Transcript of Proceedings

UNITED STATES Oɾ AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: DENWOOD ROSS
(SUPPLEMENT INCLUDING EXHIBITS 8 THROUGH 14)

Bethesda, Maryland

August 2, 1979

POOR ORIGINAL

Acme Reporting Company

Official Reporters
1411 K Street, N.W.
Washington, D. C. 20005
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PDR

7908290021

B/Y-79-10

April 30, 1979

MEMORANDUM FOR: Edward S. Christenbury, Chief Hearing Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management

SUBJECT: I&E EVALUATIONS - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

Our memorandum to you of April 2, 1979 recommending that I&E evaluations be provided to appropriate Boards also advised that we would request a review by DSS which could lead to additional information for the Boards.

We are 'sed by DSS that the current workload associated with the Three Mile Island-2 incident has postponed any review of the subject material. We will follow this and keep you advised.

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

cc: H. Denton
E. .ase
D. 'isenhut
J. Davis, IE
R. Boyd
R. Mattson
V. Stello
R. DeYoung
V. Moore
L. Nichols
B. Grimes
J. Stolz
R. Baer
O. Parr
S. Varga
IE (7)
R. Martin

Distribution
Central File
NRR Reading
F. Williams
F. Williams Reading
J. Lee

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OFFICE	LWR:DPM	AD:LWR:DPM				
SURNAME	FWilliams/bm	DBVassallo				

BN-79-10

April 2, 1979

MEMORANDUM FOR: Roger J. Mattson, Director, Division of Systems Safety

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management

SUBJECT: I&E EVALUATIONS - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

The enclosed I&E evaluations have been provided to OELD for transmittal to appropriate Boards.

Because the concerns are highly technical and in particular, the concern in Item 3 relates to the TMI-2 transient we are providing them to you under separate cover for your consideration. Please advise us of any additional assessment of these concerns which should be provided to the Boards.

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: w/o enclosure
H. Denton
E. Case
D. Eisenhut
J. Davis
R. Boyd
R. Mattson
V. Stello
R. DeYoung
V. Moore
L. Nichols
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D. Thompson

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SURNAME ▶	FWilliams/bm	DBVassallo				

BN-79-10

April 2, 1979

MEMORANDUM FOR: Edward S. Christenbury, Chief Hearing Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management

SUBJECT: ADDITIONAL INFORMATION - BOARD NOTIFICATION BN-79-10 - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS

Our recommendation (BN-79-10) for Board Notification on this subject was provided in our memorandum of March 6, 1979. At that time we advised that additional Board information would be provided following receipt of the I&E evaluations. The evaluations were provided to this office by the enclosed D. Thompson to D. B. Vassallo memorandum of March 29, 1979. We concur that they be provided to the appropriate Boards.

By separate memorandum we will request a review by DSS which could lead to additional information for the Boards. We recommend, however, that the enclosed I&E evaluation be provided at this time and not held up on the basis of any further review.

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: H. Denton
E. Case
D. Eisenhut
J. Davis
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J. Lee

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OFFICE	TC:DPM	AD:LWR:DPM			
SURNAME	FWilliams/bm	DBVassallo			
	4/2/79	4/2/79			

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

MAR 29 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assisant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, XOOS

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2
REFERENCES: 1. Memo: Thompson to Vassallo dtd 3/1/79
2. Memo: Thompson to Vassallo dtd 3/12/79

As noted in the above referenced submittals additional information in the form of staff discussion and evaluation would be forthcoming on the captioned board notification. Enclosed is the additional information for submittal to the appropriate Boards.

Dudley Thompson, Executive Officer
for Operations Support
Office of Inspection and Enforcement

Enclosures:
1. Memo: Moseley to Thompson dtd 3/28/79 w/encls
2. Memo: Moseley to Thompson dtd 3/29/79

cc: N. C. Moseley, IE, w/o encl
S. E. Bryan, IE, w/o encl
J. F. Streeter, RIII, w/encl
J. S. Creswell, RIII, w/encl
G. C. Gower, IE, w/encl
IE Files w/encl

POOR ORIGINAL

CONTACT: G. C. Gower, IE
49-27246

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

March 29, 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

In light of the transient experienced at Three Mile Island on March 28, 1979, we will review our preliminary evaluation of Item 3 contained in my March 28, 1979 memorandum to you. It is possible that the additional information will cause our assessment to change.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. C. Kirkpatrick
G. C. Gower
V. D. Thomas

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

MAR 2 8 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 28, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing boards. At that time only a preliminary evaluation had been done. We have completed our evaluation of each of the items and that information is enclosed. This additional information should be forwarded to the licensing boards.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. Kirkpatrick
L ~~G. C.~~ Gower
V. D. Thomas

CONTACT: J. C. Stone
(x28019)

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised in connection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9%, an analysis was done by B&W to determine what effect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unreviewed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly end fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (letter from W. O. Parker to R. C. Rusche dated 7/21/76) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% - r with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 50). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

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

MAR 29 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assisant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, XOOS

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2
REFERENCES: 1. Memo: Thompson to Vassallo dtd 3/1/79
2. Memo: Thompson to Vassallo dtd 3/12/79

As noted in the above referenced submittals additional information in the form of staff discussion and evaluation would be forthcoming on the captioned board notification. Enclosed is the additional information for submittal to the appropriate Boards.

Dudley Thompson, Executive Officer
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Office of Inspection and Enforcement

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IE Files w/encl

CONTACT: G. C. Gower, IE
49-27246

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

March 29, 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

In light of the transient experienced at Three Mile Island on March 28, 1979, we will review our preliminary evaluation of Item 3 contained in my March 28, 1979 memorandum to you. It is possible that the additional information will cause our assessment to change.

Norman C. Moseley

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Operations Inspection, IE

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Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
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CONTACT: J. C. Stone
(x28019)

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DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised ir nnection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9% an analysis was done by B&W to determine what effect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unrevi wed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly end fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (letter from W. O. Parker to R. C. Rusche dated 7/21/76) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

   Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regard: g control rod drive system trip breaker maintenance is attache as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 50). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

BN-79-10

March 6, 1979

MEMORANDUM FOR: Edward S. Christenbury, Hearing Division Director and Chief Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management, NRR

SUBJECT: BOARD NOTIFICATION - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

The enclosed memorandum from I&E provides information originated by a Reactor Inspector as Board Notification material. Although I&E concluded that the information was not relevant and material the originator still believes that Boards should be informed.

Since we have not yet received I&E's written discussion and evaluation of these matters we have not reviewed the material in any detail. Regardless, however, in accordance with established procedures the information should be provided to appropriate Boards based on the originator's concerns.

The originator recommends that the Davis Besse 2 & 3 and Midland 1 & 2 Boards be informed.

In neither case is the SER Supplement issued but we have no objection to providing the information. In addition, since the concerns appear to apply to B&W plants, we recommend that you also provide the information to the Erie, Greene County, Pebble Springs and TMI-2 Boards.

When we receive the I&E written evaluations we will review them to determine whether additional review should be provided by DSS. In any event, we will follow this up with additional information for the Boards in the near future.

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: See attached sheet

TC:DPM	AD:LWR:DPM
FWilliams/bm	DBVassallo

cc: H. Denton
E. Case
D. Eisenhut
J. Davis
R. Boyd
V. Stello
R. DeYoung
L. Nichols
B. Grimes
J. Stolz
R. Baer
O. Parr
S. Varga
IE (7)
E. Jordan
D. Thompson

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

MAR 01 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

The enclosed information is being forwarded for Board Notification. Your contact on this matter for any additional information is E. L. Jordan, ext. 28180.

Please note that the 2/28/79 cover memorandum, Moseley to Thompson, states that the originator, after being informed of IE Headquarters evaluation, still believes the information should be sent forward to the boards.

We request to be informed of your disposition on this matter.

Dudley Thompson

Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosures:
1. Memo NCMoseley to DThompson dtd 2/28/79
2. Memo JSCreswell to JFStreeter dtd 1/8/79 w/enclosures

cc: N. C. Moseley, ROI w/o encls
E. L. Jordan, ROI w/o encls
J. F. Streeter, RII w/o encls
J. S. Creswell, RII w/o encls
G. C. Gower, XOOS w/encls
IE Files w/encls

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

FEB 2 8 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING-BOARDS (AITS F30468H2)

Enclosed are six items sent in by Region III for forwarding to sitting Licensing Boards for cases involving Babcock and Wilcox as the Nuclear Steam System Supplier. Our preliminary evaluation indicates these items do not appear to be new issues or to put a different light on the issues and therefore, in our opinion, do not meet the intended criteria for Board notification.

The originator was informed, via telephone, of this determination on February 27, 1979. His position was that our evaluation did not provide any information that he did not already have and his concern was whether or not these items had been considered and resolved on a generic basis for all B&W plants. Because of this he still believed the items should be sent to the Licensing Boards. IE Manual Chapter 1530 requires that if, after a negative determination, the originator continues to believe that the information should be submitted to the Board(s), the information will be submitted. We therefore request the enclosed items be sent to the appropriate Licensing Boards.

We will provide a written discussion and evaluation of each item within seven (7) days of the date of this memorandum.

E. L. Jordan
for Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Memorandum Creswell to Streeter
dated January 8, 1979

cc w/o encl:
S. E. Bryan
E. L. Jordan
D. Kirkpatrick
J. C. Stone
G. C. Gower
R. F. Heishman, RIII

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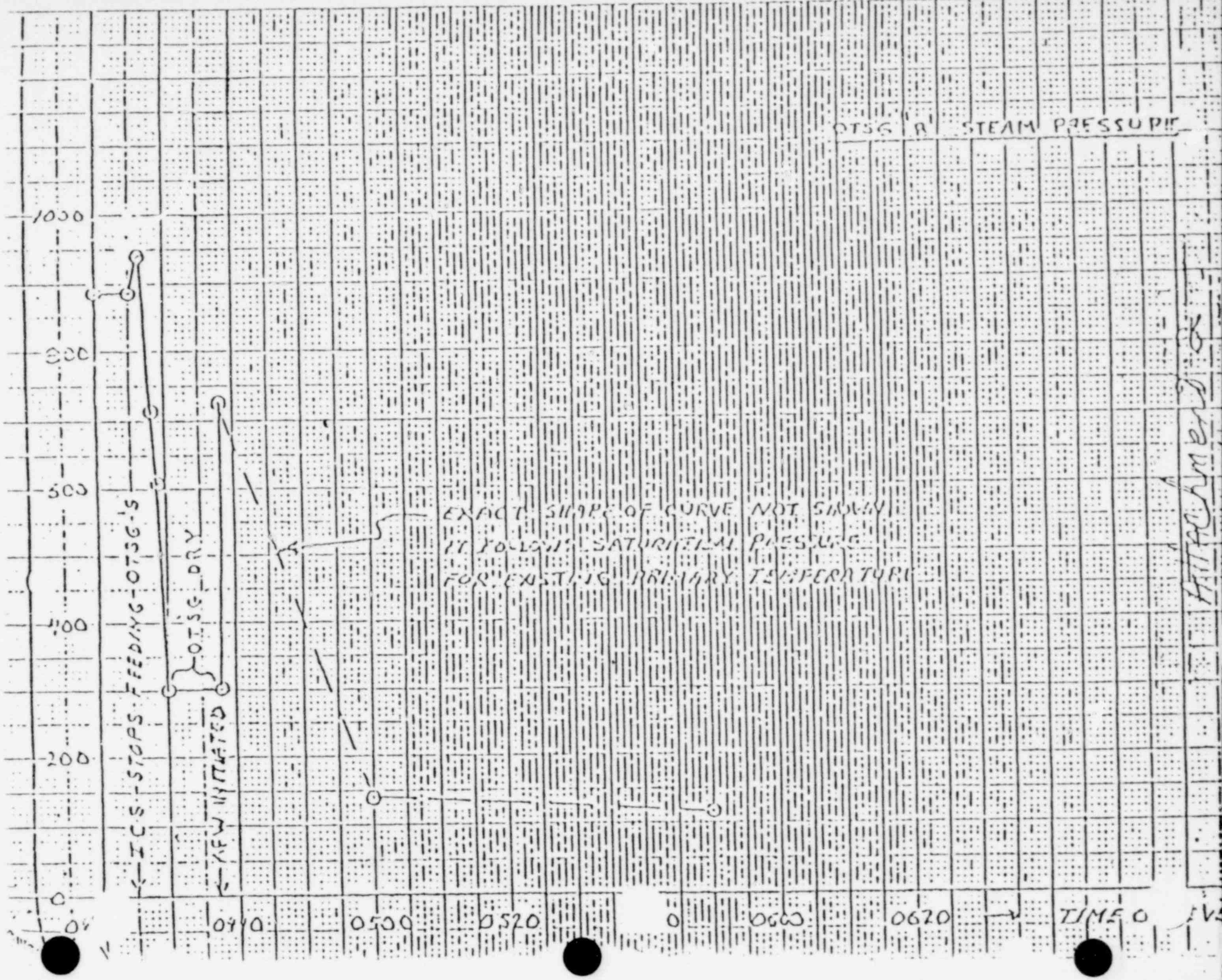

