ANY PLANT THAT COULD RESULT IN CORE DAMAGE AND WHICH CURRENTLY HAS AN ESTIMATED FREQUENCY AS HIGH AS  $5 \times 10^{-4}$ /RY. PRAs HAVE REVEALED INSTANCES OF ACCIDENT SEQUENCES WITH FREQUENCIES ABOVE  $5 \times 10^{-4}$ /RY, BUT NONE REMAIN SO HIGH TODAY. IN MOST CASES, CHANGES BY THE PLANT OWNERS SUBSTANTIALLY REDUCED THE VULNERABILITIES REVEALED BY THE PRA.

COMMISSIONER ASSELSTINE ADDS:

THE ATTACHED TABLE 1 FROM THE JANUARY 16, 1984 MEMORANDUM REFERENCED IN THE QUESTION PROVIDES ESTIMATES OF THE CORE MELTDOWN FREQUENCIES DUE TO THE LOSS OF MAIN FEEDWATER AT 10

PLANTS. I FIND THOSE ESTIMATES, PARTICULARLY WHEN ONE CONSIDERS THE UNCERTAINTIES, TO BE UNACCEPTABLY HIGH FOR THE REMAINING LIFE OF THE PLANTS. UNFORTUNATELY, THE COMMISSION FINDS, THROUGH ITS SEVERE ACCIDENT POLICY STATEMENT, THE CORE MELTDOWN RISKS TO BE ACCEPTABLE.

TABLE 1

ESTIMATES OF CORE MELT FREQUENCY (PER YEAR) AND ASSOCIATED UNCERTAINTIES  
DUE TO LOSS OF MAIN FEEDWATER EVENT

Plant	Median	Error Factor	Mean
Prairie Island	1.4E-6 <sup>(1)</sup> 2.4E-4 <sup>(2)</sup>	37 16	1.6E-5 1.0E-3
Haddam Neck	1.2E-6 2.1E-4	37 16	1.4E-5 8.0E-4
San Onofre	9.0E-7 1.6E-4	37 16	1.1E-5 6.7E-4
ANO-2	-- <sup>(3)</sup> 4E-5	-- 16	-- 1.6E-4
Ft. Calhoun	-- 4E-5	-- 16	-- 1.6E-4
ANO-1	9.0E-7 6.4E-5	37 16	9.7E-6 2.6E-4
Davis-Besse	-- 1.3E-4	-- 16	-- 5.4E-4
Crystal River	2.8E-6 2.0E-4	37 16	3.1E-5 8.3E-4
Rancho Seco	8.5E-7 6.1E-5	37 16	9.3E-6 2.5E-4
Turkey Point	4.0E-8 6.7E-6	37 16	4.5E-7 2.8E-5

(<sup>1</sup>) The top values include credit for the feed-and-bleed mode.

(<sup>2</sup>) The bottom values exclude the feed-and-bleed mode.

(<sup>3</sup>) We assumed that ANO-2, Davis-Besse, and Ft. Calhoun have no feed-and-bleed capability.

QUESTION 5:

TOLEDO EDISON HAS FINALLY DECIDED TO UPGRADE THE AUXILIARY FEEDWATER SYSTEM BY INSTALLING A THIRD DIVERSE DRIVE AUXILIARY FEEDWATER PUMP AT DAVIS-BESSE. PLEASE PROVIDE THE FOLLOWING INFORMATION:

- A. WILL THE MODIFICATION CONSTITUTE A NEW TRAIN OF AUXILIARY FEEDWATER OR JUST A NEW PUMP? WHAT OTHER OPTIONS ARE POSSIBLE? PLEASE STATE THE RELATIVE ADVANTAGES AND DIS-ADVANTAGES OF ALL OPTIONS.

ANSWER:

BASED ON THE TOLEDO EDISON PROPOSED DESIGN CHANGE THE STAFF UNDERSTANDS THE MODIFICATION TO CONSIST OF A NEW TRAIN AND NOT JUST A NEW PUMP. THE TRAIN WILL CONSIST OF A 100% CAPACITY PUMP AND THE ASSOCIATED VALVES AND PIPING. THE NEW TRAIN WILL BE INDEPENDENT OF THE EXISTING AFW SYSTEM EXCEPT FOR THE POINT OF INJECTION INTO EITHER OF THE TWO STEAM GENERATORS AND THE SOURCES OF WATER.

REGARDING OTHER POSSIBLE OPTIONS, TWO COME TO MIND AS THEY ARE PRESENTLY USED IN OPERATING PLANTS:

TURBINE DRIVE AND ELECTRIC DRIVE COUPLED TO A COMMON PUMP. THIS DESIGN HAS THE ADVANTAGE OF PROVIDING TWO DIVERSE DRIVES FOR A COMMON PUMP YET A SINGLE FAILURE, A FAILURE OF THE PUMP, WILL RESULT IN THE LOSS OF BOTH DRIVES.

DIRECT DRIVE DIESEL ENGINE COUPLED TO A PUMP. THIS DESIGN HAS THE ADVANTAGE OF PROVIDING A DRIVE WHICH IS INDEPENDENT OF AC POWER OR STEAM DRIVE. IT, HOWEVER, HAS THE DISADVANTAGES ASSOCIATED WITH DIESEL ENGINES.

QUESTION 5:      B.    HOW WILL THE NEW PUMP FEED WATER TO THE STEAM  
                         GENERATORS AND WILL IT BE FITTED WITH VALVES AND  
                         PIPING TO PROVIDE WATER IN MORE THAN ONE MANNER?

ANSWER:

THE NEW SUFP WILL BE CAPABLE OF FEEDING WATER TO EITHER STEAM  
GENERATOR THROUGH THE MAIN FEEDWATER LINES OR THROUGH THE AUXILIARY  
FEEDWATER LINES. THE SELECTION OF THE FEEDPOINT IS BY MEANS OF  
VALVES PROVIDED WITH THIS TRAIN.

QUESTION 5: C. WILL THE NEW SYSTEM AND PUMP BE SAFETY GRADE?

ANSWER:

NO. ONLY THE PORTION OF THE NEW SYSTEM WHICH CONNECTS TO SAFETY-RELATED SYSTEMS WILL BE SAFETY GRADE UP TO THE FIRST ISOLATION VALVE. THE EXISTING STEAM DRIVEN AFWS IS SAFETY GRADE. IT IS NOT NECESSARY FOR SAFETY OR TO ACHIEVE THE DESIRED INCREASE IN RELIABILITY TO REQUIRE THE THIRD BACKUP TRAIN OF AFWS TO ALSO BE SAFETY GRADE.



QUESTION 5: D. WILL THE NEW SYSTEM BE DESIGNED TO FUNCTION  
DURING A LOSS OF OFFSITE POWER EVENT AND IF  
SO, HOW?

ANSWER:

YES. THE NEW SUFP WILL BE CAPABLE OF BEING MANUALLY LOADED  
ONTO EITHER DIESEL GENERATOR IN THE EVENT OF A LOSS OF OFFSITE  
POWER. THIS IMPROVES RELIABILITY BECAUSE THE OPERATOR CAN SELECT  
THE DIESEL THAT IS FUNCTIONING IN THE EVENT THAT ONE DIESEL  
GENERATOR FAILS FOLLOWING A LOSS OF OFFSITE POWER.

QUESTION 5:     E.    WILL THE NEW SYSTEM BE AUTOMATIC OR MANUALLY  
INITIATED?

ANSWER:

THE NEW SUFP WILL BE MANUALLY INITIATED FROM THE CONTROL ROOM.  
THE STAFF DURING ITS REVIEW OF THE NEW SUFP WILL DETERMINE THE  
NEED FOR AUTOMATIC INITIATION.

QUESTION 6.

DURING THE SUBCOMMITTEE'S JUNE 21, 1985 BRIEFING, HAROLD DENTON, DIRECTOR OF THE OFFICE OF NUCLEAR REACTOR REGULATION, STATED THAT THE PERFORMANCE OF THE FEEDWATER SYSTEMS AT DAVIS-BESSE WAS WORSE THAN THAT PROJECTED BY PROBABILISTIC RISK ASSESSMENT (PRA). WHAT IS THE REASON FOR THIS DISCREPANCY AND WHAT GENERIC IMPLICATIONS DOES THIS HAVE FOR THE DEGREE OF CONFIDENCE ASCRIBED TO PRA?

ANSWER.

MR. DENTON'S COMMENTS REFERRED TO STUDIES WHICH INDICATED THAT THE FREQUENCY OF A LOSS OF MAIN FEEDWATER WAS APPROXIMATELY ONCE PER PLANT-YEAR AND THE UNRELIABILITY OF THE AUXILIARY FEEDWATER SYSTEM WAS IN THE RANGE OF  $10^{-3}$  TO  $10^{-5}$  PER DEMAND. THE FACT IS THAT THE RECENT PERFORMANCE OF THE FEEDWATER SYSTEM AT DAVIS-BESSE WAS INCONSISTENT WITH WHAT WAS EXPECTED. DAVIS-BESSE HAD EXPERIENCED THREE TEMPORARY LOSS OF MAIN FEEDWATER EVENTS DURING THE FIRST SIX MONTHS OF 1985 WHICH WHEN COMBINED WITH THE MULTIPLE COMMON MODE FAILURES OF THE AUXILIARY FEEDWATER SYSTEM ON JUNE 9, 1985 APPEARED TO BE INCONSISTENT WITH A  $10^{-3}$  TO  $10^{-5}$  UNRELIABILITY.

THE NRC HAS RECEIVED TWO PLANT-SPECIFIC RELIABILITY ANALYSES OF THE DAVIS-BESSE AUXILIARY FEEDWATER SYSTEM, THE MOST RECENT DONE BY EDS FOR TOLEDO EDISON AND SUBMITTED TO THE NRC IN DECEMBER 1981. NONE OF THESE ANALYSES COVER MANY OF THE OTHER SYSTEMS OR COMPONENTS INVOLVED IN THE INCIDENT OF JUNE 9. IN ADDITION, THESE ANALYSES DID NOT INCLUDE A DETAILED EXAMINATION OF POTENTIAL COMMON-CAUSE FAILURES NOR A REVIEW OF THE STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM. ALSO, IT IS NOT UNUSUAL FOR OPERATING EXPERIENCE TO HIGHLIGHT NEW POTENTIAL WEAKNESSES IN DESIGN AND OPERATION. IT SHOULD ALSO BE RECOGNIZED THAT THE STARTUP FEEDWATER SYSTEM FUNCTIONED AND THAT THE PRIMARY AUXILIARY FEEDWATER SYSTEM WAS RESTORED WITHIN AN ACCEPTABLE TIME FRAME TO PROVIDE ADEQUATE DECAY HEAT REMOVAL. RECOVERY ACTIONS BY THE OPERATORS MAY ALSO BE PART OF PRA METHODOLOGY.

THIS EVENT DOES NOT HAVE GENERIC IMPLICATIONS ON THE DEGREE OF CONFIDENCE ASCRIBED TO PRA. PRA PROVIDES A LOGICAL FRAMEWORK FOR SYSTEMATICALLY EVALUATING THE DESIGN AND PERFORMANCE OF A SYSTEM OR PLANT. IT HAS ACKNOWLEDGED UNCERTAINTIES ASSOCIATED WITH MODELLING, QUANTIFICATION, AND OMISSIONS. LIKE ANY OTHER EVALUATION PROCESS, THE RESULTS ARE RELATED TO THE SCOPE AND DEPTH OF THE ANALYSIS. PRA PROVIDES A COHESIVE STRUCTURE FOR ASSEMBLING AND ASSESSING THE INFORMATION.

COMMISSIONER ASSELSTINE ADDS:

I AGREE WITH THIS ANSWER, EXCEPT FOR THE COMMISSION'S STATEMENT THAT: "THIS EVENT DOES NOT HAVE GENERIC IMPLICATIONS ON THE DEGREE OF CONFIDENCE ASCRIBED TO PRA." I BELIEVE THAT DAVIS BESSE IS A GOOD TEST OF THE EXTENT TO WHICH ONE CAN HAVE CONFIDENCE IN PRA RESULTS. HERE IS A CASE WHERE TWO INDEPENDENT RELIABILITY ANALYSES WERE PERFORMED ON THE SAME SYSTEM. THE RESULTS DIFFERED BY A FACTOR OF 100 ON THE RELIABILITY OF THE SYSTEM, NOT TAKING INTO ACCOUNT THE UNCERTAINTIES. FURTHER, MR. DENTON RECENTLY WROTE TO THE EDO: "I BELIEVE THAT THE RECENT DAVIS BESSE EVENT ILLUSTRATES THAT, IN THE REAL WORLD, SYSTEM AND COMPONENT RELIABILITIES CAN DEGRADE BELOW THOSE WE AND THE INDUSTRY ROUTINELY ASSUME IN ESTIMATING CORE MELT FREQUENCIES." (SEE ATTACHED LETTER FROM HAROLD R. DENTON TO WILLIAM J. DIRCKS, DATED JUNE 27, 1985).

THERE IS ANOTHER POTENTIALLY REVEALING ASPECT IN THE JUNE 9 DAVIS BESSE EVENT. IT APPEARS THAT THE STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM HAD A SIGNIFICANT ROLE IN CAUSING THE LOSS OF AUXILIARY FEEDWATER. YET, THAT SYSTEM WAS NOT EVEN INCLUDED AS A POSSIBLE CONTRIBUTOR TO THE UNRELIABILITY OF THE AUXILIARY FEEDWATER SYSTEM IN EITHER OF THE INDEPENDENT RELIABILITY ANALYSES. THIS RAISES THE QUESTION OF WHETHER PRA'S ACTUALLY IDENTIFY THE WEAK SPOTS OF SYSTEMS, AS THEIR PROMOTERS FREQUENTLY CLAIM THE DO.

WHILE THE COMMISSION ACKNOWLEDGES THERE ARE UNCERTAINTIES ASSOCIATED WITH PRA'S, IT NOWHERE EXPLAINS HOW TO QUANTIFY THOSE UNCERTAINTIES AND, MORE IMPORTANTLY, THE COMMISSION NOWHERE EXPLAINS HOW TO FACTOR INTO THE DECISIONMAKING PROCESS THOSE UNCERTAINTIES THEY CAN QUANTIFY.

to be done  
at next step  
27 1985 mty.