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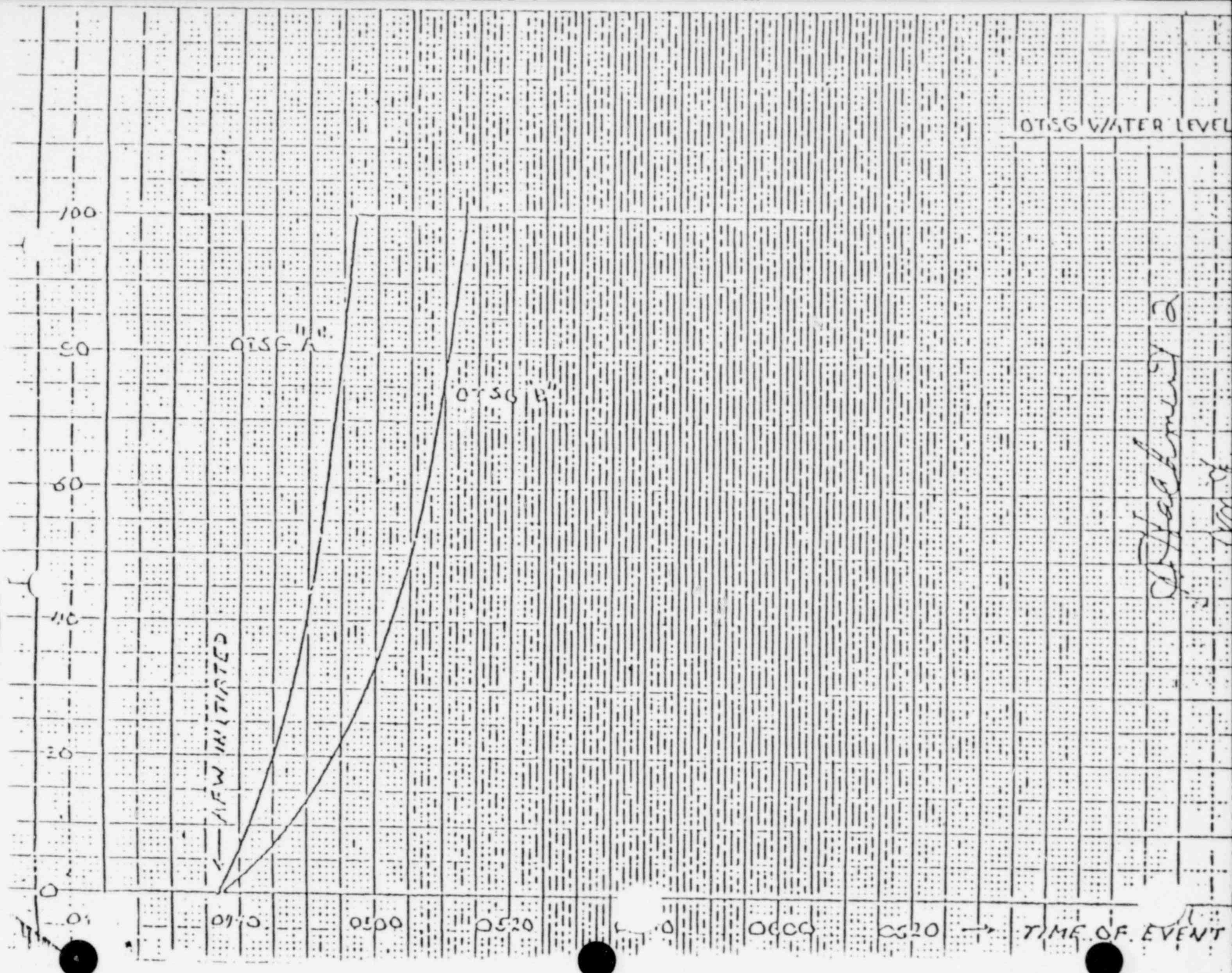

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Preliminary copy

TOLEDO EDISON

LOWELL E. ROE
Vice President
Facilities Development
(419) 259-5242

Docket No. 50-346

License No. NPF-3

Serial No. 475

December 22, 1978

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

In response to the December 20, 1978, telephone conversation between your Mr. Guy Vissing and our Mr. E. C. Novak, and the December 20, 1978 telephone conversation between NRC Region III personnel (G. Fiorelli, R. Knop, T. Tambling and J. Streeter) and our Mr. E. C. Novak, attached is an additional safety evaluation supporting continued operation of Davis-Besse Nuclear Power Station Unit 1. This additional safety evaluation supplements the analysis we provided to you by our letter dated December 11, 1978, Serial No. 471. The attached safety evaluation analyzes the transient resulting from the operator not controlling steam generator level at 35 inches in accordance with current operating procedures.

Yours very truly,

Lowell E. Roe

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LER:CRD

Enclosure

bj e/7

Docket No. 50-346
License No. NPF-3
Serial No. 475
December 22, 1978

Additional Safety Evaluation
of Transient Resulting from
Inability of Operator to Control
Steam Generator Level at 35 Inches

I. INTRODUCTION

The Davis-Besse Unit 1 Steam and Feedwater Line Rupture Control System (SFRCS) design objectives are to prevent the release of high energy steam, to automatically start auxiliary feedwater (AFW), and to provide adequate AFW, via essential steam generator level control, to remove decay heat during anticipated and design basis events when AFW is required. Table 1 correlates the station variables and accident conditions for which AFW actuation is required. For all actuation signals, the SFRCS initiates and controls AFW addition automatically to maintain a 120" level (96" indicated on the startup range instrumentation) in the steam generator

The recent natural circulation test at Davis-Besse 1 (TP800.04) demonstrate that a 35-inch (indicated) steam generator level of AFW provides adequate natural circulation for decay heat removal.

The auto essential SG level control setpoint of 120-inches (96-inch-indicat is thus in excess of minimum SG level requirements.

Operating procedures requiring manual control of steam generator level at 35-inches on the startup range level indicators following non-LOCA events were developed and used at Davis-Besse Unit 1 pending installation of permanent design changes to the SFRCS. Margin in maintenance of indicated pressurizer level and assurance of adequate natural circulation capability will exist through operator intervention during conditions where AFW is required.

Inability of the operator to comply with the present operating procedures will possibly result in a momentary loss of pressurizer level and/or level indication under certain conditions, but will not produce consequences whic are non-reversible or detrimental to safe operation of Davis-Besse Unit 1.

II. DISCUSSION

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The following section is divided into three segments: Relationship with Events Presented in the Davis-Besse Unit 1 FSAR, Loss of Offsite Power, and Loss of Feedwater.

A. Relationship with events presented in the FSAR

Addition of auxiliary feedwater at rates considerably greater than the decay heat generation rate will result in overcooling of the reactor coolant, contraction and a reduction of pressurizer level. This sequen of events is typical of several transients presented in the FSAR operator interactions. From a practical viewpoint each single discoverable possible transient cannot be analyzed and presented as a part of the FSAR analysis, but a broad variety of transients have been selected. This specific transient fits within that broad category. Each of the FSAR transients has been demonstrated to produce acceptable results.

Overcooling transients resulting from a variety of causes are described in Section 15.2.10 "Excessive Heat Removal due to Feedwater Malfunctions This section describes a transient resulting from excessive main feedwater addition, which is similar to the specific transient of increased level addition by auxiliary feedwater. Further information is presented in response to questions 15.2.15 and 15.2.16.

The steam line break (see FSAR sections 15.4.4, 15.4.8, 15.4.1) is the most severe overcooling transient,in that the reactor coolant system is decreased 50°F in average core temperature over a 30 second time period.

This is compared with the cooldown in question, which takes a much longe time o achieve a similar temperature drop and system conditions. During the steam line break, RC system pressure is reduced from 2200 psi to about 900 psi as system temperature is driven toward equilibrium with the unaffected (pressurized) steam generator attaining saturation temper ture of about 530°F. The pressurizer is near empty at about 20 seconds and thereafter loses its influence on the system, thus permitting the upper elevations of the reactor coolant loop to approach saturation as cooldown continues toward 530°F. High pressure injection (HPI) pumps ar actuated on low RC pressure such that pressurizer level will be restored As shown in Figures 15.4.4-1 and 15.4.4-2 of the Davis-Besse Unit 1 FSAR the rapid cooldown of RCS after reactor trip is limited by the pressure maintained in the pressurized steam generator in much the same fashion as anticipated for events such as the event of concern. As the RCS approaches saturation, core cooling is not impeded. Minimum DNBR>1.3 occurs just before reactor trip and subsequently increases with substant margin throughout the remainder of the cooldown.

The close relationship of the auxiliary feedwater level increase as an overcooling transient with these similar overcooling transients allows us to draw the conclusion that no unreviewed safety question exists. To show a comparison to the detailed analyses reported in the FSAR, we have performed conservative bounding analyses of two representative cases

B. Loss of Feedwater and Loss of Offsite Power

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We have analyzed two transients resulting from auxiliary feedwater addit and establishment of SG level above the operating procedure 35" limit. The two transients examined are a loss of offsite power (reactor coolant pumps stop, makeup stops, main feedwater stops) and a loss of feedwater (reactor coolant pumps continue, makeup continues).

Of these two transients the loss of feedwater results in the greater volumetric coolant contraction, because the forced coolant flow (RC Pumps operating) causes a faster rate of heat rejection to the steam generators.

1. Loss of Offsite Power

Preliminary calculations for a reactor trip following a loss of offsite power show that the pressurizer loses indication but does not empty. The assumptions used to derive this result included full runout auxiliary feedwater flow (~2400 gpm) resulting in a fill time to 120" of about 4 minutes. No net mass change to the primary coolant (no makeup, no letdown) was considered, even though the makeup controls would respond to decreasing pressurizer level by increasing the net input to above 200 gpm. At the termination of the transient the pressurizer level is slightly above the outlet into the surge line. Reactor coolant pressure reaches about 1600 p and high pressure injection may be automatically initiated.

Although the net makeup was not considered, it would in fact cause the pressurizer to refill to the normal level. At the same time compression of the steam would cause a partial repressurization of the system ensuring that the coolant remains subcooled. This tran presents no safety concerns.

2. Loss of Feedwater

This transient has a greater reactor coolant contraction than the loss of offsite power case, resulting in emptying of the pressuriz Consequently it will be described in greater detail.

A brief summary of the events is:

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• Reactor trip	Time = 0
• Makeup control valve opens wide admitting full makeup to reactor coolant system	Time = 0^+
• AFW initiated	Time $\approx$ 40 sec
• Pressurizer empties; RC system pressure slightly greater than 1800 psi	Time $\approx$ 2 min
• HPI initiated by SFAS; makeup isolated	Time $\approx$ 2^+ min
• Steam generator level = 10 ft; voids exist in reactor coolant	Time $\approx$ 4 min
• HPI inflow replaces volume occupied by voids; pressurizer level begins to be restored	Time $\approx$ 7-8 min

The major concerns that evolve from this transient are the

Steam voids will not collect in reactor coolant piping and no flow blockage will occur because of dispersal and mixing by the forced flow. DNB acceptance criterion limit will be met because the power output of the core is at the decay heat level and all reactor pumps are operating, maintaining core heat removal. We conclude that no safety problem exists.

TABLE 1: STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM (SFRCS)ACTUATION PARAMETERS

Actuation Parameter		Accident
Station Variables	Setpoint	
1. Low Steam Line Pressure	$<$ 591.6 psig[1,2]	Steam Line Break Feedwater Line Break
2. Low SG Level	$\leq$ 17 inches[1]	Loss of F/W
3. SG Pressure Minus Main Feedwater Line Pressure	$>$ 197.6 psi[1]	FWLB, LOMFW
4. Loss of All RC Pumps[3]		Loss of Off-Site Power

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NOTES:

1. When actuated, SFRCS closes both main steam isolation valves, closes both main FW control and stop valves, initiates AFW and controls AFW to maintain a 120 inch level in the SGs.

2. Alignment of AFW to a pressurized SG is provided for steam and feedwater line breaks.

3. AFW initiation but steam and feedwater line isolation does not occur.

III. Bounding Analysis of Loss of Feedwater Event With Failure of Operator to Control Feedwater Level at 35"

Introduction:

The following bounding analysis conservatively predicts the events occurring within the primary reactor coolant system and reactor following a loss of main feedwater from 100% power for the Davis-Besse Unit 1. Auxiliary feedwater control has been assumed at 10 feet within both steam generators.

Results:

Because of the conservative, bounding, nature of this calculation, the overcooling of the primary system due to auxiliary feedwater injection causes a contraction of coolant volume sufficient to create steam within the primary system. The steam is shown to be uniformly distributed within the RCS and the void fraction is 4%. The reactor coolant pumps maintain full capability. The DNB ratio is shown to exceed 2.0 and no return to criticality potential exists. Thus, during the course of the incident, no core problems develop. Further, following the time of maximum contraction, the system recovers to full pressure, pressurizer function is regained and the reactor coolant returns to a subcooled water configuration without operator action.

Analysis:

The following assumptions have been made to assure the bounding nature of the results:

Reactor Power:

100% until boiling stops in the steam generators; 0% after that time. This assumption is conservative as core heat would compensate for the cooling caused by the auxiliary feedwater.

Initial Coolant Inventories Water:

RCS = 11290 ft^3

Pressurizer = 864 ft^3

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These assumptions are nominal operating values.