Attachment  
Question 6

Severe Acidity

JECT: COMMISSION PAPER ON STATION BLACKOUT

ly support the recommendation in the enclosed paper that the  
ion proceed with the proposed rulemaking.

[illegible]

QUESTION 7. AT THE BEGINNING OF THE DAVIS-BESSE INCIDENT ON JUNE 9, 1985, HOW MANY OPERATORS AND WHICH ONES WERE IN THE CONTROL ROOM? DURING THE INCIDENT HOW MANY OPERATORS WERE OUTSIDE THE CONTROL ROOM OR STATIONED POST, WHAT WERE THEIR TITLES, WHY DID THEY LEAVE THE CONTROL ROOM OR STATIONED POST, AND SPECIFICALLY WHAT ACTIONS WERE ACTUALLY ACCOMPLISHED OUTSIDE OF THE CONTROL ROOM, IN LIGHT OF THOSE ACTIONS THAT NEEDED TO BE TAKEN FROM OUTSIDE THE CONTROL ROOM, IS DAVIS-BESSE MEETING GENERAL DESIGN CRITERION 19 WHICH STATES:

"A CONTROL ROOM SHALL BE PROVIDED FROM WHICH ACTIONS CAN BE TAKEN TO OPERATE THE NUCLEAR POWER UNIT SAFELY UNDER NORMAL CONDITIONS AND TO MAINTAIN IT IN A SAFE CONDITION UNDER ACCIDENT CONDITIONS...."

ANSWER.

THE NRC INVESTIGATION OF THE DAVIS-BESSE LOSS OF FEEDWATER INCIDENT IS ONGOING AT THIS TIME. THEREFORE, WE HAVE NOT COMPLETED OUR EVALUATION OF THE IMPLICATIONS OF OPERATOR ACTIONS REQUIRED OUTSIDE OF THE CONTROL ROOM. WE WILL CONSIDER THIS QUESTION IN THE EVALUATION AND INFORM THE SUBCOMMITTEE OF OUR FINDINGS WITH RESPECT TO THE GENERAL DESIGN CRITERIA.



QUESTION 8 ACCORDING TO THE NRC STAFF, FEED AND BLEED PROCEDURES FOR EMERGENCY COOLING WERE INITIATED DURING THE DAVIS-BESSE INCIDENT. NUMEROUS NRC MEMORANDA STATE THAT FEED AND BLEED EMERGENCY COOLING IS NOT CAPABLE OF AVERTING CORE DAMAGE AT DAVIS-BESSE. WHY THEN ARE FEED AND BLEED PROCEDURES PART OF THE NRC APPROVED EMERGENCY OPERATING PROCEDURES FOR THIS PLANT? IN RESPONDING TO THIS QUESTION, PLEASE STATE WHETHER SUCH A PROCEDURE WOULD LENGTHEN THE TIME BEFORE CORE UNCOVERING AND DAMAGE (AND IF SO BY HOW MUCH) AND WHETHER UNDERTAKING SUCH AN EFFORT IS DESIRABLE GIVEN OTHER PRIORITIES FACING THE OPERATORS.

ANSWER

FEED AND BLEED COOLING IS GENERALLY DEFINED AS USE OF THE HIGH PRESSURE INJECTION (HPI) SYSTEM FOR COOLANT INJECTION WITH THE PRESSURIZER PORV AND/OR CODE SAFETY VALVES (PSV) USED FOR ENERGY REMOVAL FROM THE REACTOR COOLANT SYSTEM. THE FEED AND BLEED COOLING MODE IS A BACKUP MEASURE AVAILABLE TO PLANT OPERATORS FOR COPIING WITH EVENTS INVOLVING LOSS OF SECONDARY HEAT REMOVAL CAPABILITY. THIS MODE OF COOLING CAN BE UTILIZED TO KEEP THE CORE COOLED WHILE MEASURES ARE TAKEN TO RESTORE FEEDWATER, AND/OR TO DEPRESSURIZE THE PRIMARY SYSTEM TO A POINT WHERE A LOWER PRESSURE RESIDUAL HEAT REMOVAL (RHR) SYSTEM CAN FUNCTION. BECAUSE DAVIS BESSE HAS NEITHER

SAFETY INJECTION PUMPS (SHUTOFF LEVEL OF ABOUT 1600 PSI) CAPABLE OF INJECTING SUFFICIENT COOLING WATER AT THE SET PRESSURE OF EITHER THE PORV (2425 PSI) OR SAFETY VALVE (2500 PSIA), NOR DOES IT HAVE A PORV RELIEVING CAPACITY CAPABLE OF DEPRESSURIZING THE PRIMARY SYSTEM PRESSURE TO BELOW THE SHUTOFF LEVEL OF THE SAFETY INJECTION PUMPS, IT DOES NOT HAVE A FEED AND BLEED CAPABILITY BASED ON THE ABOVE DEFINITION. THIS HAS BEEN THE BASIS FOR PREVIOUS STAFF STATEMENTS ON THIS ISSUE FOR DAVIS BESSE.

HOWEVER, DAVIS-BESSE DOES HAVE A LIMITED HEAT REMOVAL CAPABILITY UTILIZING CERTAIN COMBINATIONS OF THE HIGH PRESSURE MAKEUP (MU) PUMPS, THE PORV, AND THE STARTUP FEEDWATER PUMP. FOR EXAMPLE, SIMPLIFIED MASS AND ENERGY CALCULATIONS HAVE BEEN PERFORMED BY THE STAFF WHICH INDICATE THAT, IF FEEDWATER WAS NOT RESTORED DURING THE DAVIS-BESSE EVENT, THE CORE MIGHT NOT EXPERIENCE FUEL DAMAGE IF BOTH MAKEUP PUMPS HAD BEEN STARTED. HOWEVER, WHILE THESE SIMPLIFIED CALCULATIONS ARE NOT SUFFICIENTLY ACCURATE TO FULLY DEMONSTRATE THAT CONTINUED CORE COOLING WOULD HAVE BEEN MAINTAINED, THEY ARE SUFFICIENT TO CONCLUDE THAT THE LIMITED MAKEUP PUMP FLOW WOULD AT LEAST SUBSTANTIALLY EXTEND THE TIME BEFORE CORE DAMAGE OCCURS, OR PREVENT IT ALTOGETHER.

STAFF CALCULATIONS HAVE ALSO BEEN PERFORMED WHICH EXAMINE THE USE OF THE MAKEUP PUMPS IN A FEED AND BLEED MODE THROUGH THE PRESSURIZER CODE SAFETY VALVES ASSUMING INITIAL PLANT OPERATION AT 100% POWER. THESE CALCULATIONS INDICATE THAT IF THE FEED AND BLEED MODE IS INITIATED IN APPROXIMATELY FIVE (5) MINUTES, UNCOVERY OF THE TOP OF THE CORE MIGHT NOT OCCUR. HOWEVER, IF INITIATED TWENTY (20) MINUTES INTO A LOSS OF ALL FEEDWATER EVENT, THE CORE WOULD START TO UNCOVER IN APPROXIMATELY 150 MINUTES IF FEEDWATER IS NOT RESTORED. THIS IS IN CONTRAST TO A CORE UNCOVERY IN APPROXIMATELY 55 MINUTES IF NO ACTION TO START THE MAKEUP PUMPS IS TAKEN.

AS PART OF THE ABNORMAL TRANSIENT OPERATOR GUIDELINES (ATOG) SUBMITTAL, DEVELOPED IN RESPONSE TO ITEM I.C.1 OF NUREG-0737, THE LICENSEE HAS EXAMINED THE CAPABILITY OF FEED AND BLEED COOLING TO MITIGATE A LOSS OF ALL FEEDWATER EVENT. THESE ANALYSES CONCLUDED THAT IF THE OPERATOR STARTS BOTH MU PUMPS AND OPENS THE PORV IN APPROXIMATELY FIVE (5) MINUTES, THE REACTOR COOLANT SYSTEM WOULD DEPRESSURIZE, SUFFICIENT FOR THE HPI SYSTEM TO INJECT COOLANT INTO THE PRIMARY SYSTEM AND THE CORE WOULD REMAIN COVERED AND ADEQUATELY COOLED. IN ADDITION, ANALYSES HAVE BEEN PROVIDED BY TOLEDO EDISON COMPANY, THE OWNERS OF THE DAVIS BESSE PLANT, WHICH SHOW THAT BY STARTING ONE MU PUMP, OPENING THE PORV AND INITIATING THE STARTUP FEEDWATER PUMP WITHIN 30 MINUTES, THE CORE WOULD REMAIN COVERED AND WOULD BE ADEQUATELY COOLED.

GENERIC GUIDELINES HAVE BEEN DEVELOPED BASED ON THE ANALYSES DESCRIBED ABOVE. PERMISSION TO IMPLEMENT PROCEDURES WHICH CALL FOR THE OPERATOR TO INITIATE VARIOUS OPTIONS FOR EMERGENCY COOLING OF THE REACTOR CORE WAS INDICATED IN NUREG-0737, ITEM I.C.1. AND THE SER ON THE B&W ATOG PROGRAM. STAFF REVIEW OF THE DAVIS-BESSE PROGRAM FOR GENERATING PLANT-SPECIFIC PROCEDURES FOR THIS AND OTHER EVENTS BASED ON ATOG IS ONGOING.

FINALLY, YOU EXPRESSED A CONCERN AS TO WHETHER UNDERTAKING SUCH AN EFFORT IS DESIRABLE GIVEN OTHER PRIORITIES FACING THE OPERATORS. THE FIRST ACTIONS FOR THE OPERATOR UPON RECOGNITION OF THE LOSS OF ALL FEEDWATER IS TO ACTUATE THE MU AND HPI PUMPS AND OPEN THE PORV. THESE ACTIONS CAN BE PERFORMED READILY FROM THE CONTROL ROOM. THE OPERATOR IS THEN DIRECTED TO RESTORE FEEDWATER TO THE STEAM GENERATORS UTILIZING EITHER THE AUXILIARY FEEDWATER SYSTEMS, THE MAIN FEEDWATER SYSTEM OR THE STARTUP FEEDWATER PUMP. THE STAFF HAS CONCLUDED THAT THIS APPROACH IS PRUDENT, AND DOES NOT PLACE AN UNDUE BURDEN ON THE OPERATORS BECAUSE THE FEED AND BLEED COOLING MODE USING MAKEUP PUMPS AND OPENING THE PORV IS SIMPLE TO INITIATE. IMPLEMENTATION OF THESE ACTIONS WILL PROVIDE A LONGER TIME PERIOD FOR FEEDWATER RECOVERY, INCLUDING POTENTIAL USE OF THE STARTUP FEEDWATER PUMP, BEFORE CORE UNCOVERY WOULD OCCUR, AND MAY ALSO PREVENT SIGNIFICANT CORE DAMAGE IF FEEDWATER IS NEVER RECOVERED.

COMMISSIONER ASSELSTINE ADDS:

AS INDICATED IN MY RESPONSE TO QUESTION 1.C, THE STAFF WAS  
UNABLE TO DEMONSTRATE THE EFFICACY OF THE FEED AND BLEED PROCESS  
IN THE TMI-1 RESTART PROCEEDING.

QUESTION 9 GIVEN THE DEMONSTRATED LIKELIHOOD OF A TOTAL LOSS OF FEEDWATER AT B&W REACTORS, WHY DOES THE NRC CONSIDER THIS TYPE OF INCIDENT TO BE BEYOND THE DESIGN BASIS? WHAT IS THE TECHNICAL BASIS FOR CONSIDERING A POTENTIAL FAILURE OR ACCIDENT SEQUENCE AS A DESIGN BASIS EVENT?

ANSWER

THE TECHNICAL BASIS FOR CONSIDERING A POTENTIAL FAILURE OR ACCIDENT SEQUENCE AS A DESIGN BASIS EVENT IS MULTIFACETED AND TAKES INTO ACCOUNT BOTH DETERMINISTIC AS WELL AS PROBABILISTIC CONSIDERATIONS.

DESIGN BASIS EVENTS ARE DEVELOPED BY EXAMINING FAILURES THAT CAN OCCUR DURING PLANT OPERATION. THE EXAMINATION OF THE POTENTIAL FAILURES ALLOWS THE FAILURES TO BE CATEGORIZED BY EVENT TYPE AND ESTIMATED FREQUENCY OF OCCURRENCE. EVENT TYPES INCLUDE UNDERCOOLING EVENTS, OVERCOOLING EVENTS, REACTIVITY INSERTION EVENTS, AND LOSS-OF-COOLANT EVENTS. EVENTS WHICH ARE CONSIDERED TO HAVE A FREQUENCY OF OCCURRENCE OF AT LEAST ONCE IN THE PLANT LIFETIME ARE CATEGORIZED AS "ANTICIPATED OPERATIONAL OCCURRENCES," WHEREAS EVENTS WHICH ARE NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME, BUT NEVERTHELESS ARE NOT CONSIDERED IMPROBABLE ENOUGH TO OMIT FROM THE DESIGN BASIS, ARE CATEGORIZED AS "POSTULATED ACCIDENTS."

A COMMON CHARACTERISTIC OF BOTH ANTICIPATED OPERATIONAL OCCURRENCES AND POSTULATED ACCIDENTS IS THAT THEY ARE ASSUMED TO BE CAUSED BY A SINGLE MALFUNCTION AS THE INITIATING EVENT. FOR POSTULATED ACCIDENTS, THE STAFF ALSO REQUIRES THAT THE PLANT BE ABLE TO ACCOMMODATE THE WORST SINGLE ACTIVE FAILURE IN CONJUNCTION WITH THE INITIATING EVENT, AND THAT OFFSITE POWER MUST BE ASSUMED LOST PER THE REQUIREMENT OF GENERAL DESIGN CRITERION 17, CONTAINED IN APPENDIX A TO 10CFR50. THE WORST SINGLE ACTIVE FAILURE ASSUMPTION IS A DETERMINISTIC REQUIREMENT NOT DIRECTLY RELATED TO PROBABILISTIC CONSIDERATIONS. MOREOVER, THE FAILURE ASSUMED IS LIMITED TO SYSTEMS AND EQUIPMENT THAT ARE PREDICTED OR ASSUMED TO OPERATE DURING THE EVENT. SPURIOUS OR RANDOM FAILURES ARE NOT POSTULATED.

IN THE CASE OF DAVIS-BESSE, THE PLANT WAS DESIGNED FOR A LOSS-OF-MAIN FEEDWATER AS THE INITIATING EVENT, AND THE SINGLE FAILURE ASSUMPTION REQUIRED POSTULATION OF ONE AUXILIARY FEEDWATER PUMP FAILING TO START. THUS, THIS EVENT WOULD DETERMINE THE DESIGN BASIS FOR THE AUXILIARY FEEDWATER PUMP CAPACITY; NAMELY, THAT ONE AUXILIARY FEEDWATER PUMP BE CAPABLE OF SUPPLYING SUFFICIENT WATER TO REMOVE ALL SHUTDOWN DECAY HEAT.

MULTIPLE FAILURES ARE NOT CONSIDERED BECAUSE OTHER NRC DESIGN CRITERIA REQUIRE THAT SYSTEMS BE DESIGNED WITH SUFFICIENT RELIABILITY THAT MULTIPLE FAILURES ARE OF SUFFICIENTLY LOW PROBABILITY THAT THEY NEED NOT BE CONSIDERED IN THE DESIGN BASE. THUS, IF MULTIPLE MALFUNCTIONS ARE UNCOVERED THAT CAN BE EXPECTED TO OCCUR DURING INITIATING EVENTS, AND LEAD TO A LOSS OF A SAFETY FUNCTION AND UNACCEPTABLE CONSEQUENCES, THE NRC POSITION IS TO REQUIRE THAT THE SYSTEM DESIGN BE UPGRADED AND CORRECTED AS NECESSARY SUCH THAT THE MULTIPLE FAILURES CANNOT OCCUR.

THE CAUSES CONTRIBUTING TO THE LOSS OF ALL AUXILIARY FEEDWATER ARE BEING REVIEWED BY THE NRC FACT FINDING TEAM. THEREFORE, IT IS CURRENTLY PREMATURE TO DETERMINE THE EXACT NATURE OF THE CORRECTIVE ACTION THAT MAY BE TAKEN TO ENSURE THAT THE DESIGN BASIS REMAINS ACCEPTABLE.

COMMISSIONER ASSELSTINE ADDS:

THE THRUST OF THIS QUESTION IS: DO THE NRC REGULATIONS MAKE SENSE? I THINK NOT. THE REGULATIONS HAVE BEEN DEVELOPED BASED ON MANY IMPLICIT AND EXPLICIT ASSUMPTIONS AND ARE KNOWN TO BE INCOMPLETE. IN FACT, 10 CFR PART 50, APPENDIX A - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS, WHICH WAS PROMULGATED ON FEBRUARY 20, 1971, STATES:



"THE DEVELOPMENT OF THESE GENERAL DESIGN CRITERIA IS NOT YET COMPLETE. FOR EXAMPLE, SOME OF THE DEFINITIONS NEED FURTHER AMPLIFICATION. ALSO, SOME OF THE SPECIFIC DESIGN REQUIREMENTS FOR STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY HAVE NOT AS YET BEEN SUITABLY DEFINED."

THE COMMISSION HAS YET TO COME TO GRIPS WITH IMPORTANT ISSUES SUCH AS SINGLE FAILURES OF PASSIVE COMPONENTS, AND THE MINIMUM ACCEPTABLE REDUNDANCY AND DIVERSITY OF SUBSYSTEMS AND THE REQUIRED INDEPENDENCE OF THE SUBSYSTEMS. RATHER, IT HAS ASSUMED THAT IT HAS DEVELOPED THE NECESSARY AND SUFFICIENT DESIGN BASIS EVENTS AND ACCIDENTS TO BRACKET ALL CREDIBLE SCENARIOS. I BELIEVE THAT THE JUNE 9, 1985 EVENT AT DAVIS BESSE, THE TMI-2 ACCIDENT, AS WELL AS OTHER OPERATIONAL OCCURRENCES, DRAW INTO QUESTION CONTINUED RELIANCE ON THE REGULATORY APPROACH DESCRIBED BY THE COMMISSION ABOVE. I WOULD ALSO POINT OUT THAT THE AGENCY COMMITTED TO A REEXAMINATION OF THE SINGLE FAILURE CRITERION AFTER THE TMI-2 ACCIDENT BUT LITTLE PROGRESS HAS BEEN MADE.

QUESTION 10. LIST EVERY REGULATION, REQUIREMENT, RECOMMENDATION, GENERAL DESIGN CRITERIA, STANDARD REVIEW PLAN ITEM, BRANCH TECHNICAL POSITION, CONFIRMATORY ACTION LETTER ITEM, LICENSING COMMITMENT, TMI ACTION PLAN ITEM, NUREG-0667 RECOMMENDATION, OR ANY OTHER ITEM THAT DAVIS-BESSE WAS NOT IN FULL AND COMPLETE COMPLIANCE WITH ON JUNE 9, 1985.

ANSWER

THE LISTING IN THIS QUESTION COVERS BOTH REQUIREMENTS AND NON-REQUIREMENTS WHICH ARE SEPARATELY DISCUSSED BELOW. OUR ANSWER TO THIS QUESTION IS BASED ON A BRIEF REVIEW OF THE DOCKET INCLUDING INSPECTION RECORDS AS OF JUNE 9, 1985. SINCE OUR INVESTIGATION OF THE JUNE 9, 1985 INCIDENT OF DAVIS-BESSE IS NOT COMPLETE, WE NOTE THAT OTHER ITEMS MAY BE ADDED WHEN THE INVESTIGATION IS CONCLUDED. IT SHOULD BE ALSO NOTED THAT THERE ARE A NUMBER OF OPEN AND ONGOING INSPECTION ISSUES THAT HAVE NOT BEEN CLOSED OUT. IT IS POSSIBLE THAT ITEMS OF NONCOMPLIANCE MAY BE IDENTIFIED IN SUBSEQUENT FOLLOW-UP.

1. REGULATION - THE SAFETY EVALUATION REPORT (SER) ISSUED AT THE TIME THE OPERATING LICENSE WAS GRANTED MADE SPECIFIC FINDINGS THAT THE REGULATIONS WERE SATISFIED. AT THAT TIME, ONE EXEMPTION TO THE REGULATIONS WAS GRANTED. A SUBSEQUENT REVISION TO APPENDIX J ELIMINATED THE NEED FOR THIS EXEMPTION. SINCE THAT TIME EACH NEW REGULATION ENACTED HAS BEEN SPECIFICALLY BACKFITTED TO DAVIS-BESSE, E.G., APPENDIX R ON FIRE PROTECTION. THREE EXEMPTIONS FOR NEW REGULATIONS WERE OUTSTANDING ON JUNE 9, 1985 (SEE TABLE).

IN JULY 1983, AN AUDIT/INSPECTION WAS CONDUCTED AT THE DAVIS-BESSE FACILITY TO DETERMINE COMPLIANCE TO APPENDIX R TO 10 CFR 50.48 (FIRE PROTECTION). NUMEROUS AREAS OF NON-COMPLIANCE WERE REVEALED BY THE INSPECTION. THE LICENSE HAS SCHEDULED CORRECTIVE ACTIONS TO BRING THE FACILITY INTO COMPLIANCE BY 1989.

2. REQUIREMENT - A REQUIREMENT MUST DERIVE FROM A SPECIFIC BINDING ITEM SUCH AS A REGULATION, ORDER OR, LICENSE CONDITION. THE DAVIS-BESSE PLANT WAS NOT IN COMPLIANCE ONLY WITH THE ITEMS DISCUSSED ABOVE IN REGULATIONS AS OF JUNE 9, 1985.
3. RECOMMENDATION - AN APPLICANT OR LICENSEE IS NOT REQUIRED TO BE IN COMPLIANCE WITH A RECOMMENDATION. THE NRC DOES NOT MONITOR RESULTS OF ALL RECOMMENDATIONS MADE TO A LICENSEE. IF THE MATTER HAS SUFFICIENT WEIGHT TO DEMAND COMPLIANCE THEN IT MUST BE COMMUNICATED IN A BINDING WAY TO BECOME A REQUIREMENT.
4. GENERAL DESIGN CRITERIA - THE GENERAL DESIGN CRITERIA ARE IN THE REGULATIONS AND CONSIDERED PART OF THEM; THE REMARKS ABOVE ON REGULATIONS APPLY.
5. STANDARD REVIEW PLAN - THE STANDARD REVIEW PLAN (SRP) HAS THE FORCE OF A REGULATORY GUIDE, NOT THE FORCE OF REGULATION. APPLICANTS FOR LICENSES CAN SATISFY THE STAFF'S CURRENT INTERPRETATION OF THE REQUIREMENTS OF THE REGULATIONS BY THE MEANS DESCRIBED IN THE SRP. ALTERNATIVES TO THE SRP

ARE EVALUATED IN THE SER TO PROVIDE A BASIS FOR DETERMINING THE ACCEPTABILITY OF THE ALTERNATIVE. THUS, A REQUIREMENT MAY BE SATISFIED WITHOUT CONFORMING WITH THE SRP. THE DAVIS-BESSE PLANT OPERATING LICENSE REVIEW WAS NOT REQUIRED TO BE EVALUATED USING SRP CRITERIA BECAUSE THE REVIEW WAS STARTED SEVERAL YEARS BEFORE SRP WAS ISSUED. A COMPLETE LISTING OF THESE NON-CONFORMANCES WITH THE SRP WOULD BE TIME-CONSUMING AND DIFFICULT TO GENERATE SINCE IT WOULD REQUIRE A REVIEW OF ALL SERS ON DAVIS-BESSE AND A CONSISTENT DETERMINATION OF WHAT IS NON-CONFORMING WITH THE SRP. NORMALLY THE SRP IS USED AS A GAUGE OF ACCEPTABILITY WHEN THE NRC IS CONSIDERING AN UPGRADING OF REQUIREMENTS E.G., ITEM II.E.1.1 OF NUREG-0737.

6. BRANCH TECHNICAL POSITION - THE BRANCH TECHNICAL POSITIONS ARE ONE STEP LESS FORMAL THAN AN EQUIVALENT ELEMENT OF THE SRP, AND THE REMARKS ABOVE ON SRP ITEMS APPLY.
7. CONFIRMATORY ACTION LETTER (CAL) - THERE IS ONLY ONE CAL, REGARDING SNUBBERS AND PIPE RESTRAINTS, THAT WAS ACTIVE FOR THE DAVIS-BESSE PLANT ON JUNE 9, 1985. THERE WERE NO ITEMS IN THE CAL FOR WHICH THE DAVIS-BESSE PLANT WAS NOT IN COMPLIANCE ON JUNE 9, 1985.
8. LICENSING COMMITMENT - AS OF JUNE 9, 1985, TO THE BEST OF OUR KNOWLEDGE, THE LICENSEE WAS IN COMPLIANCE WITH ALL LICENSING COMMITMENTS.

9. TMI ACTION PLAN ITEM - THE TMI ACTION PLAN ITEMS WERE HANDLED AS NRC REQUIREMENTS FOR WHICH CONFIRMATORY ORDERS WERE ISSUED REGARDING THE LICENSEE'S SCHEDULAR COMMITMENTS. THE LICENSEE IS IN COMPLIANCE WITH THESE ORDERS. "ALTHOUGH SOME TMI ACTION ITEMS HAVE NOT BEEN COMPLETED, ALL REVIEWS ARE PROCEEDING ON SCHEDULE WHICH IS ACCEPTABLE TO THE STAFF."
10. NUREG-0667 RECOMMENDATION - THE REMARKS ON RECOMMENDATION (NUMBER 3) APPLY HERE ALSO. THE STATUS OF NUREG-0667 RECOMMENDATIONS IS DISCUSSED IN THE ANSWER TO QUESTION 14, WHICH LISTS THE RECOMMENDATIONS THAT HAVE BEEN MADE NRC REQUIREMENTS. THERE ARE NO NRC REQUIREMENTS STEMMING FROM THE NUREG-0667 RECOMMENDATIONS FOR WHICH THE DAVIS-BESSE WAS NOT IN COMPLIANCE ON JUNE 9, 1985.

TABLE Q-10

OUTSTANDING EXEMPTIONS FROM 10 CFR

<u>REGULATION</u>	<u>SUBJECT, DATE GRANTED</u>
10 CFR 50 APPENDIX R	TECHNICAL EXEMPTION TO APPENDIX R, SECTION III.G, NOVEMBER 23, 1982
10 CFR 50 § 50.44(c)(3)(III)	SCHEDULAR EXEMPTION TO FIRST REFUELING OUTAGE AFTER 1985 FOR VESSEL HEAD VENT, SEPTEMBER 7, 1983
10 CFR 50	TECHNICAL EXEMPTIONS TO APPENDIX R SECTIONS III.G, III.O, AND III.L, AUGUST 20, 1984



QUESTION 11. ACCORDING TO A JUNE 19, 1985 MEMORANDUM FROM C. J. HELTEMES JR., DIRECTOR OF THE OFFICE OF ANALYSIS AND EVALUATION OF OPERATIONAL DATA, TO WILLIAM J. DIRCKS, EXECUTIVE DIRECTOR FOR OPERATIONS, THE SAFETY PARAMETER DISPLAY SYSTEM (SPDS) WAS INOPERABLE AT TIME OF THE DAVIS-BESSE INCIDENT. THE SPDS SYSTEM WAS TO BE INSTALLED AT DAVIS-BESSE IN ACCORDANCE WITH THE TMI ACTION PLAN IN ORDER TO PROVIDE REACTOR OPERATORS WITH MINIMUM INFORMATION WHICH DEFINES THE SAFETY PARAMETERS OF THE PLANT. PLEASE STATE:

(A) WHEN THIS SYSTEM WAS TO HAVE BEEN OPERATIONAL ACCORDING TO THE TMI ACTION PLAN;

ANSWER

THE RELEVANT ACTION PLAN GUIDANCE IS ITEM I.D.2, SAFETY PARAMETER DISPLAY SYSTEM, DESCRIBED IN NUREG 0737, SUPPLEMENT 1, CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS. THIS ITEM WAS NOT GIVEN A SINGLE, GENERIC DUE DATE. THE PLANT SPECIFIC DUE DATE FOR DAVIS BESSE WAS NOVEMBER 30, 1984 AS DOCUMENTED IN AN NRC CONFIRMATORY ORDER DATED FEBRUARY 21, 1984.