Initial Temperatures:

The whole system is taken to be at $T_{average} = 582^{\circ}F$.

This assumption is a reasonable average.

Initial System Mass: ~ 500,000 lbm

The mass is figured from the temperature and volumes above.

Makeup System:

No credit is taken for additional makeup flow which will occur as the pressurizer loses level. (In all likelihood, the makeup system will contribute approximately 200 ft^3 extra liquid volume).

Local Power (kw/ft): 18.4 kw/ft

This value is taken as the maximum allowed by Technical Specifications.

Secondary Side Volume At 10 Foot Level

711 ft^3 per generator, actual volume.

Auxiliary Feedwater Flow:

166.5 ft^3/min. per generator actual value.

Auxiliary Feedwater Enthalpy:

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8 Btu/lbm lower bound for maximum cooling.

With the initiating event, loss of main feedwater, the reactor coolant system pressure will start to rise. Reactor trip will occur on high RCS pressure. Following trip, the RCS pressure will fall because core power has been reduced and boiling of residual main feedwater or auxiliary feedwater is occurring in the steam generators. These events are almost identical to those which occur in a main feed line break and are analyzed in detail in Section 15.2.8 of the FSAR.

In short order, the system will return to its initial configuration but, because the auxiliary feedwater heat absorption rate exceeds the decay heat generation rate, the RCS continues to depressurize. During this phase, residual main feedwater and injected auxiliary feedwater will be boiled and vented through the steam generator safety relief valves. The primary system average temperature will fall to the saturation temperature of water at the safety valve set pressure. At this time, primary and secondary conditions are expected to be approximately as follows:

	Primary	Secondary
Pressure	1800 psia	980 psia
Temperature	542 F	542 F
Mass	503344 lbm	0 lbm
Liquid Volume in Press.	400 ft^3	N.A.
Time into Transient	$\simeq$ 2 min.	$\simeq$ 2 min.

It is conservative to assume complete boiling of the secondary side water and complete equilibrium between primary and secondary sides, as these assumptions lead to the maximum follow on injection of auxiliary feedwater and therefore, maximum contraction. RCS pressure is held up by the steam bubble in the pressurizer.

The time has been estimated by calculating the necessary energy loss by the primary system from its initial conditions, the mass of auxiliary feedwater required to remove this energy and then dividing by the auxiliary feedwater flow rate.

$$\text{time} \sim \frac{(586 - 542)\ 503344}{(1194-8)\quad 383 \quad 62} \sim 54 \text{ sec.}$$

Six seconds was used to estimate the initial pressurization portion of the transient.

In performing the remainder of the evaluation 10 feet of cooled (40 F) auxiliary feedwater is placed in each steam generator and the thermal equilibrium condition calculated. Because after a 10 foot level is obtained this auxiliary feedwater flow stops, this condition represents the maximum contraction possible. The state variables resulting are:

	Primary	Secondary
Pressure	560 psia	560 psia
Temperature	478 F	478 F
Enthalpy of Water	462 Btu/lbm	462 Btu/lbm
Specific Volume	.020 ft^3/lbm	.020 ft^3/lbm

From the specific volume, the primary liquid volume can be calculated:

$$\text{Vol} = MV_f = 10052 \text{ ft}^3$$

As 10052 is smaller than the RCS minus pressurizer volume, the remaining volume must be filled with steam.

$$V_{st} = 10426 - 10552 = 374 \text{ ft}^3 \simeq 400 \text{ ft}^3$$

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400 ft^3 corresponds to a system void fraction of 3.8% ~ 4%, and as will be shown later, is of no consequence as far as core heating or system performance is concerned. This steam volume is larger than actually expected for two reasons: 1) some temperature difference would always exist between the primary and secondary systems, and 2) the effect of core decay heat has been ignored. Both of these would increase the primary side liquid temperature, thus increasing its volume and reducing the steam volume.

Following this state of maximum contraction, no further heat is removed from the RCS via the secondary side until the RCS rises in temperature due to decay heating; this will expand the liquid volume, compress the steam and repressurize the RCS. As no mass can be lost from the secondary system prior to achieving 980 psia the first reheating stage will end at a primary system pressure, temperature, and liquid volume of 980 psia, 542 F, 10832 ft^3. Subtracting 10426 from 10832 shows that about 400 ft^3 of fluid has been forced back into the pressurizer. Pressurizer function would then be restored (if not directly, then, by either the makeup or HPI system), the RCS subcooled and the transient ended.

Several questions exist about the transient:

I. How is the 400 ft^3 dispersed within the primary system and can that volume collect in one location? From the auxiliary feedwater flow rate, over 4 minutes are required to fill the generators. As the pressurizer has 400 ft^3 in it at 980 psia and the RCS has 400 ft^3 in it at maximum contraction, approximately 2 minutes are used to eject steam from the pressurizer to the RCS. Because this steam will be superheated when it enters the RCS it will first desuperheat and then condense at a rate governed by its expanding pressure compared to the contraction of the liquid coolant. In the time of 2 minutes the reactor coolant will have made about 8 complete circles of the primary system and the steam can be considered well mixed. As the flow velocity in the RCS will remain normal, about 25 ft/sec, steam water separation will tend not to occur. Some limited steam accumulation may occur in the upper head of the reactor vessel as in that specific location of the RCS, velocity is low.

II. How well will the pumps work? Experiments performed on steam carry over capability show that for void fractions up to 10% no loss of pump capability is observed. This is documented in Figure 5-47 of BAW-10104, "B&W's ECCS Evaluation Report With Specific Application to 177 FA Class Plants With Lower Loop Arrangement." Actually pump capability increases for the first 5% of void introduced into the system.

III. Will any return to power be encountered because of the low RCS temperature? A return to power can occur for a non-borated core at 490F. This temperature includes the assumption of the most reactive rod stuck out of the core; if that rod were taken as inserted the critical temperature would fall to at or below 400F. Although no credit was taken for HPI in calculating the RC steam volume below 1600 psia, the HPI will be injecting borated water and, therefore, preventing any return to power condition. If the primary system were to stabilize at 1600 psia and thus prevent the HPI from providing boron the RCS temperature would be at least 511F and, therefore, no return to power would be expected.

IV. Will DNB be encountered in the core? The maximum contraction condition is again:

P = 560 psia
T = 478F
a = 4%,

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and occurs at least 5 minutes after power shutdown (trip occurs very early within 10 seconds of main feedwater loss). At this time, the decay heat rate is less than 3.2% using ANS + 20% (or LOCA evaluation curve). As low pressure and high void and high power are conservative bounds a DNB evaluation was performed at:

P = 500 psia
T = corresponding saturated value
a = 8%
power = 10%
W = full volumetric flow.

The resultant DNBR was >15 in the hottest channel with maximum design conditions assumed and well within acceptable values.

V. Will any steam remain trapped in the primary system? Some may be trapped for a short period of time in the upper head of the reactor vessel but this will be of no consequence and will eventually be condensed by thermal conduction through the interfacing water.

Conclusion

The maximum contraction of the RCS water has been calculated taking no credit for mitigating systems (makeup flow, HPI) and no credit for decay heating. No adverse consequences of the transient have been shown and, therefore, this transient poses no concerns to the safe operation of the plant.

IV. CONCLUSIONS

For SFRCS actuation and fill of the steam generators to the auto-essential level control point of 120" without operator action:

- No unreviewed safety question exists
- The loss of offsite power transient will not cause the pressurizer to drain although a loss of pressurizer indicated level will occur.
- The loss of feedwater transient may result in pressurize emptying, however acceptance criteria for DNB will be met. Steam bubbles which exist in the reactor coolant for a short time will be collapsed by HPI injection. Pressurizer refilling by HPI will occur.
- No return to power will result in the long term.

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 245

Telephone: (804) 384-5111

June 12, 1978

SOM #382 620-0
12343 T3.3.
SIP #14/289

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: CRDCS Trip Breaker Maintenance

Dear Terry:

In the past, some of our plants have experienced problems with CRDCS Trip Breakers. The problems have been traced to lack of preventive maintenance. B&W suggests that a planned, carefully executed, maintenance program be established using the maintenance program outlined in the Diamond Power CRDC System Vendor Manual. Particular attention should be directed to proper cycling, cleaning, and lubrication of the breakers.

B&W further recommends that this program be scheduled at a minimum frequency of every refueling cycle and more frequently for plants during startup, when the equipment is subject to adverse environmental conditions.

Our concern is that if proper maintenance is not accomplished, additional failures will occur resulting in an NRC demand for diverse qualified trip breakers. Also, we need to prevent all failures we can to reduce the number of lost capacity days.

If we can be of further assistance, please advise.

Yours truly,

for
F. R. Faist
Site Operations Manager

FRF:IDG:nlf
cc: W. E. Spangler
R. C. Luken
R. L. Pittman
D. A. Lee
J. S. Grant, TECo
E. C. Novak, TECo
C. R. Domeck, TECo
J. G. Evans, TECo
B. R. Beyer, TECo
J. D. Lenardson, TECo
R. E. Blanchong, TECo
J. C. Buck, TECo

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0
Bam B7-11

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24

Telephone: (804) 384-5111

August 9, 1978

SOM #403 620-0014
12B22 T3.3.1
SIP #14/295

file

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: SMUD Rapid Cooldown Transient

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Dear Terry:

On March 20, 1978, Rancho Seco experienced a severe thermal transient initiat by the loss of electrical power to a substantial portion of the Non-Nuclear Instrumentation (NNI). The loss of power directly caused the loss of Control Room indication of many plant parameters, the loss of input of these parameters to the plant computer, and erroneous input signals (midrange, zero, or otherwise incorrect) to the Integrated Control System (ICS).

The plant response was not the usual transient in that the ICS responded to the erroneous input signals rather than actual plant conditions, and resulted in a Reactor Protection System (RPS) trip on high pressure. Subsequent to the Reactor Trip, the erroneous signals to the ICS contributed to the rapid cooldown of the RCS. Plant operators had extreme difficulty in determining the true status of some of the plant parameters and in controlling the plant because of the erroneous indications in the Control Room.

An investigation of the events following this loss of power points out a need for a close look at operator training and emergency operating procedures for any loss of NNI power (or portion thereof). The following recommendations are made to assist your staff in a review of training and procedures to assur proper operator action for events of this nature.

1. Operators should be trained to recognize a loss of power to all or a majority of their NNI (e.g. indicators fail to mid-range, automatic or manual transfer to alternate instrument strings brings no response). The loss of power is emphasized here rather than the failure of any one instrument or control signal which are adequately covered in current simulator training courses.

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2. Given that the operator can determine that electrical power has been lost to all or part of the NNI, he should know the location of the power supply breakers, and have a procedure available to quickly regain power.

3. If the fault cannot be cleared (i.e. the breakers to the power supplies reopen), the operator should have a list of alternate instrumentation available to him, and he should be thoroughly trained in its use. Examples are:

   a. ESFAS panels
   b. RPS panels
   c. ECI (Essential Controls and Instrumentation)
   d. SRCI (Safety Related Controls and Instrumentation)
   e. Remote shutdown panels
   f. Local gages
   g. Plant computer

4. Recognizing that no procedure can cover all possible combinations of NNI failures, the operator's response should be keyed to certain variables. If the operator realizes that he has an instrumentation problem (as opposed to a LOCA or steam line break, for example), he can limit the transient by controlling a few critical variables:

   a. Pressurizer level (via HPI or normal Makeup Pumps)
   b. RCS pressure (via Pressurizer heaters, spray, E/M relief valves, etc.)
   c. Steam Generator level (via feed flow, feedwater valves, etc.)
   d. Steam Generator pressure (via turbine bypass system)

   The pressurizer level and RCS pressure assure that the Reactor Coolant System is filled; the Steam Generator level and pressure assure adequate decay heat removal.

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Attachments 1 and 2 are provided to give a brief description of the events following this loss of NNI power at Rancho Seco. As can be seen by this transient, prompt precise operator action and the ability to recognize a loss of NNI power are critical factors in limiting the severity of a transient such as this.

If you have any questions or comments, please advise.

Yours truly,

Ivan D Green

Ivan D. Green
Site Operations Manager

IDG:TFS:nlf
encl.

ATTACHMENT 1

QUENCE OF EVENTS - SMUD 04:25 to 05:34 - MARCH 20, 1978

(Revision 1, 5/25/78)

	EVENT
5:35	- Lost NNI power supply cabinets 5, 6, & 7
	- This caused a loss of valid signals to the ICS. BTU limits ran back feedwater, resulting in a partial loss of feedwater (actual Rx power was 72%),
	- Probable opening of "B" turbine bypass valves to the condenser (timing uncertain).
25:44	- Reactor trip on high pressure, turbine trip on interlock.
	- Pressurizer code relief setting was known to be low (approximately 2225 psig). The electromatic relief was isolated due to previous leakage problems. The data indicates primary pressure went ≈2400 psig => code relief valve lifted.
	- ICS closes main control and start-up feed valves and drive main feed pumps to minimum speed following trip.
	- Decay heat and RC pumps energy removal accomplished through generators by inventory boil off and the addition of main feedwater.
26:15	- Pressurizer code relief valve reseats at approximately 2100 psig.
	- Operator starts HPI pump "B".
28:23	- Operator stops HPI pump "B".
30	- OTSG "B" pressure reaches 435 psig set-point of Steam Line Failure Logic.
	- OTSG "B" goes dry.

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- Operator increases speed of a MFP and feeds "A" OTSG. This starts RCS pressure and temperature decrease.

:25 - RC pressure ≈1900 psi

:7:16 - SFAS actuation at 1600 psig

This starts HPI, LPI and initiates emergency feed. The emergency FW pump is started and the bypass emergency FW valves are opened to full open position. The system makes no automatic attempt to control steam generator water level.

:0 - RC pressure at 1475 psig. It starts to recover from this point due to HPI. T_{ave} = 528°F.

:3:56 - "A" HPI pump secured.

:6 - LPI secured.

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49:54 - "A" HPI initiated. From this point on, the operator started and stopped HPI pumps as necessary to maintain pressurizer level.

50 - Steam Line Failure Logic closes ICS-controlle start-up feed valves to each OTSG when the corresponding OTSG pressure falls below 435 psig.

:51:25 - Secured RCP-D (T_{ave} ≈435°F)

This reduced #RCP's to three

:7:27 - OTSG "A" water level - 599.7"

Speculate that ≈2 ft. of tubes are not flooded (at top) due to steam line arrangement.

:0 - Hourly computer log print-out
Steam temp. 380°F (OTSG "B")
Steam pressure 171 psig (OTSG "B")

3:47

- O⁻G "B" level - 599.1"

- Power restored to NNI cabinets 5,6,&7

T_{ave} = 285°F

RCS Pressure ≈2000 psig

Both OTSG full level ranges pegged high

Operator begins to reduce RC pressure using pressurizer spray.

ICS closes turbine bypass valves to condenser

Operator stops emergency FW flow.

Operator stops main FW pumps.

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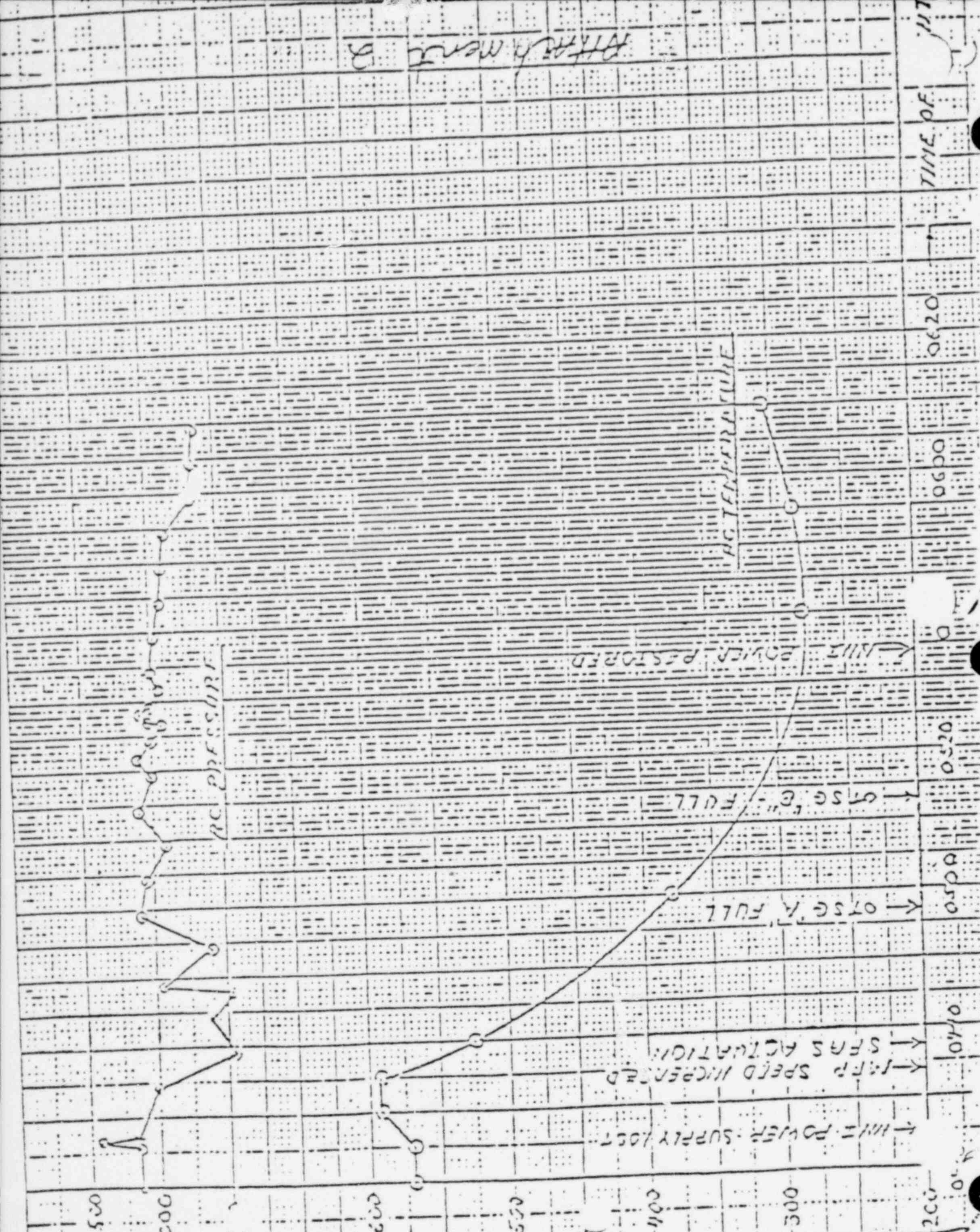
Attachment B



POOR ORIGINAL

UNITED STATES
NUCL R REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

January 8, 1979

Docket No. 50-500/501
50-329/330

MEMORANDUM FOR: J. F. Streeter, Chief, Nuclear Support Section 1

FROM: J. S. Creswell, Reactor Inspector

SUBJECT: CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

During the course of my inspections at Davis-Besse, certain issues have come to my attention which I am submitting for consideration for forwarding to the Atomic Safety and Licensing Board which has proceedings pending for the aforementioned facilities. This submittal is made pursuant to Regional Procedure 1530A (November 16, 1978), step 3 and information supplied to me per step 1. The issues for consideration are:

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight ard a hold down spring force, which resulted in a net downward 'orce at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

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The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.

It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

   Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

J. S. Creswell
Reactor Inspector

Enclosures: As stated

cc w/o enclosures:
G. Fiorelli
R. C. Knop
T. N. Tambling

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CP! ...

FEB 8 1979

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MEMORANDUM FOR: Richard C. DeYoung, Director, Division of Site Safety and Environmental Analysis

Roger J. Mattson, Director, Division of Systems Safety

THRU: Roger S. Boyd, Director, Division of Project Management

FROM: Donald J. Skovholt, Assistant Director for Quality Assurance & Operations, Division of Project Management

SUBJECT: REVIEW OF APPLICANT'S Q-LIST FOR ACCEPTABILITY

During our recent efforts to upgrade Chapter 17 of the Standard Review Plan, we identified a need for clarification of assigned branch responsibility to review for completeness and accuracy the list of systems, components, and structures (Q-List) to which the quality assurance program, developed in accordance with Appendix B to 10 CFR 50, is applicable. The need for such a Q-List is a requirement of Appendix B. SRP Sections 17.1 and 17.2 require the QAB reviewer to assure that a Q-List is either included in Chapter 17 of the SAR or appropriately referenced therein. However, the assignment of responsibility to assure that the Q-List is complete and accurate is not well in hand. Under present practices, the QAB developed a Q-List based on past SAR reviews against which each applicant's Q-List is compared. Applicants are then requested to resolve any discrepancies.

We believe that this practice needs revision. Within the staff, the persons best qualified to determine the safety significance of plant items are the technical reviewers that perform the safety evaluations of each plant. Therefore, we believe that these reviewers should be assigned primary responsibility for determining the adequacy of each Q-List, and comparison with a checklist by the QAB reviewer should be a back-up check for apparent inconsistencies or omissions.

Accordingly, we recommend that the SRP be modified to identify the need for review of the Q-List and to assign responsibility for the review. Specifically, we recommend that:

1. Section 3.2.1 of Reg. Guide 1.70 (Rev. 2) be revised to request applicants to provide a Q-List which would be referenced in Chapter 17. Guidance to the applicant should also indicate a suggested tabular format, the criteria for determining whether an item is safety-related, and the level of detail to which an item should be specified.

2. SRP Section 3.2.1 be revised to include the review of the Q-List as an area of review, and to include the assignment of primary and secondary

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branch review responsibilities to assure the accuracy and completeness of the items on this list. The latter responsibility should be consistent with the areas of review assigned to the branches through other SRP sections. This change in review responsibility is not meant to infer that the scope of the Q-List should be changed.

3. By memorandum to the QAB, each technical review branch should indicate that the adequacy of the Q-List has been verified within the assigned area of responsibility.

As an interim measure, to accommodate the current projects under review, we have proposed that each cognizant LPM issue a memorandum to the assigned reviewers requesting that the Q-List presented in the SAR be reviewed for completeness and accuracy in the areas for which each reviewer is cognizant.

At the January meeting of the LSRC, this proposal was considered. The Committee considers it to be an improvement to the review process with very nominal staff impact and recommends that the Directors of DSS, DPM and DSE approve the proposal.

Your approval is requested. We are available for further discussion of this matter at your earliest convenience.

Original Signed by.
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for Quality
Assurance and Operations
Division of Project Management

cc: D. B. Vassallo
F. Schroeder, Jr.
D. R. Muller
W. P. Haass
J. W. Gilray
R. L. Baer
O. D. Parr
J. F. Stolz
S. A. Varga

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Distribution:
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RSBoyd, PM:D
DRoss, PM:DD
DJSkovholt, PM:ADQAO
~~XXXXXXXXXXXXXX~~

FLiederbach, PM:QAB

OFFICE ▸	PM:QAB	PM:ADQAO	PM:D			

MAR 1 6 1979

Docket Nos.: 50-329
50-330

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MEMORANDUM FOR: Assigned Reviewers for Midland 1 & 2

FROM: Walter P. Haass, Chief, Quality Assurance Branch, Division of Project Management

Darl Hood, Project Manager, Light Water Reactors Branch No. 4, Division of Project Management

SUBJECT: DETERMINATION OF ACCEPTABILITY OF Q-LIST FOR MIDLAND 1 & 2

In a memorandum dated February 8, 1979 (to R. DeYoung and R. Mattson from D. Skovholt), recommendations were made regarding the formal documentation of the staff review of the appl ant's Q-list which identifies those safety-related structures, systems, and components (SSC) that fall under the control of their QA program (QAP) described in Section 17 of the SAR. The February 8 memorandum also recommended an interim procedure for accomplishing such reviews for projects currently under review. Based on oral agreement by DSS and DSE to adopt this interim procedure, this memorandum is written to request that all reviewers assigned to the Midland 1 & 2 OL application review the list given in Table 3.2-1 and other parts of Section 3.2 of the FSAR as it applies to their areas of review responsibility to determine if there is an adequate listing of those SSC that should fall under the Midland QAP. This program satisfies the provisions of Appendix B to 10 CFR 50.

Midland, in response to our question 421.2 (note Enclosure 1), has identified by reference to Section 3.2 of the FSAR those SSC classified as safety-related (Q-listed items) and which are subject to the requirements of the Midland (Consumers Power) QAP. This response should be used in conjunction with Section 3.2 of the FSAR in your evaluation.

The criterion that you should use in determining whether SSC fall under the requirements of the Appendix B QAP is as follows:

> Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Additional guidance in this regard is provided in the regulatory position of Regulatory Guide 1.29, "Seismic Design Classification."

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SURNAME						
DATE						

It is our intent to arrive at a safety-related Q-list for the Midland 1 & 2 application that is generally consistent, both in scope and level of detail, to safety-related Q-lists shown in the past OL applications. It is requested, therefore, that adequate justification be provided for substantive additions, deletions, or expansion in level of detail should such situations arise.

To facilitate your review effort, enclosed is a generic safety-related Q-list that has been developed by QAB and utilized in past reviews. This list is an outgrowth of the results of work performed by an NRC work group with members from SD, I&E, and NRR whose purpose was to develop a Regulatory Guide to identify the items under the control of the QA program. The work group has essentially turned this effort over to SD. The QAB has initiated the review process of the Midland Q-list by comparing the Midland Q-list (safety-related SSC) against the generic safety-related Q-list and indicated those items for which correspondence can be found and those items that appear to be missing. Using your copy of the enclosed Q-list, you are requested to:

1. Identify those SSC that fall within your area of review responsibility.

2. Identify those SSC that should be added or deleted from the list.

3. Provide an expansion of the level of detail where necessary.

4. Provide the justification for any changes in the list due to items (2) and (3) based on the criterion presented above.

Your response should be transmitted by memorandum to the QAB (W. Belke) by no later than April 13, 1979.