MARKEY/IE

7/8/85

QUESTION 11. (B) WHEN IT BECAME OPERATIONAL AT DAVIS BESSE;

ANSWER.

AT DAVIS BESSE, THE PLANT WAS IN A SHUTDOWN STATUS ON NOVEMBER 30, 1984. THE SAFETY PARAMETER DISPLAY SYSTEM WAS MADE OPERABLE BY THE TIME OF THE NEXT STARTUP ON JANUARY 1, 1985. THE NRC STAFF HAD PREVIOUSLY ACCEPTED THIS CHANGE IN SCHEDULE AS DOCUMENTED IN AN NRC LETTER TO THE LICENSEE ON JUNE 5, 1984.

MARKEY/IE

7/8/85



QUESTION 11. (C) WHY THIS SYSTEM WAS INOPERABLE DURING THE  
DAVIS BESSE INCIDENT; AND

ANSWER.

AT DAVIS BESSE THERE ARE TWO COMPUTERS CAPABLE OF OPERATING THE SAFETY PARAMETER DISPLAY SYSTEM. ONE COMPUTER HAD BEEN UNABLE TO OPERATE THE SPDS FOR SOME TIME DUE TO INTERMITTENT NOISE PROBLEMS. THE SECOND COMPUTER BECAME UNABLE TO OPERATE THE SPDS AT 3:30 P.M. ON JUNE 8, 1985 DUE TO FAILURE OF A CABINET COOLING FAN AND/OR NOISE PROBLEMS. THE SYSTEM REMAINED IN THIS STATUS UNTIL THE TIME OF THE INCIDENT ON JUNE 9, 1985. (IT SHOULD BE NOTED THAT THE SPDS IS CONSIDERED TO BE AN AID TO OPERATORS IN ACCIDENT ASSESSMENT, AND THERE IS NO TECHNICAL SPECIFICATION REQUIREMENT THAT SPDS BE AVAILABLE IN ORDER TO CONTINUE PLANT OPERATION. OTHER SAFETY GRADE INSTRUMENTS ARE REQUIRED TO BE AVAILABLE. FURTHER, PROCEDURES TO DIAGNOSE CONDITIONS ARE REQUIRED, INCLUDING PROCEDURES THAT DO NOT REQUIRE INFORMATION FROM THE SPDS.)

MARKEY/IE

REVISED 7/16/85

QUESTION 11. (D) WILL IT BE OPERABLE PRIOR TO RESTART?

ANSWER.

THE SYSTEM HAS BEEN MADE OPERABLE WITH ONE COMPUTER. THE COOLING FAN HAS BEEN REPLACED BUT THE SOURCE OF NOISE HAS NOT BEEN IDENTIFIED. THE NRC WILL PURSUE RELIABILITY IMPROVEMENT OF THE SYSTEM PRIOR TO STARTUP. ACCORDINGLY, THE SYSTEM WILL LIKELY BE OPERABLE AT THE TIME OF RESTART. HOWEVER, AS NOTED IN THE RESPONSE TO 11C, THERE IS NO REQUIREMENT THAT THE SPDS BE OPERABLE IN ORDER TO START UP THE PLANT.

MARKEY/IE

REVISED 7/16/85

QUESTION 12: LIST SEPARATELY THOSE PLANTS THAT DO NOT HAVE AUXILIARY FEEDWATER SYSTEMS THAT ARE FULLY SAFETY GRADE, THOSE THAT HAVE ONLY TWO AUXILIARY FEEDWATER PUMPS AND THOSE THAT HAVE AUXILIARY FEEDWATER PUMPS POWERED ONLY BY ONE SOURCE.

ANSWER:

THE ANSWER WE ARE PROVIDING TO THIS QUESTION IS LIMITED TO B&W DESIGNED PLANTS BECAUSE OF THE TIME AVAILABLE TO RESPOND; HOWEVER, THE QUESTION COULD BE APPLIED TO OTHER VENDOR DESIGNS. THE B&W OPERATING PLANTS WERE DESIGNED PRIOR TO THE ISSUANCE OF THE STANDARD REVIEW PLAN (1975). PRIOR TO THAT TIME, THE AUXILIARY FEEDWATER SYSTEMS (AFWS) WERE NOT DESIGNED TO SAFETY GRADE STANDARDS BUT RATHER WERE TO GOOD ENGINEERING PRACTICE. THE AFWS WAS NOT CONSIDERED A PART OF THE NSSS SYSTEM AND, THEREFORE, WAS NOT SUPPLIED BY THE NSSS VENDOR BUT WAS SUPPLIED BY THE UTILITY. SUBSEQUENT TO THE ACCIDENT AT TMI-2, THE B&W PLANTS WERE SHUT DOWN AND CERTAIN MODIFICATIONS WERE MADE PRIOR TO THEIR RESTART. SINCE THE RESTARTS, THE STAFF HAS CONTINUED WORKING WITH THE LICENSEES TO UPGRADE THEIR AFWS BASED ON THE REQUIREMENTS DESCRIBED IN THE TMI ACTION PLAN NUREG-0737 ITEMS II.E.1.1 AND II.E.1.2. THE STATUS OF THE B&W PLANTS REGARDING CURRENT AFWS CRITERIA IS PROVIDED IN TABLE 1. THE ONLY OPERATING B&W PLANT THAT HAS AN AUXILIARY

FEEDWATER SYSTEM CONSIDERED SAFETY GRADE IS DAVIS-BESSE. FOUR B&W PLANTS HAVE MADE COMMITMENTS TO IMPLEMENT IMPROVEMENTS TO THE AFW SYSTEMS BASED ON REVIEWS USING THE STANDARD REVIEW PLAN CRITERIA. THE THREE OCONEE UNITS AFW SYSTEMS ARE STILL UNDER STAFF REVIEW REGARDING THEIR SEISMIC DESIGN CAPABILITY. THE LIST OF THOSE B&W PLANTS THAT HAVE ONLY 2 AFWS PUMPS AND THOSE THAT HAVE AFWS PUMPS POWERED BY ONLY ONE SOURCE IS PROVIDED IN TABLES 2 AND 3, RESPECTIVELY.

COMMISSIONER ASSELSTINE ADDS:

I FIND IT VERY DISQUIETING THAT THE TECHNICAL STAFF CHARACTERIZES IN TABLE 1 THE TMI-1 EMERGENCY FEEDWATER SYSTEM AS ONE OF "LOW" RELIABILITY, BASED ON A PLANT-SPECIFIC ANALYSIS. GIVEN THAT FEED AND BLEED CAPABILITY HAS NOT BEEN DEMONSTRATED AT TMI-1, I MUST NOW QUESTION THE ADEQUACY OF THE DESIGN OF THE TMI-1 EMERGENCY FEEDWATER SYSTEM.

TABLE 1

## AUXILIARY FEEDWATER SYSTEM

Operating Plants	SRP 10.49 Compliance	NPA C-14 Compliance	NUREG-0737 TAP-II.E.1.1 Compliance	NUREG-0737 TAP-II.E.1.2 Compliance	Reliability Criterion Compliance
Arkansas-1	Committed to upgrade system for seismic capability & tornado missile protection.	Committed to upgrade system for compliance.	Yes	Yes	Low (Unavailability $>10^{-3}$ ); Proposed new design will raise it to medium (Unavail.: $10^{-3}$ - $10^{-4}$ ).
Crystal River-3	Committed to upgrade system for tornado missile, int. generated missile, single failure vulnerability & auto. initiation & control.	Committed to upgrade system for compliance.	Yes	Yes	Medium (Unavailability: $10^{-3}$ - $10^{-4}$ ).
Davis Besse-1	Committed to upgrade system for improved reliability & power diversity.	System meets the intent.	Yes But see response to questions 2 & 5.	Yes	Low (Unavailability $>10^{-3}$ ).
Oconee 1, 2, & 3	Seismic capability & tornado missile protection for the system not demonstrated.	Seismic capability not demonstrated.	Yes	Yes	Medium (Unavailability: $10^{-3}$ - $10^{-4}$ ).

TABLE 1

## AUXILIARY FEEDWATER SYSTEM (Cont'd.)

Operating Plants	SRP 10.49 Compliance	MPA C-14 Compliance	NUREG-0737 TAP-II.E.1.1 Compliance	NUREG-0737 TAP-II.E.1.2 Compliance	Reliability Criterion Compliance
Rancho Seco	Committed to upgrade auto. initiation as a safety grade system.	System meets the intent.	Compliance complete by June 1986	Yes	Medium (Unavailability: $10^{-3}$ - $10^{-4}$ ).
TMI-1	Committed to upgrade auto. initiation as a safety grade system at refueling after restart.	System meets the intent. Add'l. seismic upgrade committed at refueling after restart.	Short term modifications completed. Long term modifications to be completed at refueling after restart.	Short term modifications completed. Long term modifications to be completed at refueling after restart.	Low (Unavailability $>10^{-3}$ ). (This reliability is based upon a plant specific analysis) After upgrades reliability will be medium.

TABLE 2

## B&amp;W PLANTS WITH TWO AFW PUMPS

Operating Plant	Pump Data			Remarks
	No. of Pumps	Type of Drive		
		Steam	Electric	
Arkansas-1	2	1	1	
Crystal River-3	2	1	1	
Davis Besse-1	2	2	-	Committed to install 3rd motor driven pump
Rancho Seco	2	* 1	1	* Dual drive: steam & motor

TABLE 3

B&W PLANT(S) WITH AFW PUMPS  
POWERED BY ONE SOURCE

Operating Plant	No. of Pumps	Type of Drive	Remarks
Davis Besse-1	2	Steam	Committed to install third motor driven pump

QUESTION 13. WHAT, IF ANY, PLANTS ARE NOT IN FULL COMPLIANCE WITH THE TMI ACTION PLAN ITEMS RELATING TO AUXILIARY FEEDWATER SYSTEMS?

ANSWER.

THE ANSWER WE ARE PROVIDING TO THIS QUESTION IS LIMITED TO B&W DESIGNED PLANTS. FIRST, THE SPECIFIC REQUIREMENTS ARE LISTED. THEN THE IMPLEMENTATION STATUS IS DISCUSSED.

THE RELEVANT ACTION PLAN ITEMS ARE ITEM II.E.1.1, AUXILIARY FEEDWATER SYSTEM EVALUATION, AND ITEM II.E.1.2, AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION, AS DESCRIBED IN NUREG 0737, CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS. THESE ITEMS ARE BRIEFLY SUMMARIZED BELOW.

#### II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

PROVIDE EVALUATIONS FOR NRC STAFF REVIEW AND IMPLEMENT SHORT-TERM AND LONG-TERM MODIFICATIONS DETERMINED TO BE NECESSARY.

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II.E.1.2 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION  
AND FLOW INDICATION.

1. INITIATION

A. IMPLEMENT CONTROL-GRADE (SHORT-TERM) AUTOMATIC  
INITIATION.

B. IMPLEMENT SAFETY-GRADE (LONG-TERM) AUTOMATIC INITIA-  
TION AND SUBMIT DOCUMENTATION FOR NRC STAFF REVIEW.

2. FLOW INDICATION

A. IMPLEMENT CONTROL-GRADE (SHORT-TERM) INDICATION.

B. IMPLEMENT LESSONS LEARNED, CATEGORY A TECHNICAL  
SPECIFICATIONS. (NRC ACTION TO CHANGE TECHNICAL SPECI-  
FICATIONS FOLLOWING RECEIPT AND REVIEW OF LICENSEE  
PROPOSALS.)

C. IMPLEMENT SAFETY-GRADE (LONG-TERM) INDICATION AND  
SUBMIT DOCUMENTATION FOR NRC STAFF REVIEW.

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WITH THE ABOVE REQUIREMENTS IN MIND, THE IMPLEMENTATION STATUS FOR B&W DESIGNED PLANTS IS PROVIDED BELOW.

REGARDING LICENSEE EVALUATIONS AND COMMITMENTS, LICENSEES HAVE NOW PROVIDED EVALUATIONS AND COMMITMENTS WHICH, AFTER NRC STAFF REVIEW, WERE FOUND ACCEPTABLE.

AS A RESULT OF SHORT TERM LESSONS LEARNED IMPLEMENTATION TECHNICAL SPECIFICATIONS CONCERNING AUXILIARY FEEDWATER FLOW INDICATION ARE IN PLACE. AS LONG-TERM MODIFICATIONS ARE COMPLETED, THE TECHNICAL SPECIFICATIONS ARE BEING REVIEWED AGAIN TO DETERMINE WHETHER OR NOT FURTHER CHANGES ARE NEEDED.

WITH REGARD TO LICENSEE IMPLEMENTATION OF SYSTEM MODIFICATIONS, THE INCOMPLETE AREAS ARE LISTED BELOW.

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QUESTION 13. (CONTINUED)

-4-

PLANT NAME

INCOMPLETE AREAS

ARKANSAS 1

WORK IS COMPLETE ON THE  
ACTION PLAN ITEMS.  
IN RELATED MODIFICATIONS THE  
LICENSEE PLANS TO UPGRADE THE  
CONDENSATE STORAGE TANK IN 1986.

CRYSTAL RIVER 3

WORK IS COMPLETE ON THE  
ACTION PLAN ITEMS.  
IN RELATED MODIFICATIONS MOST  
WORK WILL BE COMPLETED DURING  
THE CURRENT OUTAGE AND THE  
CONTROL SYSTEM WILL BE UPGRADED.  
ONE UPGRADED TANK IS TO BE  
INSTALLED IN 1986.