POOR ORIGINAL

Original signed by
Walter P. Haass

Walter P. Haass, Chief
Quality Assurance Branch
Division of Project Management

Original signed by
Darl Hood

Darl Hood, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

DISTRIBUTION:
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QAB Projects
QAB Chron. File
NRR Reading File
DJSkovholt, DPM
DVassallo, DPM
WBelke, QAB
JGilray, QAB
WHaass, QAB
DHood, DPM

Enclosures:
1. ~~Response to Question 421.2~~
2. Q-List

OFFICE			DPM:QAB	DPM:QAB	DPM:QAB	DPM:LWR
SURNAME			WBelke:cf	JGilray	WPHaass	DHood
DATE			3/16/79	3/16/79	3/ /79	3/16/79

CP[illegible]

[illegible] Memos
[illegible] memo of 2/4/79 to D. Young

Status

	[illegible]	[illegible]	Status
1. [illegible] (Belke)			
2. [illegible] 1 & 2 (Belke)			
3. [illegible] 1 & 2 (Belke)			
4. Comanche Peak 1 & 2 (Belke)			
5. [illegible] (Spraul)			
6. Grand Gulf 1 & 2 (Cinnury)			
7. [illegible] (Lederbeck)			
8. Haven 1 & 2 (Spraul)			
9. New Haven 1 & 2 (Lederbeck)			
10. [illegible] 1 & 2 (Belke)			
11. South Texas 1 & 2 (Spraul)			
12. Wash. [illegible] 2 (Cinnury)			
13. Watts Bar 1 & 2 (Belke)			

? [illegible]
1. [illegible]
2. Fermi 2
3. Hatch

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17.2.7 QUALITY ASSURANCE PROG

This program is applied to the safety-related items of the Station that prevent or mitigate the consequences of postulated accidents which could cause undue risk to the health and safety of the public. A summary of structures and systems covered in whole or in part by this program are identified in Table 17.2-1. The actual boundaries of these systems and structures will be specified in the Quality Assurance Systems List. The Manager-Generation Engineering, with concurrence of the Manager- Generation Quality Assurance, is responsible for development of this list.

The Manager- Generation-Quality Assurance has the direct responsibility for ensuring that this Operating Quality Assurance Program is implemented and that it provides for control of all activities affecting quality on nuclear safety related items. He is also responsible for ensuring that the program is modified and updated as standards, regulations, results, and experience dictate. The various groups involved in the Operational Quality Assurance Program, and their responsibilities, are described in 17.2.6.

The Operating Quality Assurance Program is described by written policy, plan, and procedure documents. The basic company policy is established by the President as described in 17.2.1. This Operating Quality Assurance Plan is issued by the Vice President-Generation. The procedures, which are the Operating Quality Assurance Program's detailed requirements, are originated and approved as shown in Table 17.2-2, Quality Assurance Program Procedure Categories and Approvals.

An outline of the quality assurance procedures to be used to implement the Operating Quality Assurance Program is included in 17.2.25.

The Manager- Generation Quality Assurance is responsible for maintaining a comprehensive training program for both the original and refresher training of personnel in the Operating Quality Assurance Staff. He also ensures that quality assurance indoctrination is given to Generation Department personnel who are not in the Operating Quality Assurance Staff but whose job responsibility will affect quality. The training program shall comply with Regulatory Guide 1.8 (March 1971) and Regulatory Guide 1.58 (August 1973) including applicable requirements of ANSI N45.2.6 - Rev. 1 and shall consist of lectures, formal schools, job experience, and individual study, as appropriate.

Each manager maintains formal training programs and procedures to ensure the proper job related training and qualification of his personnel. The Unit Superintendent is responsible for the indoctrination and training of unit staff personnel performing activities affecting quality or operations, and for ensuring that, where required by Technical Specification Section 6 operators are formally licensed or qualified.

All contractors who perform engineering, construction, or other technical services on structures, components, or systems are required to meet those portions of the AEC Regulation 10CFR50, Appendix B, which are applicable to their services and the materials and equipment which they supply. The Manager-Generation Engineering is responsible for ensuring that these requirements are contained in the specifications and purchase documents as appropriate along with the quality assurance safety class of the component or system involved.

TABLE 17.2-1

SUMMARY OF QUALITY ASSURANCE SYSTEMS OR PARTIAL SYSTEM LISTS

1. Control Building Heating and Ventilation System
2. Fuel Handling/Auxiliary Building Ventilation System
3. Reactor Building Emergency Cooling System
4. Reactor Building Spray System
5. Nuclear Chemical Addition and Sampling System
6. Condensate System
7. Core Flood System
8. Chilled Water System
9. Containment Monitoring System
10. Decay Heat Closed Cycle Cooling Water System
11. Emergency Diesel Generator Fuel System
12. Decay Heat Removal System
13. Decay Heat Water System
14. Emergency Feedwater System
15. Emergency Diesel Generator Services
16. Feedwater System
17. Hydrogen Purge Discharge System
18. Main Steam
19. Make-up and Purification System
20. Nuclear Services Closed Cooling River Water System
21. Nuclear Services Closed Cooling System
22. Penetration Cooling System

23. Reactor Coolant System

24. Control Rod Drive Mechanisms

25. Reactor Building Emergency River Water System

26. Spent Fuel Cooling System

27. Screen House Ventilation and River Water System

28. Waste Gas System

29. Liquid Waste Disposal System

30. Solid Waste Disposal System

31. Reactor Building Isolation System

32. 4160 & 480V Class IE Distribution System

33. Emergency Diesel Generators

34. 250/125V D.C. System

35. 120V A.C. Vital Instrumentation Distribution System

36. Reactor Protection System

37. Engineered Safeguards Actuation Systems

38. Air Intake Structure

39. Auxiliary Building

40. Fuel Handling Building

41. Control Building

42. Diesel Generator Building

43. Intermediate Building

44. Reactor Building

45. Intake Screen and Pump House

46. Nuclear Instrumentation and In-Core Monitoring System

47. Radiation Monitoring

/17.2

The description of Metropolitan Edison (Met-Ed) QA Program should be amended to include, in accordance with Part 50.34 (b) (ii), a more complete discussion of how the 18 QA criteria of 10 CFR Part 50 Appendix B will be implemented throughout the operations phase.

RESPONSE

See Section 17.2 of the FSAR.

10-2/17.2, 17.3

Met-Ed should commit to comply with the guidance contained in AEC's "Orange Book" (dated October 26, 1973), including ANSI Standards therein, or identify any exceptions and describe acceptable alternatives.

RESPONSE

See Sections 17.2 and 17.3 of the FSAR.

10-3/17.2

Met-Ed should identify a position within its organization with the assigned responsibility for the review of and concurrence with QA Program(s) developed and/or implemented for them by other organizations, including GPU Service Corporation (GPUSC).

RESPONSE

Resolved concurrently with Question 10-1.

10-1/17.2.18

Although Met-Ed has provided a description of its audit program, the description should include provisions for the conduct of periodic audits, by Met-Ed, of the implementation of the QA Program activities delegated to GPUSC and other major organizational contractors.

RESPONSE

Resolved concurrently with Question 10-1.

10-5/17.2.18

The scope and frequency of the Met-Ed and GPUSC internal and external audit programs should be provided.

RESPONSE

Resolved concurrently with Question 10-1.

10-4 17.2.1.3

Position qualifications should be provided for Met-Ed's Manager of Operational Quality Assurance.

RESPONSE

Resolved concurrently with Question 10-1.

10-7/17.2.2

The structures, systems and components covered by the QA Program should be identified or cross-referenced in Chapter 17.

RESPONSE

Resolved concurrently with Question 10-1.

10-8/17.2, 17.3

A listing of the titles of QA procedures applicable to Sections 17.2 and 17.3 should be provided along with a brief abstract of their scope and purposes; and a matrix or tabulation relating these to the 18 QA requirements of 10 CFR Part 50 Appendix B.

RESPONSE

See Sections 17.2 and 17.3 of the FSAR.

When using 17.2.25 and Table 17.3-1, it should be noted that:

1. Design, procurement, and installation of items during the Startup and Test Phase, with the exception of test equipment and nonsafety related supplies are handled in accordance with the Design and Construction or Operational QA Programs covered in Sections 17.1 and 17.2, and hence, are not described in the Table.

2. The Met-Ed Operational QA Plan and supporting plant operating, maintenance, and QA procedures apply to the activities of Met-Ed personnel in support of the Startup and Test Program, but are not covered in the Table since the Operational QA program is covered in Section 17.2.

10-9/17.2, 17.3

ovisions for the indoctrination and training program for QA activities ould be described.

RESPONSE

ee Sections 17.2 and 17.3 of the FSAR.

10-10/17.2.10

Discussion of the inspection program during plant turnover and during the operations phase is inadequate. Criteria/bases should be included denoting when inspections are required and when they are not required.

RESPONSE

Resolved concurrently with Question 10-1.

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

JUN 30 1975

Docket No. 50-320

V. A. Moore, Assistant Director for Light Water Reactors, Group 2, RL

SAFETY EVALUATION REPORT: THREE MILE ISLAND NUCLEAR STATION, UNIT 2, QUALITY ASSURANCE BRANCH

Plant Name: Three Mile Island Nuclear Station, Unit 2
Licensing Stage: OL
Docket No. 50-320
Responsible Branch: LWR #2-2
Project Manager: B. Washburn
Requested Completion Date: June 27, 1975
Applicant's Response Date: N/A
Description of Response: N/A
Review Status: Complete

The QA Branch has reviewed and evaluated Sections 13.1, 13.4, 13.6, and 17 of the FSAR (through Amendment 28) for Three Mile Island Nuclear Station, Unit 2. Our SER input for Section 13.6 and 17 is enclosed.

We have not included an input for Sections 13.1 and 13.4 for the following reasons:

1. The applicant has not been responsive to our requests for information relative to his offsite technical support for the operation of Three Mile Island, Unit 2. We are therefore unable to reach a conclusion as to the acceptability of this technical support.

2. In Amendment 28, the applicant revised the description of his plant staff. This revision deletes the number of persons assigned to each plant staff position. We are therefore unable to reach a conclusion as to the acceptability of the plant staff.

   The applicant has been advised of these two deficiencies.

3. The applicant has submitted a proposed revision to the review and audit provisions of Section 6.0 of the technical specifications for Three Mile Island 1. We are reviewing this submittal for conformance to the Regulatory position set forth in Regulatory Guide 1.33 and for consistency with Section 6.0 of the NRC Standard Technical Specifications.

Upon staff approval of this revision, the applicant will amend the Three Mile Island 2 application to include these review and audit provisions. We consider this to be an acceptable approach.

An SER supplement will be issued when the above matters are resolved.

Donald J. Skovholt

Donald J. Skovholt, Assistant Director
for Quality Assurance & Operations
Division of Reactor Licensing

Enclosure:
As stated

cc: w/o enclosure
W. McDonald

w/enclosure
K. Kniel
B. Washburn

QA SER Input
from QAB
6/30/75

SAFETY EVALUATION REPORT

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

OPERATIONAL QUALITY ASSURANCE PROGRAM

13.0 ~~Conduct of Operations~~

13.6 Plant Records

The applicant has described his program for maintaining plant records and has committed to maintaining records according to ANSI N45.2.9-1974, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants." Specific records and their retention periods will be delineated in the facility technical specifications.

Based on our review, we conclude that the applicant's provisions for maintaining records meet the position described in ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are satisfactory.

17.2 Quality Assurance For Operations

Organization

Metropolitan Edison Company (MET-ED) has established an organization which is responsible for establishing and implementing the operational QA program for the Three Mile Island Nuclear Station,Unit 2. The President of MET-ED has delegated to the Operational QA Manager, through the Vice President-Generation, the responsibility for establishing and implementing the QA program. As shown in Figure 1, the Operational QA Manager has equal organizational level with the Managers of Engineering, Nuclear Generating Stations, and Maintenance. The onsite Plant QA Supervisor and QA Specialists are under the direct control of the Operational QA Manager.

The qualifications, duties, responsibilities, and authority for the various individual positions performing QA functions have been adequately described and are acceptable. The Operational QA Manager has the specific responsibility to develop, implement, and maintain the operational QA program and manual. QA program procedures are reviewed and approved by the Operational QA Manager. QA related procedures, originated by other MET-ED organizations, are reviewed and approved by the respective organizations and reviewed and concurred in by the Operational QA Manager. To assure continuous implementation of the QA program policies and procedures, the Operational QA Manager conducts a system of preplanned audits, inspections, and review activities. In addition, the Vice President-Generation performs a review and audit evaluation of the QA program effectiveness at least every two years and reports the results to the MET-ED President. We find that the QA organization has adequate authority to identify quality problems; initiate, recommend or provide solutions; and verify implementation of corrective action for nonconforming items or activities. This authority includes the right to stop work.

Based on our evaluation, we conclude that the MET-ED QA organization has the sufficient organizational freedom necessary to effectively execute their QA responsibilities without undue influences of cost and schedule. We have therefore determined that this organizational arrangement is acceptable and complies with the requirements of Appendix B to 10 CFR Part 50.

Quality Assurance Program

MET-ED has committed in the FSAR to structure and implement their QA program in accordance with the NRC guidelines contained in NRC documents WASH 1284, "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," WASH 13 "Guidance on Quality Assurance Requirements During the Constr on Phase of Nuclear Power Plants," and WASH 1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," with the exception of certain areas which are described by alternatives which we have evaluated and found acceptable. The QA program provides for a formal training program for those personnel performing QA related activities to assure they are knowledgeable as to the proper interpretation and implementation of the QA manual including its requirements and implementing procedures. In addition, the QA program provides for the necessary controlled procedures which describe how each of the eighteen criteria of Appendix B to 10 CFR Part 50 will be complied with. MET-ED requires a formalized inspection program to be established and implemented by qualified QA personnel independent of the personnel or group performing the work being inspected. This also applies to procurement sources. Provisions are provided to assure inspection instructions describe the method of inspection, the accept and reject criteria, and the degree of documenting and verifying the inspection results.

The audit program provides for regularly scheduled audits of the operation of Three Mile Island Unit 2 and for the prompt reporting of audit results and corrective actions to responsible management levels for their review and assessment. The audit program is under the direction of the Operational QA Department, the Plant Operations Review Committee and the General Office Review Board. To assure proper visibility of problem areas and implementation of corrective action, audit results are distributed to responsible members of management. In addition to the audit program, the Vice President-Generation performs an independent review of the QA program procedures and activities at least once every two years to assure that the QA program is meaningful and effective.

Conclusion

In summary, the staff has determined that MET-ED's QA program for Three Mile Island Nuclear Station, Unit 2, as described in the FSAR through Amendment 28, provides a comprehensive system of planned and systematic controls which adequately demonstrate compliance with each of the eighteen criteria of Appendix B to 10 CFR Part 50. In addition, MET-ED has described an acceptable QA organization which has sufficient authority and independence to permit effective implementation of their QA program without undue influences from costs and schedules. We therefore conclude that the MET-ED QA program is acceptable for control of the quality related activities during the operational phase of the Three Mile Island Nuclear Station, Unit 2.

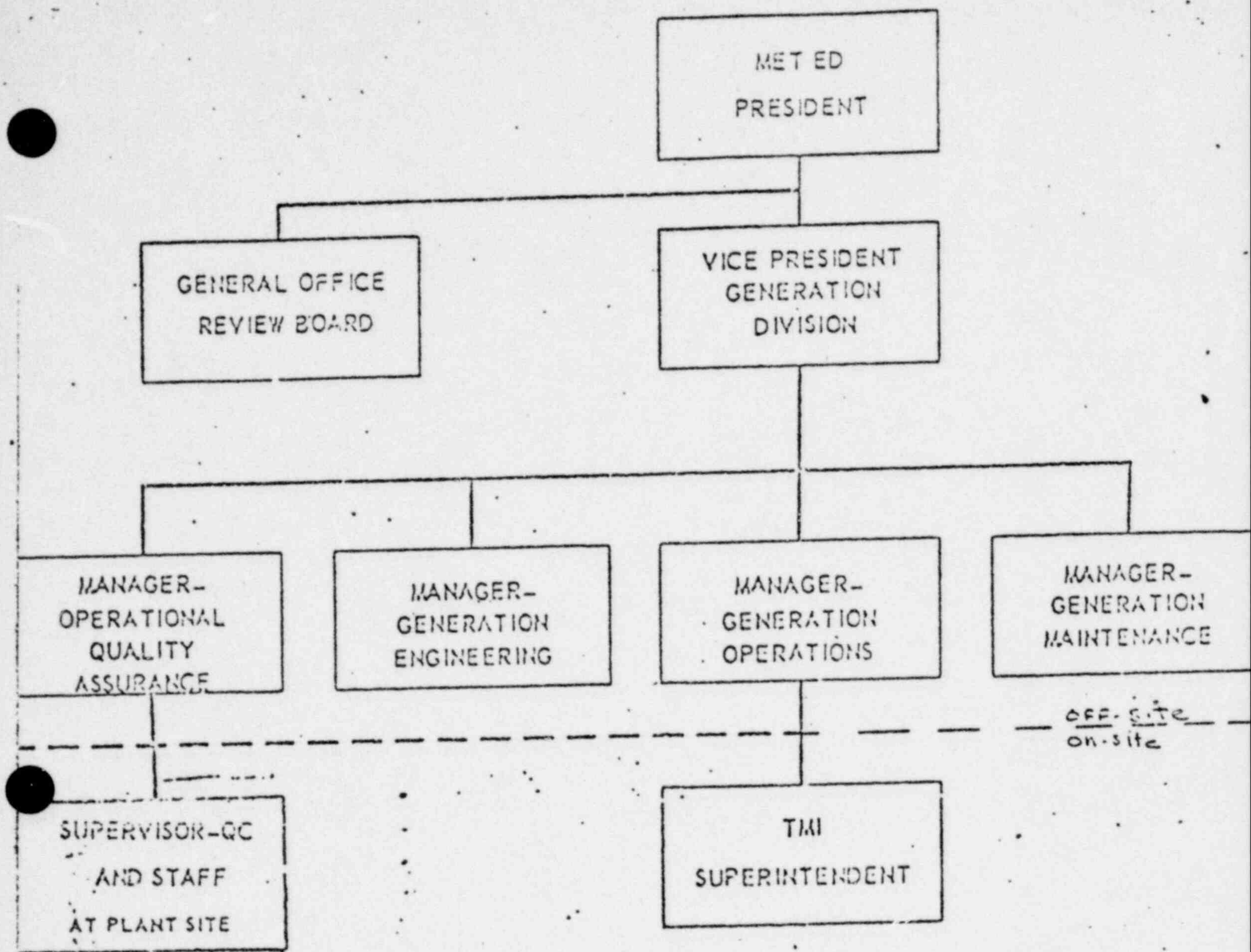

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FIGURE 1
ORGANIZATION CHART
THREE MILE ISLAND NUCLEAR STATION UNIT 2

41.1 (RSP)/17.2.7

Met-Ed is requested to amend section 17.2.7 of its description of the QA Program for Station Operation for Unit 2 to include a statement which commits to comply with, the "Gray Book", WASH-1283 Revision 1, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants - Revision 1," dated May 24, 1974, and the "Green Book", WASH-1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," dated May 10, 1974, for those design procurement, and construction activities which may occur during the operations phase and for which the guidance is applicable, or identify any exceptions to the guidance contained within "Gray Book and/or the "Green Book" and described acceptable alternates.

RESPONSE

See revised 17.2.7.

POOR ORIGINAL

17.0 QUALITY ASSURANCE

17.1 Organization

The applicant has established an organization which is responsible for establishing and implementing the operational Quality Assurance program for the Three Mile Island Nuclear Station, Unit 2. The President of Metropolitan Edison Company has delegated to the Operational Quality Assurance Manager, through the Vice President and Manager, Generation Division, the responsibility for establishing and implementing the Quality Assurance program. As shown in Figure 17.1, the Operational Quality Assurance Manager has equal organizational level with the Managers of Engineering, Nuclear Generating Stations, and Maintenance. The onsite Plant Quality Assurance Supervisor and Quality Assurance Specialists are under the direct control of the Operational Quality Manager.

The qualifications, duties, responsibilities, and authority for the various individual positions performing Quality Assurance functions have been adequately described and are acceptable. The Operational Quality Assurance Manager has the specific responsibility to develop, implement, and maintain the operational Quality Assurance program and manual. Quality Assurance program procedures are reviewed and approved by the Operational Quality Assurance Manager. Quality Assurance related procedures, originated by other Metropolitan Edison Company organizations, are reviewed and approved by the respective organizations and reviewed and concurrred in by the Operational Quality Assurance Manager. To assure continuous implementation of the Quality Assurance program policies and procedures, the Operational Quality Assurance Manager conducts a system of preplanned audits, inspections, and review activities. In addition, the Vice President and Manager, Generation Division performs a review and audit evaluation of the Quality Assurance program effectiveness at least every two years and reports the results to the Metropolitan Edision Company President. We find that the Quality Assurance organization has adequate authority to identify quality problems, to initiate, recommend or provide solutions, and to verify implementation of corrective action for nonconforming items or activities. This authority includes the right to stop work.

Based on our evaluation, we conclude that the Quality Assurance organization has sufficient organizational freedom necessary to execute effectively their quality assurance responsibilities without undue influences of cost and schedule. We have therefore determined that this organizational arrangement is acceptable and complies with the requirements of Appendix B to 10 CFR Part 50.

17.2 Quality Assurance Program

The applicant has committed in the Final Safety Analysis Report to structure and implement their Quality Assurance program in accordance with the guidelines contained in Nuclear Regulatory Commission documents WASH-12 , "Guidance on Quality Assurance

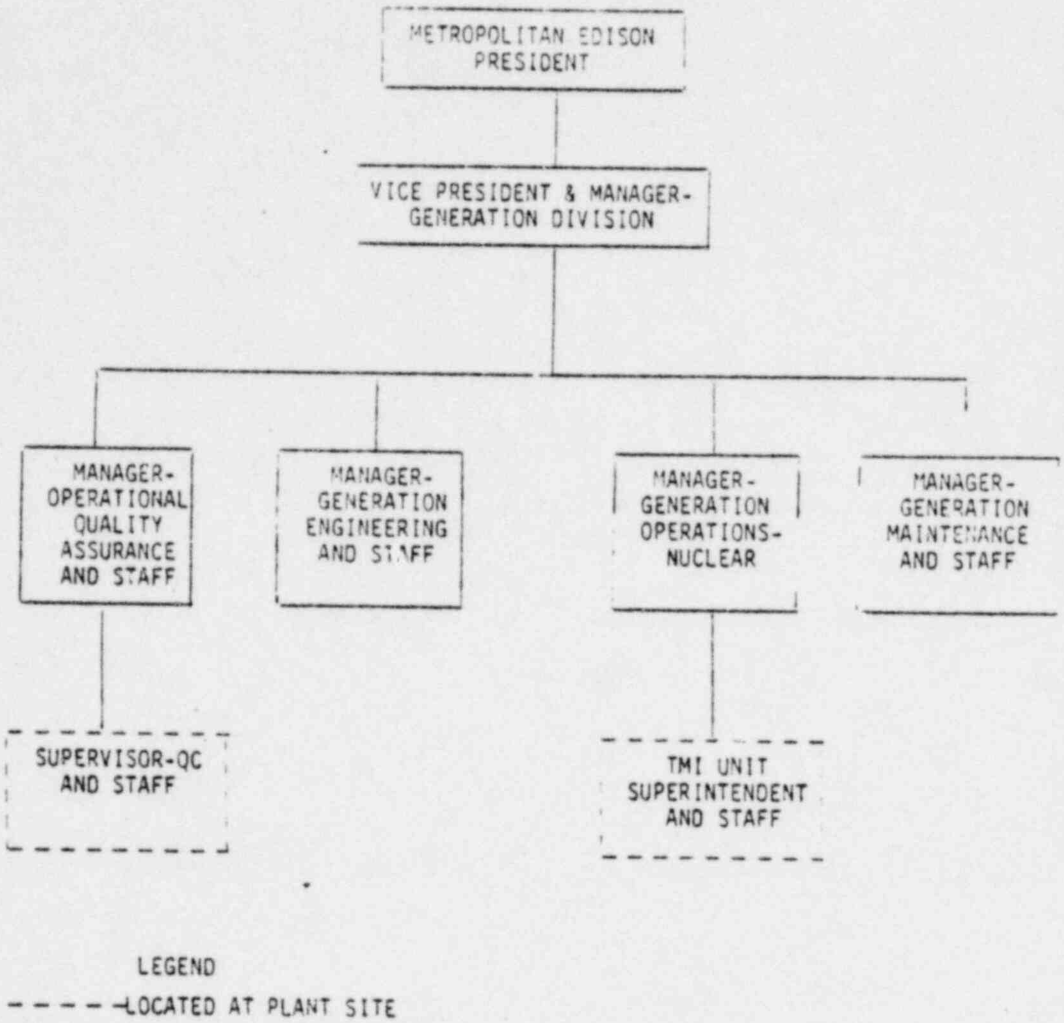
Requirements During the Operations Phase of Nuclear Power Plants," WASH-1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," and WASH-1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants." The Quality Assurance program provides for a formal training program for those personnel performing Quality Assurance related activities to assure they are knowledgeable as to the proper interpretation and implementation of the Quality Assurance manual including its requirements and implementing procedures. In addition, the Quality Assurance program provides for the necessary controlled procedures which describe how each of the eighteen criteria of Appendix B to 10 CFR Part 50 will be complied with. The applicant requires a formalized inspection program to be established and implemented by qualified Quality Assurance personnel independent of the personnel or group performing the work being inspected. This also applies to procurement sources. Provisions are provided to assure that inspection instructions describe the method of inspection, the acceptance and rejection criteria, and the degree of documenting and verifying the inspection results.

The audit program provides for regularly scheduled audits of the operation of Three Mile Island Unit 2 and for the ~~prompt reporting~~ of audit results and corrective actions to responsible managment levels for their review and assessment. The audit program is under the direction of the Operational Quality Assurance Department, the Plant Operations Review Committee and the General Office Review Board. To assure proper visibility of problem areas and implementation of corrective action, audit results are distributed to responsible members of management. In addition to the audit program, the Vice President and Manager, Generation Division performs an independent review of the Quality Assurance program procedures and activities at least once every two years to assure that the program is meaningful and effective.