DAVIS BESSE

WORK IS COMPLETE ON THE ACTION  
PLAN ITEMS.  
IN RELATED MODIFICATIONS, THE  
LICENSEE PLANS TO INSTALL A NEW  
PUMP/TRAIN AT THE NEXT REFUELING  
OUTAGE.

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QUESTION 13. (CONTINUED)

-5-

PLANT NAME

INCOMPLETE AREAS

OCONEE 1, 2 AND 3

WORK IS COMPLETE ON THE ACTION PLAN ITEMS. A RELATED QUESTION REGARDING SEISMIC AND TORNADO PROTECTION OF THE AUXILIARY FEEDWATER SYSTEM IS BEING PURSUED AS A PLANT SPECIFIC TOPIC.

RANCHO SECO

MOST WORK HAS BEEN COMPLETED. REMAINING WORK INCLUDES MISSILE PROTECTION, UPGRADING FLOW CONTROL VALVES TO SEISMIC CATEGORY 1 AND INSTALLATION OF A NEW EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM. THESE ITEMS ARE SCHEDULED FOR COMPLETION DURING THE REFUELING OUTAGE OF 1986.

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THREE MILE ISLAND 1

SHORT-TERM MODIFICATIONS  
HAVE BEEN MADE AND LONG-TERM  
MODIFICATIONS ARE SCHEDULED FOR  
COMPLETION BY THE FIRST REFUEL-  
ING OUTAGE AFTER RESTART, AS  
REQUIRED IN CONFORMANCE WITH  
RELATED LICENSING BOARD DECISIONS  
IN THE RESTART HEARINGS.

MARKEY/IE

7/12/85

QUESTION 14. PROVIDE THE CURRENT STATUS OF IMPLEMENTATION OF ALL RECOMMENDATIONS OF NUREG-0667 AT EACH PLANT TO WHICH THEY ARE APPLICABLE.

ANSWER.

NUREG-0667 CONTAINS 22 RECOMMENDATIONS OF ACTIONS IN ORDER TO (1) REDUCE THE SENSITIVITY OF B&W REACTORS TO TRANSIENTS INVOLVING UNDERCOOLING EVENTS AND SMALL BREAK LOCA, AND (2) TO REDUCE THE LIKELIHOOD OF REACTOR TRANSIENTS BEING INITIATED BY THE INTEGRATED CONTROL SYSTEMS, NON-NUCLEAR INSTRUMENTATION AND ASSOCIATED POWER SUPPLIES. THE NUREG-0667 RECOMMENDATIONS WERE CLASSIFIED INTO HIGH PRIORITY ITEMS FOR NEAR-TERM IMPLEMENTATION (CATEGORY A) AND LOW PRIORITY ITEMS FOR LONG-TERM RECONSIDERATION (CATEGORY B). THE LONG-TERM ACTIONS HAVE NOT BEEN ISSUED AS REQUIREMENTS ALTHOUGH A NUMBER OF THEM HAVE BEEN IMPLEMENTED BY SOME UTILITIES WITH B&W PLANTS. THE IMPLEMENTATION STATUS FOR THE CATEGORY A ACTIONS AT ALL OPERATING B&W FACILITIES (INCLUDING TMI-1) IS PROVIDED ON THE ATTACHED TABLE (ITEM NUMBERS RELATE TO THE RECOMMENDATIONS CONTAINED IN NUREG-0667).

MARKEY/NRR  
7/3/85

# STATUS OF NUREG-0667 RECOMMENDATIONS AT B&W FACILITIES

ITEM NO.	RECOMMENDATION	STATUS
1.A	AFW SEISMIC QUAL.	ALL REVIEWS COMPLETE EXCEPT OCONEE (TO COMPLETE IN FALL 1985),
1.B	AFW DIVERSE POWER SUPPLIES	COMPLETE
1.D	SAFETY GRADE AFW	ALL COMPLETE EXCEPT TMI-1 (TO COMPLETE RESTART HEARING LONG-RANGE ITEMS BY NEXT REFUELING), RANCHO SECO (REPLACE TWO VALVES), AND CRYSTAL RIVER 3 (UPGRADE CONTROL SYSTEM). POWER SUPPLY TO TWO VALVES AT OCONEE ARE NOT SAFETY GRADE (SEE QUESTION 12)
2.B	AFW AUTO START	ALL COMPLETE EXCEPT TMI-1 (COMPLETE SYSTEM TEST AFTER RESTART, AND COMPLETE AUTOMATIC CONTROL AT NEXT REFUELING) AND CRYSTAL RIVER (ISSUE TECHNICAL SPECIFICATION)
3	DAVIS BESSE 1 DIVERSE DRIVE AFW	LICENSE CONDITION OF JANUARY 8, 1985 REQUIRES INSTALLATION AT 1986 REFUELING (SEE STAFF RESPONSE TO QUESTION No. 5)

ITEM No.	RECOMMENDATION	STATUS
5.A	IMPROVE ICS/NNI POWER BUS SEPARATION AND SIGNAL PATH CHANNELIZATION	IMPROVEMENTS IMPLEMENTED AT ALL B&W FACILITIES. RESPONSES TO STAFF QUESTIONS BEING EVALUATED BY STAFF (B&W OWNERS GROUP SUBMITTAL OF JANUARY 11, 1985)
5.C	IMPROVE ICS/NNI MULTIPLE INSTRUMENT FAILURE INDICATION	IMPROVEMENTS IMPLEMENTED AT ALL B&W FACILITIES. RESPONSES TO STAFF QUESTIONS BEING EVALUATED BY STAFF (B&W OWNERS GROUP SUBMITTAL OF JANUARY 11, 1985)
5.E	ICS/NNI POWER BUS REDUNDANCY	IMPROVEMENTS IMPLEMENTED AT ALL B&W FACILITIES. RESPONSES TO STAFF QUESTIONS BEING EVALUATED BY STAFF (B&W OWNERS GROUP SUBMITTAL OF JANUARY 11, 1985)
5.H	ICS/NNI FOLLOW-UP OF RESPONSES TO IEB 79-27	IMPROVEMENTS IMPLEMENTED AT ALL B&W FACILITIES. RESPONSES TO STAFF QUESTIONS BEING EVALUATED BY STAFF (B&W OWNERS GROUP SUBMITTAL OF JANUARY 11, 1985)
6	INSTALLATION OF SAFETY VITAL INSTRUMENT PANEL	INSTALLATIONS AT ALL B&W FACILITIES IS COMPLETE. STAFF REVIEW COMPLETE RANCHO SECO. REVIEW IN PROGRESS AT ANO-1, DAVIS BESSE, CRYSTAL RIVER, OCONEE, TMI-1
7	IMPROVE USE AND DISPLAY OF IN-CORE THERMOCOUPLES	COMPLETE



ITEM NO.	RECOMMENDATION	STATUS
8	SAFETY GRADE VENT AND PURGE ISOLATION ON HIGH RADIATION	ALL COMPLETE EXCEPT RANCHO SECO (ISSUE TECHNICAL SPECIFICATIONS)
13	OPERATOR TRAINING ON CRYSTAL RIVER 3 EVENT	COMPLETE
14	EMERGENCY PROCEDURES FOR LOSS OF NNI/ICS	IMPROVEMENTS IMPLEMENTED AT ALL B&W FACILITIES. RESPONSES TO STAFF QUESTIONS BEING EVALUATED BY STAFF (B&W OWNERS GROUP SUBMITTAL OF JANUARY 11, 1985)
15	MANDATORY SIMULATOR TRAINING FOR OPERATORS	COMPLETE
16	RCP RESTART CRITERIA	COMPLETE
17	ALTERNATE SOLUTION TO PORV UNRELIABILITY/SAFETY SYSTEM CHALLENGE	<p>COMPLETE. FOR PARTS (A)-(F) OF ITEM 17, THE FOLLOWING ACTIONS WERE TAKEN FOR B&amp;W FACILITIES:</p> <ul style="list-style-type: none"> <li>(A) SAFETY GRADE PORV NOT REQUIRED.</li> <li>(B) REDUNDANT PORV POSITION INDIC. INSTRUMENTATION INSTALLED.</li> <li>(C) DUAL SAFETY-GRADE PORV BLOCK VALVES NOT REQUIRED.</li> <li>(D) EPRI TEST PROGRAM COMPLETED.</li> <li>(E) ANTICIPATORY REACTOR TRIP ON TOTAL LOSS OF FEEDWATER INSTALLED.</li> <li>(F) PORV AND REACTOR TRIP SETPOINTS RESET. PORV LIFT PRESSURE SETPOINT IS ABOVE REACTOR TRIP SETPOINT.</li> </ul>

ITEM No.	RECOMMENDATION	STATUS
18	IREP CRYSTAL RIVER 3 STUDY	COMPLETE
20	RCP TRIP DURING SMALL BREAK LOCA	PROCEDURES IN PLACE. B&W OWNERS GROUP SUBMITTAL UNDER STAFF REVIEW.

QUESTION 15

AFTER CONCERNS WERE RAISED BY COMMISSIONER GILINSKY AND THEN REGION III IN 1983, TOLEDO EDISON PLEDGED TO TAKE STEPS TO IMPROVE ITS REGULATORY PERFORMANCE AND MANAGEMENT. ACCORDING TO THE NRC STAFF, THESE MEASURES WERE INEFFECTIVE. WHY DIDN'T THE NRC TAKE ACTION TO REQUIRE EFFECTIVE IMPROVEMENTS AND HAS THE NRC DETERMINED WHY THOSE ACTIONS THAT WERE TAKEN WERE NOT EFFECTIVE? ADDITIONALLY, WHAT MEASURES DOES THE COMMISSION PLAN TO TAKE NOW TO ASSURE REGULATORY AND MANAGEMENT IMPROVEMENTS AND HOW WILL THEY DIFFER FROM THOSE TAKEN OVER THE PAST TWO YEARS?

ANSWER

THE REGION III STAFF HAS WORKED CLOSELY WITH TOLEDO EDISON FOR THE PAST TWO YEARS IN AN ATTEMPT TO IMPROVE THEIR PERFORMANCE AND MANAGEMENT. WHILE IT WAS RECOGNIZED THAT IT WOULD TAKE SOME TIME TO ACHIEVE IMPROVED PERFORMANCE, THE PROGRESS WAS NOT AS TIMELY AS WE DESIRED. HOWEVER, WE ARE NOW BEGINNING TO SEE SOME POSITIVE RESULTS. SIGNIFICANT MANAGEMENT CHANGES HAVE RECENTLY BEEN ANNOUNCED AND PERSONAL INVOLVEMENT OF THE CHIEF EXECUTIVE OFFICER HAS BEEN ACHIEVED. A SUMMARY OF THE ACTIONS TAKEN BY NRC TO IMPROVE PERFORMANCE ARE AS FOLLOWS:

MARKEY/RIII

ON NOVEMBER 4, 1983 REGION III AND NRR MANAGEMENT MET WITH MR. JOHNSON, TOLEDO EDISON'S PRESIDENT, AND OTHER MEMBERS OF TOLEDO EDISON AND REQUESTED THEY DEVELOP A REGULATORY IMPROVEMENT PROGRAM FOR THE DAVIS-BESSE NUCLEAR PLANT. THE REQUEST WAS BASED ON CONCERNS RAISED BY COMMISSIONER GILINSKY DURING A SITE TOUR AND REGION III PERCEPTION THAT MANAGEMENT CONTROLS WERE NOT EFFECTIVELY PREVENTING VIOLATIONS OF REGULATORY REQUIREMENTS.

IN RECOGNITION OF THE NEED TO INCREASE OUR INSPECTION PRESENCE ONSITE, THE REGIONAL ADMINISTRATOR AUTHORIZED AN ADDITIONAL RESIDENT INSPECTOR AT THE DAVIS-BESSE SITE. THE ADDITIONAL RESIDENT INSPECTOR REPORTED ONSITE IN JANUARY 1984.

BEGINNING IN EARLY 1984, REGION III CONDUCTED WORKING MEETINGS APPROXIMATELY MONTHLY AT THE SECTION CHIEF AND BRANCH CHIEF LEVEL TO PROVIDE INPUT IN THE LICENSEE'S DEVELOPMENT OF WHAT THEY CALLED THEIR PERFORMANCE ENHANCEMENT PROGRAM (PEP). PERIODIC SENIOR MANAGEMENT MEETINGS WERE ALSO HELD TO MONITOR AND ASSESS THE LICENSEE'S PROGRESS. THESE MEETINGS DEMONSTRATED THE LICENSEE HAD PROGRESSED IN SELECTED AREAS, BUT OVERALL PROGRESS WAS SLOW. FURTHERMORE, IN JULY, 1984 REGION III LEARNED THAT TOLEDO EDISON MANAGEMENT WAS NOT AGGRESSIVELY SUPPORTING THE TRAINING PROGRAM AND IMPROVEMENTS RECOMMENDED BY THEIR TRAINING CONSULTANT.

MARKEY/RIII

DUE TO CONTINUING CONCERNS WITH THE LICENSEE'S PERFORMANCE, DURING THE MONTHS OF APRIL AND MAY, 1984 REGION III INITIATED A SPECIAL QUALITY ASSURANCE PROGRAM INSPECTION AIMED AT DETERMINING THE DEPTH OF DETERIORATION NOTED IN THE LICENSEE'S PERFORMANCE. THE INSPECTION TEAM CONCLUDED THAT WITH THE EXCEPTION OF QA PROGRAM IMPLEMENTATION PROBLEMS THE LICENSEE'S PERFORMANCE WAS PROGRESSING ADEQUATELY. (IN RETROSPECT REGION III FAILED TO FULLY APPRECIATE THE IMPLICATIONS WITH RESPECT TO THE PROGRESS OF THE LICENSEE.)

SUBSEQUENT TO THE QA INSPECTION, THERE WERE THREE INDICATORS THAT THE LICENSEE'S PROGRAM WAS NOT BEING IMPLEMENTED EFFECTIVELY: OBSERVATIONS BY THE RESIDENT INSPECTOR DID NOT SUPPORT THE CONCLUSIONS THAT PROGRESS WAS BEING MADE, A NUMBER OF TECHNICAL SPECIFICATION VIOLATIONS WERE EXPERIENCED, AND A PERFORMANCE APPRAISAL TEAM INSPECTION CONDUCTED IN AUGUST 1984 DISCLOSED A NUMBER OF SIGNIFICANT WEAKNESSES.

BASED ON THE ABOVE, THE REGIONAL ADMINISTRATOR AND THE DIRECTOR OF THE OFFICE OF INSPECTION AND ENFORCEMENT MET WITH TOLEDO EDISON SENIOR MANAGEMENT TO DISCUSS THE NEED FOR IMPROVED COMMUNICATION AND MANAGEMENT SUPPORT FOR PROGRAM IMPROVEMENTS RECOMMENDED BY THEIR OWN STAFF AND CONSULTANTS. SUBSEQUENTLY, A NOTICE OF

MARKEY/RI:1

VIOLATION AND A PROPOSED IMPOSITION OF CIVIL PENALTIES IN THE AMOUNT OF \$90,000 WAS ISSUED ON NOVEMBER 21, 1984, FOR VIOLATIONS, WHICH WERE INDICATIVE OF A BREAKDOWN IN THE LICENSEE'S MANAGEMENT CONTROL SYSTEM.

IN ADDITION THE SALP APPRAISAL GAVE THE UTILITY A CATEGORY 3 RATING IN FIVE FUNCTIONAL AREAS. THIS APPRAISAL GENERATED CONSIDERABLE NEGATIVE PUBLICITY FOR THE UTILITY. DURING THE SALP MEETING, WHICH WAS ATTENDED BY THE MEDIA, STATE AND LOCAL OFFICIALS, AND MEMBERS OF THE PUBLIC, THE REGIONAL ADMINISTRATOR TOLD UTILITY MANAGEMENT THAT "A CONTINUANCE OR FURTHER DECLINE OF THIS PERFORMANCE LEVEL AT DAVIS BESSE IS UNACCEPTABLE TO ME" AND THAT IMPROVED PERFORMANCE MUST BE REALIZED. UTILITY MANAGEMENT STATED THEY WERE COMMITTED TO IMPROVING REGULATORY PERFORMANCE AND OUTLINED A NUMBER OF ACTIONS BEING UNDERTAKEN TO ACCOMPLISH THE REGULATORY IMPROVEMENT PROGRAM. A WRITTEN RESPONSE TO THE SALP REPORT WAS SUBMITTED DESCRIBING PLANNED CORRECTIVE ACTIONS. (AS A DIRECT RESULT OF THE SALP APPRAISAL, COMMISSIONER ASSELSTINE VISITED THE DAVIS BESSE SITE ON JANUARY 11, 1985.)

IN THE SAME TIME PERIOD AS THE PUBLIC SALP MEETING, THE REGIONAL ADMINISTRATOR MET WITH THE CHIEF EXECUTIVE OFFICER OF TOLEDO EDISON (TECO) FOR A CANDID DISCUSSION OF REGULATORY PROBLEMS AT DAVIS BESSE. DURING THIS MEETING, TECO WAS ENCOURAGED TO BRING IN OUTSIDE MANAGEMENT ASSISTANCE TO DEAL WITH THEIR PROBLEMS.

MARKEY/RIII

AS FOLLOWUP TO THE SALP, THE REGIONAL ADMINISTRATOR DIRECTED THAT THE REGIONAL STAFF INCREASE THE NUMBER OF INSPECTIONS AT THE DAVIS-BESSE SITE. THE INSPECTIONS WERE DESIGNED TO REVIEW CURRENT ONGOING ACTIVITIES, AND TO MONITOR THE LICENSEE'S PROGRESS IN THEIR PERFORMANCE. WHILE IMPROVEMENTS HAVE BEEN NOTED, OVERALL PROGRESS CONTINUED TO BE SLOW. OVER 2,000 MANHOURS OF INSPECTION WERE SPENT AT THE DAVIS-BESSE SITE FROM JANUARY THROUGH MAY 1985.

IN ADDITION, THE REGIONAL ADMINISTRATOR HAS CONDUCTED PERIODIC MEETINGS WITH TOP MANAGEMENT IN AN EFFORT TO FOCUS MANAGEMENT ATTENTION ON ISSUES OF IMPORTANCE TO NRC. ALSO ON MAY 17, 1985, THE REGIONAL ADMINISTRATOR AND SENIOR NRC MANAGERS MET WITH THE CHIEF EXECUTIVE OFFICER TO HEAR FIRSTHAND THE ACTIONS BEING TAKEN TO IMPROVE REGULATORY PERFORMANCE. NRC MANAGERS ARE CONVINCED THAT THE CHIEF EXECUTIVE OFFICER IS DEDICATED TO ACHIEVING A HIGH LEVEL OF PERFORMANCE.

NOTWITHSTANDING, IN MAY OF 1985, AN ENFORCEMENT CONFERENCE WAS HELD WITH THE LICENSEE. DURING THE ENFORCEMENT MEETING, IT WAS APPARENT THAT THE MIDDLE MANAGEMENT AT TECO WAS CONTINUING TO HAVE SUBSTANTIAL INTERNAL COMMUNICATION PROBLEMS AND STILL DID NOT HAVE AN APPRECIATION TO PROMPTLY IDENTIFY AND CORRECT REGULATORY PROBLEMS. FURTHER ENFORCEMENT ACTION IS PENDING.

MARKEY/RIII



QUESTION 15 (CONTINUED) - 6 -

SIGNIFICANT MANAGEMENT CHANGES AT THE STATION AND CORPORATE LEVEL ARE IN PROGRESS. REGION III IS CLOSELY INVOLVED IN THE MONITORING OF NEW MANAGEMENT.

MARKEY/RIII



QUESTION 16. SHOULD COST-BENEFIT ANALYSIS BE USED TO DECIDE WHETHER TO REQUIRE A SAFETY IMPROVEMENT THAT WOULD BRING A NUCLEAR REACTOR INTO COMPLIANCE WITH NRC REGULATIONS?

ANSWER.

NO. THE LICENSEE FOR A NUCLEAR REACTOR MUST COMPLY WITH NRC REGULATIONS. IT SHOULD BE UNDERSTOOD, HOWEVER, THAT ALTERNATIVE MEANS MAY EXIST FOR COMPLYING WITH NRC REGULATIONS. COST-BENEFIT ANALYSES ARE USEFUL IN DECISIONS ON WHICH OF THE ALTERNATIVES MAY BE THE PREFERRED MEANS OF COMPLIANCE.

MARKEY/DEDROGR  
7/9/85

QUESTION 17. WHAT IS THE ROLE OF COST-BENEFIT ANALYSIS IN DECIDING WHETHER TO REQUIRE A SAFETY IMPROVEMENT AT A NUCLEAR REACTOR. YOUR RESPONSE SHOULD ANSWER THE FOLLOWING QUESTIONS:

- (A) HOW AND WHAT UNCERTAINTIES ARE FACTORED INTO ANY SUCH ANALYSES?
- (B) WHAT COSTS ARE CONSIDERED IN ANY SUCH ANALYSIS?
- (C) WHAT COSTS ARE NOT CONSIDERED IN ANY SUCH ANALYSIS?
- (D) WHAT BENEFITS ARE CONSIDERED IN ANY SUCH ANALYSIS?

ANSWER

COMMISSION ASSELSTINE HAS THE FOLLOWING COMMENTS CONCERNING QUESTIONS 17.A THROUGH 17.D:

THE COMMISSION SKEWS THE COST-BENEFIT ANALYSES OF POSSIBLE SAFETY IMPROVEMENTS BY INCLUDING DIRECT AND INDIRECT COSTS (I.E., ESSENTIALLY ALL COSTS ARE TO BE CONSIDERED) AND BY LIMITING THE BENEFIT SIDE OF THE EQUATION (E.G., BY IGNORING THE UNCERTAINTIES

AND BY IGNORING THE BENEFITS OF AVERTING ANOTHER SEVERE ACCIDENT LIKE THAT WHICH OCCURRED AT TMI-2). I WOULD ALSO POINT OUT THAT THE COMMISSION DOES NOT EVEN IDENTIFY AVERTED OFF-SITE PROPERTY DAMAGE COSTS (SUCH AS CONTAMINATED PRIVATE PROPERTY AS A RESULT OF A SEVERE ACCIDENT) AS A BENEFIT OF IMPROVED SAFETY. THIS LATTER BENEFIT TYPICALLY EXCEEDS, IN DOLLAR VALUE, THE MONETIZED VALUE OF AVERTED PERSON-REM. THUS, THE COMMISSION WEIGHTS THE FACTORS TO BE CONSIDERED ON THAT SIDE OF THE EQUATION THAT WOULD LIMIT SAFETY IMPROVEMENTS (I.E., CONSIDER ESSENTIALLY ALL COSTS) AND EITHER AVOIDS DECISIONS ON OR LIMITS THE FACTORS TO BE CONSIDERED ON THE OTHER SIDE OF THE EQUATION THAT WOULD SUPPORT SAFETY IMPROVEMENTS.

THE LOGICAL CONSEQUENCE OF THIS APPROACH TO BACKFITTING IS THAT WE CAN EXPECT MORE SIGNIFICANT EVENTS LIKE THE JUNE 9, 1985 EVENT AT DAVIS BESSE BECAUSE ONLY THEN WILL THERE BE A CLEAR DEMONSTRATION OF THE BENEFITS OF A SAFETY CHANGE SUFFICIENT TO OVERCOME THE BIASES IN THE COMMISSION'S BACKFIT STANDARD AND COST-BENEFIT ANALYSES.

QUESTION 17. WHAT IS THE ROLE OF COST-BENEFIT ANALYSIS IN DECIDING WHETHER TO REQUIRE A SAFETY IMPROVEMENT AT A NUCLEAR REACTOR. YOUR RESPONSE SHOULD ANSWER THE FOLLOWING QUESTIONS:

(A) HOW AND WHAT UNCERTAINTIES ARE FACTORED INTO ANY SUCH ANALYSES?

ANSWER.

THE COMMISSION'S POLICY FOR CONDUCT OF COST-BENEFIT ANALYSES IS DOCUMENTED IN NUREG/BR-0058, REVISION 1, "REGULATORY ANALYSIS GUIDELINES OF THE U.S. NUCLEAR REGULATORY COMMISSION, MAY 1984. NUREG/CP-3568, "A HANDBOOK FOR VALUE-IMPACT ASSESSMENT," DECEMBER 1983, IS AVAILABLE FOR METHODOLOGICAL REFERENCE.

COST-BENEFIT ANALYSIS IS PART OF THE CONSIDERATIONS THAT GUIDE DECISIONS CONCERNING PROPOSED NEW OR CHANGED GENERIC REQUIREMENTS. IT IS, FURTHER, PART OF THE CONSIDERATIONS IN DETERMINATION OF WHETHER A REQUIREMENT SHOULD BE BACKFITTED TO PLANTS ALREADY LICENSED. HOWEVER, COST-BENEFIT ANALYSIS IS NOT USED WHEN IT IS NECESSARY FOR A PLANT TO COME INTO COMPLIANCE WITH NRC REGULATIONS OR MEET AN ACCEPTABLE LEVEL OF SAFETY. IN SUCH INSTANCES THE PLANT SIMPLY MUST COMPLY WITH SUCH REGULATIONS AND MEET THE ACCEPTABLE LEVEL OF SAFETY.

(A) UNCERTAINTIES ARE OF THREE GENERAL TYPES: VARIABILITY OF DATA, ACCURACY OF MODELS AND OMISSION. IN SOME ANALYSES THE VARIABILITY OF DATA IS EXPLICITLY CALCULATED USING THE MATHEMATICAL RULES FOR COMBINING UNCERTAINTY. HOWEVER, THIS IS RECOGNIZED AS BEING MORE PRECISE THAN USUALLY JUSTIFIED BY THE CURRENT KNOWLEDGE OF THE ACTUAL STATISTICAL VARIANCE OF DATA SUCH AS COMPONENT FAILURE RATES. MODELS OF SYSTEM BEHAVIOR BOTH FOR RELIABILITY AND PERFORMANCE ARE INACCURATE TO VARYING AND USUALLY UNKNOWN DEGREES. THE EVENT AND FAULT TREE MODELS MAY NOT ACCURATELY REPRESENT THE SYSTEM FUNCTIONS. FOR EXAMPLE THESE MODELS ARE USUALLY BINARY, THAT IS THE SYSTEM EITHER FAILS OR SUCCEEDS. BUT MANY SYSTEMS CAN HAVE A SPECTRUM OF PARTIAL SUCCESS (OR FAILURE). THE POSSIBILITY OF OMISSIONS CAN NEVER BE TOTALLY EXCLUDED. ALL OF THE EVENTS SUCH AS INITIATING EVENTS AND SUBSEQUENT BRANCHING SEQUENCES MAY NOT BE IDENTIFIED. UNKNOWN OR UNRECOGNIZED PHYSICAL PHENOMENA MAY OCCUR. THEREFORE, UNCERTAINTY CANNOT BE AND IS NOT FACTORED INTO ANALYSES BY A RIGOROUS RULE OR FORMULA. THE UNCERTAINTIES IN EACH ANALYSIS MUST BE EVALUATED FOR THAT SPECIFIC ANALYSIS.

THE INEVITABLE SUBSTANTIAL UNCERTAINTIES INVOLVED IN PROBABILISTIC RISK ASSESSMENTS LEAD TO A DULY CAUTIOUS USE OF COST-BENEFIT ANALYSIS RESULTS. WHEN UNCERTAINTIES ARE UNUSUAL IN STRUCTURE OR UNUSUALLY LARGE, OR WHEN THE ISSUE INVOLVED IS PARTICULARLY SENSITIVE TO THEM, SENSITIVITY ANALYSES MAY BE PERFORMED, TO DETERMINE THE EFFECT OF ALTERNATIVE REASONABLE ASSUMPTIONS ON THE OUTCOME. IMPORTANT SPECIFIC UNCERTAINTIES ARE QUANTIFIED WHEN PRACTICAL AND MEANINGFUL, BUT ARE QUALITATIVELY RECOGNIZED WHEN THEY CANNOT BE QUANTIFIED. THE COST-BENEFIT ANALYSIS RESULTS ARE TAKEN INTO ACCOUNT WITH A CONSIDERATION OF THE UNCERTAINTIES, TO HELP REACH DECISIONS WITH A REASONABLE CONFIDENCE COMMENSURATE WITH THE POTENTIAL CONSEQUENCES OF ERROR.