To add further assurance that the applicant can and will carry out the Quality Assurance program for Three Mile Island Unit 2 in a satisfactory manner, the Metropolitan Edison Company has demonstrated the capability to implement the Quality Assurance program on Unit 1 satisfactorily.

Conclusions

The staff has determined that the applicant's Quality Assurance program for Three Mile Island Nuclear Station, Unit 2, as described in the Final Safety Analysis Report through Amendment 43, provides a comprehensive system of planned and systematic controls which adequately demonstrate compliance with each of the eighteen criteria of Appendix B to 10 CFR 50. In addition, the applicant has described an acceptable Quality Assurance organization which has sufficient authority and independence to permit effective implementation of their Quality Assurance program without undue influence from costs and schedules. We therefore conclude that the Quality Assurance program is acceptable for control of the quality-related activities during the operational phase of the Three Mile Island Nuclear Station, Unit 2.



ORGANIZATION OF GENERATION DIVISION
THREE MILE ISLAND NUCLEAR STATION UNIT 2

Figure 17.1

POOR ORIGINAL

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

Docket No. 50-320

NOV 4 1976

MEMORANDUM FOR: S. Varga, Chief, Light Water Reactors Branch #4, PM

FROM: C. J. Heltemes, Jr., Chief, Quality Assurance Branch, PM

SUBJECT: REVIEW OF METROPOLITAN EDISON COMPANY'S AMENDMENT FOR QA TO THREE MILE ISLAND, UNIT NO. 2, NUCLEAR POWER STATION PSAR

The Quality Assurance Branch has performed a review and evaluation of the information submitted by Metropolitan Edison Company for the Three Mile Island Unit No. 2 Nuclear Power Station PSAR as contained in Amendment 43 dated July 15, 1976. As a result of our review and evaluation of Amendment 43, we conclude that these changes within the QA organization (PSAR Section 17.1) satisfy our requirements. We therefore find Amendment 43 to the QA Program PSAR acceptable for the design and construction phase of Three Mile Island Unit No. 2. This amendment does not warrant a change to the QA section of the SER.

C. J. Heltemes, Jr., Chief
Quality Assurance Branch
Division of Project Management

NUCLEAR REGULATORY COMMISSI
WASHINGTON, D. C. 20555

NOV 1 6 1976

Docket No. 50-320

MEMORANDUM FOR: S. A. Varga, Chief, Light Water Reactors Branch #4, PM

FROM: C. J. Heltemes, Jr., Chief, Quality Assurance Branch, PM

SUBJECT: REVIEW OF METROPOLITAN EDISON COMPANY'S AMENDMENT FOR QA TO THREE MILE ISLAND, UNIT NO. 2, NUCLEAR POWER STATION PSAR

The Quality Assurance Branch has performed a review and evaluation of the information submitted by Metropolitan Edison Company for the Three Mile Island Unit No. 2 Nuclear Power Station PSAR as contained in Amendment 44 dated September 3, 1976. As a result of our review and evaluation of Amendment 44, we conclude that these changes within the QA organization (PSAR Section 17.1) satisfy our requirements. We therefore find Amendment 44 to the QA Program PSAR acceptable for the design and construction phase of Three Mile Island Unit No. 2. This amendment does not warrant a change to the QA section of the SER.

C. J. Heltemes, Jr.
C. J. Heltemes, Jr., Chief
Quality Assurance Branch
Division of Project Management

cc: H. Silver. PM

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CJHeltemes, PM
WLBelke, PM

Docket No. 50-320

FEB 16 1978

MEMORANDUM FOR: S. A. Varga, Chief, Light Water Reactors Branch #4, DPM

FROM: C. J. Heltemes, Jr., Chief, Quality Assurance Branch DPM

SUBJECT: REVIEW OF METROPOLITAN EDISON COMPANY'S AMENDMENT FOR QA TO THREE MILE ISLAND, UNIT NO. 2, NUCLEAR POWER STATION FSAR

The Quality Assurance Branch has performed a review and evaluation of the information submitted by Metropolitan Edison Company for the Three Mile Island Unit No. 2 Nuclear Power Station FSAR in Amendment 63 dated February 10, 1978. As a result of our review and evaluation of Amendment 63, we conclude that this change relative to the QA program, satisfies our requirements. We therefore find the quality-related changes described in Amendment 63 acceptable for the operational phase of Three Mile Island Unit No. 2. This amendment does not warrant a change to the QA section of the SER.

Original signed by:
C. J. Heltemes, Jr.

C. J. Heltemes, Jr., Chief
Quality Assurance Branch
Division of Project Management

cc: H. Silver
G. Zwetzig

OFFICE ➤	PM:QAB	PM:QAB	PM:QAB			
SURNAME ➤	WLBelke:jk	JWGilray	CJHeltemes			
DATE ➤	2/15/78	2/15/78	2/15/78			

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

NRR STATUS REPORT
ON FEEDWATER TRANSIENTS
IN B&W PLANTS

APRIL 25, 1979

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Statement of Problem

This paper considers the sensitivity of B&W plants to feedwater transients, and the role that this sensitivity might play as a precursor or contributor to TMI-2 type of accident. We examine the sequence of events that accompanies typical B&W feedwater transients and the role that control and safety equipment plays. We identify some design and analysis deficiencies of this class of plant and note some possible remedial measures.

There are several design differences that distinguish a B&W plant in its response to feedwater transients:

a. The mass of liquid in the secondary side of the steam generator is less than that for other PWRs. More importantly, the B&W design operates as a superheat boiler. Thus, the steam generator tubes are uncovered for a major portion of their length in steady operation. In this mode, changes in feed flow are quickly manifested as changes in heat transfer from the primary system. In this manner, absent prompt and remedial action by the control system (and in some cases a safety system), the steam generator will dry out.

b. The integrated control system is more complex than other designs and has a greater burden placed on it in terms of fast response.

c. The B&W design does not have reactor trips generated from the secondary side of the plant (for example, low steam generator level). Thus the steam generator level may drop somewhat on afeedwater transient before the reactor trips, on high pressure. (At this point, following reactor trip, the control system may overcompensate and cool to an excessive degree, with wide swings in pressure, pressurizer level, and temperature.)

In consideration of these design differences, we are concerned that a transient with a delayed or total failure of auxiliary feedwater may progress into a steam generator dryout condition. Once the steam generator substantially dries out, the reactor system will heat up. The potential for voids in the primary system increases. The reactor pressure may go up to the point where the PORV lifts. Eventually, if natural circulation is not restored or if auxiliary feedwater is not made effective, then core cooling will be dependent on initiation (manually) of the high pressure injection (HPI) system of ECCS. It is this degraded sequence which is the subject of this paper.

1.2 Meeting on April 24, 1979

We met with B&W and four utilities (Duke Power, SMUD, Toledo Edison, and AP and L) on April 24, 1979 to discuss several matters related to core coolability. We discussed the arrival rate of challenging transients, the role of the control system in responding to these transients, the analyses that exist on these transients, the mitigating equipment for plant transients, and finally we asked the utilities to propose remedial measures that might tend to make AFW more reliable such that core coolability is not so dependent on ECCS for anticipated transients.

1.3 Defense in Depth

During normal operation the reactor is cooled by the main feedwater system. This system is fairly reliable; if this were not so, the plants would not be able to produce reliable electric power. In the event of disruption of this normal cooling source, each PWR is provided with an auxiliary feedwater system. These systems differ in redundancy (some are redundant, and some are not), actuation (some are manual, and some are automatic), and in coupling with control systems (some failure modes of the B&W control function may inhibit AFW). Provided that AFW does come on, the reactor is expected to be cooled, by natural circulation if necessary. Representative tests in the natural circulation mode have been run on PWRs in the past. If AFW is not supplied, or if it is supplied too late and the natural circulation path is inhibited by voids and gases, then the system will boil off intermittently until either the HPI is initiated manually or later automatically (perhaps). If HPI is initiated, this system could operate in the inventory mode (since there is no LOCA) and balance losses through relief and safety valves. This mode of core cooling needs to be confirmed by further analyses (Section 3).

On the face of it there are thus three main systems that could remove heat from the core: main feedwater system; auxiliary feedwater, and HPI. The AFW and HPI are discussed further in Chapters 2 and 3.

1.4 Conclusions

The question we address in this paper is whether there is reasonable assurance of protection of the public health and safety in continued operation of B&W plants pending improvements related to feedwater transients such as: (1) further analyses and tests on transient performance; (2) a failure modes and effects analysis on the Integrated Control System; (3) system design changes based on the results of these first items; (4) design and installation of additional reactor trip circuits for faults originating in the secondary side of the system; and (5) operator training, including stationing of a full-time dedicated operator assigned to take any needed prompt manual actions. We have considered three alternatives (and they are documented in further detail in Chapter 4):

1. Issue further bulletins to obtain more knowledge about the four items listed above, and implement design and procedural changes on a schedule consistent with the arrival of and evaluation of information.
2. Specify needed design and procedural changes now, and place continued operation as being contingent on implementation within a specified period of time.
3. Require plant shutdown until satisfactory answers to the items 1-4 are provided and evaluated.

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These alternatives have been evaluated solely on the basis of safety considerations; i.e., whether there is adequate assurance that the facilities can be operated without endangering the health and safety of the public. We considered the following questions:

1. Do challenging transients arrive at a frequency high enough to be of concern?

   Our answer is yes (Section 2.3.1)

2. Does the ICS perform satisfactorily?

   a. B&W has stated and we agree, that "we are not satisfied with the reliability of the integrated control system".

   b. The failure modes and effects have not been systematically analyzed (Section 2.3.5).

   c. The ICS may initiate a feedwater transient (on the order of 10-15% of all events in the past).

   d. The ICS controls AFW in some plants (Section 2.2.5) and could contribute to loss of AFW.

   e. Even when the ICS works well there may be, in response to a feedwater transient, wide swings in reactor pressure, pressurizer level, and average reactor coolant temperature.

3. Is the system response to loss-of-feedwater transient well known? Again, we split our answer in several parts:

   a. Detailed analyses on loss or delay of AFW, with or without PORV operation, of the system response have not yet been made available to us (Section 3.1).

   b. For very small breaks (e.g., stuck-open PORV) the role of HPI in maintaining core cooling is not well analyzed (Section 3.2)

   c. The heat removal path by natural circulation is not well understood, especially when it is aggravated by void formation (Section 3.3).

4. Are the plant mitigating systems (AFW, ECCS) generally reliable? Our answer is that in most plants these systems are reliable; i.e., state of the art (Section 2.4.2. An exception is the AFW systems which are active at Oconee, which have only one pump per unit. Some other old B&W plants have lesser single failure vulnerabilities.

On the basis of the foregoing it appears that Alternative 1 should not be selected. There is too much unknown about the two items (ICS, plant transient response) to await the several months necessary to generate and evaluate the information.

Thus the choices are whether to shut down the plants now (for one or more months) or whether remedial measures exist or can be generated shortly so that interim operation poses no undue risk.

We asked the industry to propose remedial measures, and have received little to date. We note that Duke Power is considering some AFW redundancy measures (Section 2.3.3). Remedial measures could include improved operator manning, partial power or other changes to increase the thermal margin of reactor operations to reduce the boil-off rate of the steam generator and subsequent core heatup rate); increased testing of AFW; or, in the case of Oconee, perhaps full-time operation of one AFW; removal of AFW from ICS control, if possible, and placement on a separate and independent control system of high reliability; escalated delivery of analyses~~.~~ However, we believe that our role is to diagnose the ailment (this we have done); it is up to the utilities to propose the treatment.

We conclude that we do not now ha : reasonable assurance that these B&W plants can continue to operate without undue risk. We believe that these plants should be shutdown now, and that the following information is necessary before restart can be permitted.

In the short-term, we must take all reasonable steps to reduce the likelihood of occurrence of transients at B&W plants and to improve standing instructions, training and emergency procedures available to plant operators. This can be accomplished by:

a. Reviewing and upgrading, as appropriate, auxiliary feed reliability and performance (timeliness);

b. Reviewing results of FMEA analysis of ICS and taking actions, as to reduce its likelihood of initiating or exacerbating transients;

c. Hard wiring anticipatory scram based on FW transients;

d. Reviewing detailed analyses of plant response to transients to effects of HPI injection, and return to natural circulation cooling and

e. Reviewing new and augmented standing instructions and emergency procedures for plant operators deve ed as a result of a-d above, and training plant operators and the new and augmented instructions and procedures including the stationing of a full-time dedicated operator to take appropriate prompt manual actions.

In the long-term, we must either reduce the sensitivity of the response of B&W plants to transients by design changes, or substantially upgrade the instrumentation and controls available to the plant operator and substantially upgrade plant operator education training and experience.

2. AUXILIARY FEEDWATER REQUIREMENTS

2.1 Overview

The auxiliary feedwater system (AFW) requirements are related to its performance and reliability. In this context, reliability measures the probability that the system will function when called upon, whereas performance measures the adequacy of the amount, rate, and timeliness of the water actually supplied to the steam generators.

Both the performance and the reliability of installed AFW systems vary from plant to plant. The principal differences are related to (1) differences in plant parameters, (2) differences in system configurations, and (3) differences in regulatory requirements over the years. The characteristics of AFW in the operating B&W plants are given in Table 2.1. The AFW is not in the B&W scope of supply, so the different plants have quite different AFW configurations, as is evident from the table.