COMMISSIONER ASSELSTINE ADDS THE FOLLOWING:

THE MOST STRAIGHTFORWARD RESPONSE TO THE QUESTION OF HOW UNCERTAINTIES ARE FACTORED INTO COMMISSION DECISIONS ON WHETHER TO REQUIRE A SAFETY IMPROVEMENT IS: THEY ARE NOT. EXAMPLES WHICH ILLUSTRATE THE COMMISSION'S REFUSAL TO CONSIDER UNCERTAINTIES IN ITS DECISIONMAKING INCLUDE: THE INDIAN POINT SPECIAL PROCEEDING, THE SEVERE ACCIDENT POLICY STATEMENT, THE

PROPOSED BACKFIT RULE AND THE PROVISIONAL SAFETY GOAL POLICY STATEMENT. THE COMMISSION'S TREATMENT OF UNCERTAINTIES IS DESCRIBED IN NUREG-1070 "NRC POLICY ON FUTURE REACTOR DESIGN-DECISIONS ON SEVERE ACCIDENT ISSUES IN NUCLEAR POWER PLANT REGULATION." SEE ATTACHED PAGES 133-140. I FIND THAT DISCUSSION WOEFULLY INADEQUATE.

FURTHER, THE COMMISSION'S TYPICAL PRACTICE IS TO RELY ON A "POINT ESTIMATE" IN REACHING A DECISION. THE "POINT ESTIMATE" ITSELF IS FREQUENTLY A FAULTY NUMBER TO THE EXTENT THAT THE SO-CALLED "MEDIAN" VALUE OF RISK IS USED. THE MEDIAN VALUE IS THAT POINT ON A SPECTRUM OF ESTIMATED RISKS AT WHICH HALF OF THE VALUES FALL ABOVE AND HALF FALL BELOW. THE COMMISSION OFTEN ARBITRARILY CHOOSES THAT POINT FOR PURPOSES OF REACHING DECISIONS. HOWEVER, USE OF THE MEAN OR "EXPECTED VALUE" OF RISK, WHICH IS THE AVERAGE VALUE OF THE SPECTRUM OF RISKS, IS THE MORE DEFENSIBLE APPROACH. THE ACRS RECOMMENDED, DURING ITS JULY 11, 1985 MEETING WITH THE COMMISSION, THAT THE MEAN VALUE RATHER THAN THE MEDIAN ESTIMATE BE USED AND NOTED THAT USE OF THE MEDIAN RATHER THAN MEAN ESTIMATE CAN RESULT IN A SUBSTANTIAL UNDERESTIMATE OF THE EFFECTS OF UNCERTAINTIES IN MAKING REACTOR ACCIDENT RISK ESTIMATES.

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# A Handbook for Value-Impact Assessment

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Prepared for  
U.S. Nuclear Regulatory  
Commission





**United States  
Nuclear Regulatory Commission**

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# **Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission**

Office of the Executive Director for Operations

May 1984

**DRAFT**

ATTACHMENT QUESTION 17A

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# NRC Policy on Future Reactor Designs

Decisions on Severe Accident Issues in  
Nuclear Power Plant Regulation

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**U.S. Nuclear Regulatory  
Commission**

Office of Nuclear Reactor Regulation



APPENDIX B

TREATMENT OF UNCERTAINTY IN SEVERE ACCIDENT PROGRAM

## APPENDIX B

### TREATMENT OF UNCERTAINTY IN SEVERE ACCIDENT PROGRAM

There are many uncertainties surrounding a comprehensive assessment of regulatory formulations. Of these, the largest and most troublesome originate in the analyses that are used to measure the safety benefits and attendant risks of the regulatory alternatives. The mutually supportive deterministic and probabilistic safety analysis techniques will be jointly employed, as noted in the main text.

#### I. ANALYSIS UNCERTAINTY

Deterministic safety analyses proceed from the judgmental selection of one or a few reference (or design-basis) accident scenarios as surrogates for the variety of accident scenarios to which the plants might be subject. The selection of these reference or design-basis accident scenarios introduces uncertainties. The necessity and sufficiency of measures to assure good plant performance to mitigate these particular accidents is often controversial. In fact, the need for severe accident policy development can be traced to the realization that the risk of nuclear power plants is dominated by events beyond the design basis. Probabilistic safety analyses approach the problem by considering the full array of possible accidents, each weighted according to estimates of their likelihood.

Once a catalog of accident scenarios is identified, the two forms of safety analysis (PRA and deterministic) utilize the same analytic technology: deterministic phenomenological analysis of accident progression, radiological releases, offsite doses and consequences. Historically, there have been differences: deterministic safety analysis has been done traditionally with conservative phenomenological analysis; whereas, the tendency in PRAs has been to use more realistic analysis. A deterministic safety analysis typically selects just one of the alternate accident scenarios for evaluation of containment performance, releases, and consequences; whereas, PRA commonly employs likelihood-weighted models of a spectrum of possible outcomes. However, there are few fundamental differences in the phenomenological or consequence models employed in deterministic and probabilistic safety analyses. Uncertainties originate in these analyses through modeling approximations, omissions arising from less-than-complete understanding or coverage of potentially contributory physical or chemical processes, and input parameters.

Neither deterministic nor probabilistic safety analysis is amenable to calculations of the magnitude or character of the uncertainties, because many of the important contributors to uncertainty (such as modeling approximations and omissions that are not stochastic) are not quantifiable. Nonetheless, a disciplined approach to the exploration of these uncertainties can be achieved by (a) employing both deterministic and probabilistic methods, (b) uniformly employing the latest state-of-the-art techniques in the application of both methods, and (c) employing sensitivity studies within the framework of both methods

by varying parametric assumptions over the full range of uncertainty. The staff is working to assure the reliability of the severe accident safety analyses by using evaluation models within the context in which they give trustworthy results.

## II. GENERIC APPLICABILITY OF REFERENCE PLANTS

Additional uncertainties arise when safety analyses of one or a few reference plants are utilized to draw inferences about a class of plants. There is a considerable evidence that severe accident risk, and therefore the incentive for additional requirements, is a function of subtle details of balance-of-plant (BOP) design. For example, PRAs of Indian Point Units 2 and 3, which are nearly identical in most respects, found significant differences in the severe accident susceptibility of the two units. To address this source of uncertainty for new standard plants, the NRC has already chosen to require extensive plant-specific, probabilistic analyses of severe accident risk. These analyses are to be employed as design tools, as design review tools, and as a disciplined method to assure that safety is not compromised by problems in the interfacing or coordination among the several design disciplines, procurement, construction, startup, development of operational and maintenance procedures, or the conduct of operations.

Because of the substantial time and resources required, the staff is not automatically presuming to employ severe accident safety analyses of all the operating plants or those under construction. Rather, to the extent practicable, we will seek to employ surrogates in the process of severe accident standards development for current plant design. However, the staff will employ performance criteria and required plant-specific analyses or other decision considerations in the implementation of new requirements to the extent necessary to assure that any retrofits are warranted and achieve the intended risk reductions.



### III. DECISION-MAKERS' PREFERENCES

There are important sources of uncertainty in regulatory standards development apart from those in safety analysis. The quantification and comparison of costs and benefits (including non-risk-related benefits such as non-political factors) together with the decision-makers' preferences can introduce substantial uncertainties. The report, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058), indicates that the scope and thoroughness of a regulatory analysis should be proportional to the safety significance and costs of the issue. It is important that the inquiry identify the dominant contributors to the costs and benefits, even if they are subtle or indirect. A regulatory analysis can give a severely distorted result if a dominant contributor to the cost or benefits is omitted, seriously underestimated, or exaggerated. Thus, it is incumbent upon the staff, in its preparation of regulatory analyses, to make a thorough search for potentially dominant contributors to costs and benefits.





QUESTION 17. (B) WHAT COSTS ARE CONSIDERED IN ANY SUCH ANALYSIS?

ANSWER.

ALL COSTS ATTRIBUTABLE TO A SAFETY IMPROVEMENT ARE CONSIDERED. THESE INCLUDE COSTS BORNE BY ANY SEGMENT OF THE ECONOMY, INCLUDING THE INDUSTRY, THE NRC AND OTHER GOVERNMENT ENTITIES, AND THE PUBLIC. THE COSTS ARE TAKEN INTO ACCOUNT AS NET COSTS, I.E., WHEN ONE COST ELEMENT IS SUBSTITUTED FOR ANOTHER, THE DIFFERENCE IS COUNTED. FUTURE COSTS ARE DISCOUNTED TO PRESENT WORTH. COSTS ARE MULTIPLIED BY THE PROBABILITY OF THEIR OCCURRENCE DURING THE LIFETIME OF THE PLANT.

THE LICENSEE'S COSTS INCLUDE THE NET COST TO DESIGN, ANALYZE, PURCHASE, LICENSE, AND INSTALL STRUCTURES OR EQUIPMENT; HIRE, PAY, TRAIN, AND LICENSE ANY NEW PERSONNEL; AND WRITE, REVIEW, REVISE, AND GET APPROVAL OF ANY ADMINISTRATIVE, OPERATING OR EMERGENCY PROCEDURES. ALSO INCLUDED ARE THE ADDITIONAL COSTS OF REPLACEMENT POWER DUE TO SHUTDOWN (OR START-UP DELAY) TO MAKE CHANGES, AND ANY COST DIFFERENTIALS DUE TO DECREASED (OR INCREASED) PLANT RELIABILITY BECAUSE OF CHANGES.

NRC COSTS, WHICH ARE USUALLY MUCH LESS THAN THE LICENSEE'S COSTS, INCLUDE THE COST TO DEVELOP, IMPLEMENT, REVIEW, AND INSPECT ANY NEW REQUIREMENT.

QUESTION 17. (c) WHAT COSTS ARE NOT CONSIDERED IN ANY SUCH ANALYSIS?

ANSWER.

HIGHLY INDIRECT AND SPECULATIVE COSTS ARE NOT CONSIDERED. (FOR EXAMPLE, THE POSSIBLE INTERNATIONAL TRADE IMPACT OF NUCLEAR-PLANT DOWN TIME, THROUGH POSSIBLE SUBSTITUTION OF OIL-FIRED CAPACITY, IS VIEWED AS TOO SPECULATIVE AN IMPACT TO RECOGNIZE.)

SINCE THE NET NATIONAL COST IMPACT IS SOUGHT, AND COSTS ARE CONSIDERED REGARDLESS OF WHO BEARS THEM, TRANSFER PAYMENTS (SUCH AS TAXES AND INSURANCE PREMIUMS AND PAYMENTS) ARE NOT CONSIDERED.

THE AVERTED COST OF PLANT DAMAGE AND ITS EFFECT ON THE NET COST AND ON THE COST-BENEFIT RELATION ARE STATED SEPARATELY. THE QUESTION OF WHETHER AND TO WHAT EXTENT TO TAKE SUCH AVERTED COSTS INTO ACCOUNT IS UNSETTLED. IT IS BEING CONSIDERED AS PART OF THE NRC SAFETY GOAL EVALUATION PROGRAM.

QUESTION 17. (D) WHAT BENEFITS ARE CONSIDERED IN ANY SUCH ANALYSIS?

ANSWER.

THE BENEFITS CONSIDERED ARE THE SAFETY AND OTHER (ENVIRONMENTAL, PUBLIC PROPERTY, NATIONAL-SECURITY, AND ANTI-TRUST) VALUES THAT IT IS THE NRC'S MISSION TO PROTECT.

BENEFITS ARE QUANTIFIED WHEN IT IS PRACTICAL TO DO SO IN A MEANINGFUL WAY, BUT ARE TAKEN INTO ACCOUNT QUALITATIVELY WHEN THEY CANNOT BE QUANTIFIED.

IN COST-BENEFIT ANALYSIS OF PROPOSED SAFETY IMPROVEMENTS, THE BENEFITS INVOLVED USUALLY APPEAR AS A NET RISK REDUCTION, EXPRESSED IN PERSON-REM OF RADIATION DOSE TO THE PUBLIC AND WORKERS, PROBABILISTICALLY ESTIMATED FOR THE REMAINING PLANT LIFE.

QUESTION 17. (E) WHAT BENEFITS ARE NOT CONSIDERED IN ANY SUCH ANALYSIS?

ANSWER.

PRIVATE ECONOMIC BENEFITS, SUCH AS ECONOMIC BENEFIT TO THE LICENSEES, ARE NOT CONSIDERED AS SUCH. (THEY MAY, HOWEVER, ENTER AS FAVORABLE COST IMPACTS IN THE COMPUTATION OF NET COSTS, AS DISCUSSED UNDER (B) AND (C), ABOVE.)

QUESTION 18. SHOULD COST-BENEFIT ANALYSIS BE USED TO DECIDE WHETHER TO REQUIRE DAVIS BESSE TO INSTALL A THIRD AUXILIARY FEEDWATER PUMP WHEN IT DOES NOT MEET THE STANDARD REVIEW PLAN RELIABILITY CRITERION AND WHEN NO OTHER U.S. REACTOR HAS THE SAME DESIGN VULNERABILITY OF ONLY TWO STEAM DRIVEN PUMPS?

ANSWER.

THE EXISTING NRC REGULATIONS (10 CFR PART 50, APPENDIX A GENERAL DESIGN CRITERIA) REQUIRE THE DESIGN OF A SAFETY SYSTEM WITH SUITABLE REDUNDANCY TO MEET THE SINGLE FAILURE CRITERION AND TO ACCOMPLISH ITS SAFETY FUNCTION ASSUMING A SINGLE FAILURE WITH EITHER ONSITE ELECTRIC POWER OR OFF-SITE POWER UNAVAILABLE. IN GENERAL, A TWO TRAIN SAFETY SYSTEM WOULD BE EXPECTED TO MEET SUCH GENERAL DESIGN CRITERIA (GDC).

AS THE GDC NOTES, THE MINIMUM ACCEPTABLE REDUNDANCY AND DIVERSITY OF SUBSYSTEMS AND THE REQUIRED INTERCONNECTION AND INDEPENDENCE OF THE SUBSYSTEMS HAVE NOT YET BEEN DEVELOPED OR DEFINED AS PART OF THE REGULATIONS. SEVERAL REACTOR PLANTS BUILT UNDER THESE REGULATIONS CHOSE TO DESIGN THE AUXILIARY FEEDWATER SYSTEMS WITH USE OF STEAM TURBINE DRIVEN PUMPS ONLY. (FOR EXAMPLE, DAVIS-BESSE, <sup>1/</sup> TURKEY POINT 3 AND 4, CALVERT CLIFFS <sup>1/</sup> 1 AND 2 AND HADDAM NECK <sup>1/</sup>

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<sup>1/</sup> USED TWO STEAM DRIVEN PUMPS.

ALL USED STEAM TURBINE PUMPS.) THESE SYSTEMS ARE CONSISTENT WITH THE GDC REGULATIONS AND CAN HAVE THE ADVANTAGE OF BEING LESS DEPENDENT ON AC POWER SUPPLIES.

SUBSEQUENT TO THE TMI-2 ACCIDENT AND THE CONDUCT OF AFWS STUDIES REPORTED IN NUREG 0611 AND 0635, THE STAFF CHOSE TO IMPROVE ITS DEFINITION OF THE RELIABILITY AND DIVERSITY SOUGHT FOR THE AFWS DESIGN. THIS WAS DONE THROUGH THE STANDARD REVIEW PLAN (SRP). THESE EFFORTS LED TO THE STAFF'S RECOGNITION OF AFWS DESIGN VULNERABILITIES AND THE POTENTIAL USEFULNESS AND NEED FOR ADDITIONAL REDUNDANCY AND DIVERSITY. THIS RECOGNITION CAME ABOUT INDEPENDENT OF ANY COST-BENEFIT ANALYSES. THE SRP WAS SUBSEQUENTLY CODIFIED INTO THE REGULATIONS AND ANY APPLICATIONS (FOR PLANT OPERATING LICENSES, CONSTRUCTION PERMITS, MANUFACTURING LICENSES AND PRELIMINARY OR FINAL DESIGN APPROVAL FOR STANDARD PLANTS) DOCKETED AFTER MAY 17, 1982 ARE REQUIRED TO EVALUATE CONFORMANCE WITH THE SRP.

PLANTS SUCH AS DAVIS BESSE LICENSED TO OPERATE PRIOR TO THIS DATE HAVE BEEN JUDGED TO HAVE AN AFWS DESIGN MEETING THE MINIMUM REQUIREMENTS OF NRC REGULATIONS. OUR RESPONSE TO QUESTION 16 STATES THAT A NUCLEAR REACTOR MUST COMPLY WITH NRC REGULATIONS. THE DAVIS BESSE TWO TRAIN AFW DESIGN DOES COMPLY WITH THE NRC REGULATIONS. THE STAFF HAS, HOWEVER, RECOMMENDED IMPROVEMENTS TO

THE AFWS RELIABILITY SINCE 1979 AS THE CHRONOLOGY OF THIS MATTER REVEALS. (SEE PREVIOUS NRC RESPONSES TO REP. MARKEY QUESTIONS OF JUNE 17, 1985.) FOR DAVIS BESSE THE STAFF RECOMMENDED THAT AFWS RELIABILITY IMPROVEMENTS BE ACHIEVED BY ADDITION OF A THIRD AND DIVERSE POWERED AFW PUMP; THE LICENSEE WAS NOT IN AGREEMENT WITH THIS RECOMMENDATION. WHEN SAFETY IMPROVEMENTS SUCH AS THESE BEYOND THE NRC REGULATIONS ARE BEING SOUGHT, IT IS EXPECTED THAT THE STAFF AND THE LICENSEE WILL CONTINUE TO USE COST-BENEFIT TECHNIQUES TO EXPLORE THE MERITS OF VARIOUS SAFETY ALTERNATIVES THAT MAY BE AVAILABLE.

COMMISSIONER ASSELSTINE ADDS THE FOLLOWING:

DAVIS BESSE AS WELL AS MOST OTHER OPERATING PLANTS WERE "GRANDFATHERED" FROM ANY RELIABILITY REQUIREMENT OR OBJECTIVE. THUS, BECAUSE THE PARTICULAR DESIGN OF THE AUXILIARY FEEDWATER SYSTEM FOR THOSE PLANTS MET THE COMMISSION'S REGULATIONS WHEN THEY WERE LICENSED, THE COMMISSION USES COST-BENEFIT ANALYSES IN DECIDING WHETHER TO REQUIRE ENHANCED RELIABILITY IN THAT SAFETY SYSTEM EVEN IF THE PARTICULAR DESIGN HAS LOW RELIABILITY. GIVEN THE WAY THE COMMISSION CONDUCTS COST-BENEFIT ANALYSES AND WEIGHS THE FACTORS TO BE CONSIDERED (SEE MY COMMENTS IN RESPONSE TO QUESTION 17), I FIND THIS AN INADEQUATE WAY OF GOING ABOUT PROTECTING THE PUBLIC HEALTH AND SAFETY.



QUESTION 19. DOES THE COMMITTEE TO REVIEW GENERIC REQUIREMENTS AND THE EXECUTIVE DIRECTOR FOR OPERATIONS HAVE ANY AUTHORITY OVER THE DECISIONS AND JUDGMENTS OF THE DIRECTOR OF NUCLEAR REACTOR REGULATION TO REQUIRE SAFETY IMPROVEMENTS AT NUCLEAR REACTORS?

ANSWER.

THE CRGR DOES NOT HAVE ANY AUTHORITY OVER THE DECISIONS AND JUDGMENTS OF THE DIRECTOR, NRR. HOWEVER, IN ACCORDANCE WITH THE COMMISSION APPROVED CRGR CHARTER, ALL SIGNIFICANT GENERIC ACTIONS PROPOSED BY NRR TO BE IMPLEMENTED UPON A CLASS OF NUCLEAR POWER PLANTS, LICENSEES, OR APPLICANTS ARE REQUIRED TO BE REVIEWED BY THE CRGR. IN ACCORDANCE WITH THE CRGR CHARTER, THE CRGR MAKES RECOMMENDATIONS TO THE EDO CONCERNING APPROVAL OR DISAPPROVAL OF PROPOSED GENERIC REQUIREMENTS WHICH HAVE BEEN FORWARDED BY THE NRR DIRECTOR (OR OTHER OFFICE DIRECTOR) FOR CRGR CONSIDERATION.

AN EXPLANATION OF THE EDO'S AUTHORITY OVER THE DECISIONS AND JUDGMENTS OF THE NRR DIRECTOR FOLLOWS:

SECTION 4(C) OF REORGANIZATION PLAN NO. 1 OF 1980, AS AMENDED, PROVIDES THAT "[T]HE FUNCTION OF THE DIRECTOR[] OF NUCLEAR REACTOR REGULATION...OF REPORTING DIRECTLY TO THE COMMISSION IS HEREBY TRANSFERRED SO THAT SUCH OFFICER[] REPORT[S] TO THE EXECUTIVE DIRECTOR FOR OPERATIONS." THE LEGISLATIVE HISTORY OF THE PLAN CLARIFIES THE NATURE OF THIS REPORTING RELATIONSHIP. THE PURPOSE

MARKEY/DEDROGR  
REVISED 7/16/85

OF SECTION 4(C) OF THE PLAN IS TO STREAMLINE THE REPORTING AND  
SUPERVISION OF LINE OFFICE DIRECTORS. S. REP. NO. 790, 96TH CONG.,  
2ND SESS. 22 (1980).

QUESTION 20

PLEASE PROVIDE THE FOLLOWING INFORMATION:

- A. AN ESTIMATE OF HOW MUCH WOULD IT COST TOLEDO EDISON TO HAVE INSTALLED, IN 1980, THE PUMP THAT IS NOW PLANNED TO BE INSTALLED IN 1986? HOW MUCH WILL THE PUMP AND ITS INSTALLATION COST IN 1986?

ANSWER

ON DECEMBER 31, 1981 TOLEDO EDISON PROVIDED THE NRC WITH COST COMPARISONS FOR IMPROVEMENTS TO THE AUXILIARY FEEDWATER SYSTEM. THE COMPARISON WAS PREPARED FOR TWO SYSTEMS CONFIGURATIONS WHICH INCLUDED THE ADDITION OF A THIRD TRAIN OF AUXILIARY FEEDWATER AND AN UPGRADE CONFIGURATION BASED ON THE RELIABILITY STUDY CONDUCTED BY EDS. THIS COMPARISON IS SHOWN BELOW.

<u>CONFIGURATION</u>	<u>OVERALL FIGURE OF MERIT</u>	<u>IMPROVEMENTS OVER POST-TMI-2 CONFIGURATION</u>	<u>ASSOCIATED COST (ESTIMATED)</u>
THIRD TRAIN	$2.2 \times 10^{-4}$	FACTOR OF 15	\$1,200,000
ANALYSIS BASED	$1.4 \times 10^{-4}$	FACTOR OF 24	\$ 300,000

MARKEY/RIII

QUESTION 20 A (CONTINUED) - 2 -

TOLEDO EDISON ESTIMATES THAT IT WILL COST APPROXIMATELY \$1,800,000 TO INSTALL THE ADDITIONAL PUMP IN 1986. IT SHOULD BE NOTED THE CONFIGURATION OF THE THIRD TRAIN ESTIMATED IN 1981 AND LISTED ABOVE IS DIFFERENT THAN THE ONE PLANNED TO BE INSTALLED IN 1986 THEREFORE A COMPARISON OF THE COST INVOLVED IS NOT APPROPRIATE.

QUESTION 20      B.    HOW MUCH DID TOLEDO EDISON PAY FOR THE  
RELIABILITY ANALYSIS BY EDS NUCLEAR INC.?

ANSWER

TOLEDO EDISON VERBALLY INFORMED THE NRC THAT THE COST OF THE  
EDS ANALYSIS WAS \$248,660.

MARKEY/RIII

QUESTION 20. (c) HOW MUCH DID NRC PAY FOR THE BROOKHAVEN  
STUDY WHICH EVALUATED THE EDS RELIABILITY  
ANALYSIS?

ANSWER.

APPROXIMATELY \$15,000.

MARKEY II/NRR  
7/3/85

QUESTION 20

D. AN ESTIMATE OF HOW MUCH TOLEDO EDISON HAS  
EXPENDED SINCE 1979 ON IMPROVEMENTS TO THE  
AUXILIARY FEEDWATER SYSTEM THAT WERE USED TO  
JUSTIFY NOT INSTALLING A THIRD AUXILIARY  
FEEDWATER PUMP?

ANSWER

TOLEDO EDISON DOES NOT HAVE THIS INFORMATION READILY  
AVAILABLE.

MARKEY/RIII

QUESTION 20 PLEASE PROVIDE THE FOLLOWING INFORMATION:

E. AN ESTIMATE OF THE COST AND STAFF  
HOURS EXPENDED SINCE 1980 TO EVALUATE  
WHETHER DAVIS-BESSE SHOULD INSTALL  
A THIRD AUXILIARY FEEDWATER PUMP

ANSWER

STARTING WITH F.Y. 81, OUR RECORDS SHOW THAT A TOTAL OF 400 PROFESSIONAL STAFF HOURS HAVE BEEN EXPENDED ON AUXILIARY FEED-WATER RELIABILITY ISSUES. OF THIS, AT LEAST 98 PROFESSIONAL STAFF HOURS CAN BE SPECIFICALLY IDENTIFIED AS RELATED TO THE DIVERSE-POWERED PUMP ISSUE. SOME OF THE REMAINING 302 HOURS EXPENDED ALSO RELATE TO THIS ISSUE BUT CANNOT BE SPECIFICALLY SO IDENTIFIED.

IN ADDITION, CONTRACTOR COSTS RELATED TO EVALUATING THE LICENSEE'S RELIABILITY STUDY AND PRODUCING AN INDEPENDENT ESTIMATE WERE \$15,000. THE CONTRACTOR EFFORT WAS RELATED TO THE DAVIS-BESSE AUXILIARY FEEDWATER RELIABILITY REVIEW, AND IT IS NOT POSSIBLE TO DETERMINE WHAT PART OF THAT EFFORT WAS DIRECTLY INVOLVED WITH THE THIRD PUMP ISSUE.

MARKEY II/NRR  
7/3/85



QUESTION 20      F.    WHAT HAS THE JUNE 9, 1985 INCIDENT COST  
                              TOLEDO EDISON UP TILL NOW? PLEASE INCLUDE AN  
                              ESTIMATE OF REPLACEMENT POWER COSTS PER DAY,

ANSWER

THE NRC BELIEVES THAT THE MAJOR ADDITIONAL COST INCURRED BY THE OWNERS OF THE DAVIS-BESSE REACTOR SINCE THE JUNE 9, 1985 EVENT IS THE REPLACEMENT ENERGY COST. ASSUMING DAVIS-BESSE WOULD HAVE HAD A TYPICAL AVAILABILITY FACTOR OVER THE 34 DAY PERIOD FROM JUNE 9, 1985 THROUGH JULY 12, 1985, THIS COST IS ESTIMATED BY THE NRC TO BE ABOUT \$10 MILLION. IT IS LIKELY THAT THERE HAVE BEEN SOME OTHER ADDITIONAL UTILITY COSTS INCURRED AS A RESULT OF THIS EVENT, HOWEVER, IT IS BELIEVED THAT THESE COSTS ARE SMALL IN COMPARISON TO THE REPLACEMENT ENERGY COST PENALTY.

THE ABOVE ESTIMATE IS BASED ON AN AVERAGE DAILY REPLACEMENT ENERGY COST FOR DAVIS-BESSE OF \$273,000. THIS AMOUNT IS DERIVED FROM POWER POOL PRODUCTION COST SIMULATIONS WHICH ESTIMATE THE DIFFERENCE IN TOTAL VARIABLE COSTS WHEN THE DAVIS-BESSE UNIT IS AVAILABLE FOR GENERATION AND WHEN IT IS NOT.

MARKEY/RIII