2.2 Performance

The performance requirements of an AFW are derived from its design basis and the assumptions made. Loss of main feedwater (LOFW) is the initiating event. The steam generator inventory decreases at a rate determined by the heat input rate, the heat removal rate

April 25, 1979

TABLE 2.1 AFW SYSTEMS

	OCONEE	CRYSTAL RIVER	RANCHO SECO	DAVIS BESSE	ARKANSAS
Auto FW Isolation Signal	None	Steam Line Failure Matrix. Closes FW block valve at P<600 psig. (Includes AFW valves.) Also MSIV's. (Isolates faulted steam generator only.)	MSL Failure-Logic: Isolates main FW from faulted SG at P<435 Psig	STM & FW Rupture Control System (IE) (1) Stm P-FW P≥170 psi (2) Steam Generator Low Level (3) Loss of all RCP's (Power Monitor) (4) Low Steam Generator Pressure (600 psig) (1) or (2) or (4) isolates main FW to both SG's, closes MSIV's (4) Also aligns both AFW PPs to the good SG. (1) or (2) or (3) or (4) starts both AFW PPs	Steam line break inst. & control (SLBIC) isolates both steam generators' main FW & MSIV's at ≤600 psig in either SG. Does not isolate emergency FW. (SLBIC is IE)
Auxiliary Feedwater					System is seismically designed. Valves are Class IE; most instr. is not IE.
Pumps: Type/No./Strainers	(Emergency FW pumps) located Turbine Bldg. 2 floors under grade. centrifugal/1 per unit/No	Located near grade level in intermediate bldg. (Seismic category I). Centrifugal/2/No	Located at CST in Missile Enclosure centrifugal/2/No	Does not start on SFAS. centrifugal/2/suction strainers	"EMERGENCY" FW on this plant centrifugal/2/none
Drive: Type	steam driven	1-motor driven 1-steam driven	1-motor driven 1-motor & turbine tandem	(800 HP) turbines (Terry/Woodward)	1-Turbine (Terry) 1-Motor (Normal p supply not Class IE. Can be put on Class IE-15 min.)
Supply/Exhaust	Main Steam/Atmosphere (≥10 min.)	Motor: Class IE Main steam either SG upstream MSIV/Atmosphere	Motors-Class IE steam from MSL/Atmosphere	main steam/atmosphere	main steam/atmosphere

Orientation of Pumps (Self Venting?)	Horizontal Yes (low point in system)	Horizontal Possibly not self-venting. Elevation same as bottom of condenser	Yes-thru mini-flow recirc line	horizontal (yes)	horizontal/yes low point in system)
Capacity	1000 gpm at 1065 psia	740 gpm each @ 3000 ft.	motor 840 GPM @2700ft. turbine 840 GPM @2650 ft.	1050 GPM @2500 ft. (250 GPM of this is recirc)	780 GPM @2600 ft.
Shutoff Head	1465 psia	motor: 3400 ft. steam: 3500 ft.	steam: 3050' @ 3560 rpm motor: 3100' @ 3560 rpm	≤3150 ft. @ 3600 rpm	
Suction Sources/Seismic Category	(1) Upper Surge Tank/ ASME VIII (3) Other Units' Surge Tanks/ ASME VIII (2) hotwell/no, Aux. SW pumps (3000 gpm @ 75 psig) (from emergency power-1 per site) Suction from CW intake located in Aux. Bldg. 1 floor below grade.	(1) CST/No. ASME Class 3, B31.1 (2) hotwell/No-These suction valves int'lked with vac. brkr. valve position (3) makeup from fossil units' demin./No (1 min)	Condensate storage tank - Seismic Cat. 1 Canal-Non-seismic (5 min.) Reservoir-Non-seismic	(1) CST/No (auto XFER to SW on low suction P-Class 1E, redundant instr.) (2) Deaerator/No (3) Fire Water System/No Last: SW pump discharges/Yes	(1) CST/No (2) SW pp disch./Yes Suction pressure switch (common-Non 1E) Remote Manual MOV's (requires only seconds to switch-Class 1E valves)
Turbine Driven Pumps Operable at What Range Of Steam Pressures	≥300 psig	≥200 psig	>213 psig (tested 1124 gpm at 213 psig)	≥50 psia (Psat for 280°F)	≥270 psig
Trips	(1) Overspeed (2) Low Hydraulic Pressure (shaft driven pump)	Overspeed/Motor trips on closed suction valve. Overcurrent	Manual (local or remote) Bus unloading Overcurrent: Inst 2000A, OST 4450 RPM, 960 for 5.15 sec, 640 for 6.43, 320 for 11.39	OST, Low Suction P, Low Steam Inlet P at >25 sec., Manual	Turbine-OST Motor-None
Instrumentation	EFW pp disch. P & Flow, SG level, SG Pressure	Drive turbine SV position Motor on-off lights Flow in SU FW line Ammeter	On-off lights for motor drive Ammeters Steam Supply valve position	Each pump: Discharge P Speed indication	Discharge P each pump

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Normal Lineup	Suction valves from surge TK open Disch. valves N.O. (check valves prevent back-flow)	All injection valves N.O. (check valve prevent back-flow)	FCV's & Bypasses N.C./ Cross-tie N.O. Suction from CST:N.O.	Suction Valves N.O. from CST Two series MOV's closed in each pump's discharge. One pump feeds one SG.	Discharge valves (MOV's-Class 1E) Closed. Cross-tie valves open.
Auto Initiation	Loss of both main FW pumps (detected by discharge header pressure <750 psig or FW pump turbine stop valve position) EFW does not start on ECCS.	Loss of both main FW pumps (as indicated by low control oil pressure) AFW does not start from ECCS initiation. Motor Driven pump-no auto start	Loss of both FW pumps (<850 psig on each pump disch.) These switches reset but pumps continue to run. (Single fail. proof) All RCP's off (Power monitor-current, volts, phase-same as RPS) Turbine only starts on ECCS initiation	Stm & FW Rupture Control System (see description under Auto FW Isolation) Does not start on SFAS.	Turbine: (1) SLBIC (see Auto FW Isol.) (2) Loss of FW sensed by governor latch on main FW pumps and "auxiliary" FW pump low disch.P (thru ICS) (3) loss of all RCP's (breaker position) Motor: No auto start. (No starts on ECCS)
Failure Mode on Loss of Air/Power	Loss of air switches 14" main header/ Valves & solenoids powered by batteries (non 1E)	FCV's lock in position Reservoir for 3 cycles/Emergency buses	Class 1E MOV Bypasses FCV on SFAS. FCV Fails Open/FCV Fails to 50%	No air-OP. Valves/MOV's fail as is - but all are powered by 1E	No air op. valves/ As-is (all valves are MOV's)
ICS Control Level: RCP/No RCP	25"/260" Sensed from breaker positions	30"/250"	30"/-318"	Not ICS. Auto essential level control system 120" from redundant, Class 1E instrumentation (Pump speed)	20" & 24"/-300" (50% OP. range)
Procedure/Practice	Same/Same	Same/Same	Same/Same	Licensee is installing dual level setpoint controller to relieve operator of this duty. 35" unless there is a loca/Same	Control RCS Temperature/Same

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Surveillance Test Method	Close manual AFW supply block valve. Recirc from/to upper surge tank. Valves do not realign automatically on SFAS.	Close discharge MOV's and recirc from/to CST thru, mini-flow lne. Valves do not realign automatically on SFAS.	Close FCV & x-tie from C.R. Pump from CST to cond. through test line. Valves do not realign automatically on SFAS.	From CST to drain thru normal recirc line (250 gpm). No valve realignment necessary.	Recirc to condenser or CST. Injection valves already closed. Operator opens the manual valve.
Steam Generator: Distance between tube sheets/AFW Inlet/Main Feedring	625"/590"/362"	625"/590"/382"	625"/603"/338"	625"/608"/388"	625"/590"/382"
Method to Protect Good S.G.	Operator action from control room.	Steam line failure matrix isolates all FW from SG if P<600 psig	Main steam line failure logic. (<435 psig isolates SG) Does not isolate AFW.	Stm & FW Rupture Control System (see description under Auto FW Isol.)	SIBIC (See Auto FW Isolation) Does not isolate EFW.

Compiled by Brent Clayton and
Art Oxfurth x27741

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and the primary-to-secondary heat transfer. The steam generator inventory as a function of time, and the time to steam generator dryout, depend on these rates and on the initial inventory.

2.2.1 Initial Inventory

We have had little discussion on whether it is practical to increase the time to steam generator dryout by increasing during normal operation the amount of fluid in the secondary side of the steam generator. As presently operated the collapsed water level at full power is quite low. The potential problems of increasing this inventory have not been discussed with NRC.

2.2.2 Scram

The scram decreases the heat input. Present B&W designs scram on primary system high pressure for LOFW transients. This typically occurs 8-10 seconds after LOFW. Alternatively, an anticipatory scram signal could be derived from one or more secondary system parameters (e.g., steam generator water level, turbine stop valve closure). This would initiate a scram ~6 seconds sooner than the present design, increasing the time to reach steam generator dryout by 1 minute or more. NRC Bulletin No. 79-05B requires B&W plants to provide for NRC approval a design review and schedule for implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, turbine trip, or significant reduction in steam generator level.

The realignment of primary pressure scram and relief-valve setpoints mandated in Bulletin 79-05B also have the effect of decreasing the scram delay and delaying dryout. The increment, whose value has not been calculated, is smaller than would be provided by the anticipatory scram. However, the setpoint changes have already been implemented on the plants whereas the anticipatory scram will be added in the future.

2.2.3 Time to Steam Generator Dryout

Table 2.1 gives the time to dryout of the steam generators of the operating B&W plants - about one-half minute at full power. Westinghouse steam generators have 2-3 times as much water in the secondary side of the steam generators, proportionately, as B&W plants; CE plants have 3-4 times as much as B&W plants.

However, these plants (W & CE) have anticipatory scram which extend the dryout times to many minutes.

After the scram, the heat input decreases rapidly and the water in the steam generator secondary boils off more slowly. Calculations for LOFW give B&W dryout times of 1-2 minutes for present B&W designs, depending on the course of the event. It is this fast dryout compared to other PWRs that makes B&W plants unique. The factor of 2-4 larger inventory and the anticipatory scram in non-B&W plants give calculated dryout times of many minutes. Thus the timing requirements for AFW delivery are substantially more stringent for B&W plants than for others. This increases the importance of timely manual initiation of AFW in B&W plants compared to the others. Moreover, there is less time to rectify operating or maintenance errors and get the AFW operational if it doesn't start initially.

2.2.4 AFW Delivery Rate

Table 2.1 shows the differences in AFW flow rate for the different plants. The actual flow will depend on the number of pumps running, the pressure in the steam generator against which they have to pump, and the action of control devices. These last are flow control valves in the AFW lines or throttle valves in the steam lines to the turbines on steam-driven pumps.

On all B&W operating plants but Davis-Besse, AFW flow is controlled automatically by values receiving a signal from the integrated control system. The controlled variable is water level, as shown in Table 2.1. A low level setpoint (2-3 feet above the tubesheet) is used when the reactor coolant recirculation pumps (RCP) are operating. This is switched automatically to a high level setpoint (21-26 feet) to enhance natural circulation when the RCP are not operating.

On Davis-Besse, a separate, safety grade, control system controls pump speed (via steam throttle valves) to maintain a level 10 feet above the tubesheet. For the "raised steam generator" configuration in this plant, the 10-feet level is sufficient to maintain natural circulation.

After a LOFW and scram, the steam-water mixture normally present in these once-through steam generators collapses to a liquid level typically 3 ft or lower. The level then decreases, and later increases as AFW comes on.

2.2.5 Long-Term Considerations - HPI

Recent operating data obtained informally from Oconee show the following:

	Unit 1	Unit 2	Unit 3
Automatic Initiation of HPI	1	1	2
Manual initiation of HPI	16	9	7

Thus HPI was initiated at a frequency of about two times per reactor-year. Not all of these initiations were for LOFW events, but some were. Manual initiations were said to have been accomplished in order to maintain pressurizer level. Evidently the primary system shrinkage after a successfully controlled transient involves HPI action.

This raises questions about the role and requirements for HPI. Rather than just being part of the ECCS, which was put in to control small breaks, it is used routinely for frequent anticipated transients. Its failure modes and the consequences of its failure should therefore be analyzed in that context in addition to reviews conducted in the LOCA context.

2.3 Reliability

Numerical criteria for AFW reliability do not exist, and estimates of the reliability actually achieved are also not available. The following discussions are therefore qualitative only.

2.3.1 Challenge Rate

Estimates by B&W and others give about two per reactor-year as the rate of LOFW events. B&W states that the rate, for all PWRs and for B&W plants, decreases to ~1.5 per reactor-year after an initial period of operation. We have no reason to doubt these values.

The HPI initiation rate reported in Section 2.2.5 above is also about 2 per reactor-year.

For a LOFW event, either AFW or HPI must function to protect the core. (There are some other alternatives, such as restoring main feedwater flow, but they do not significantly change the picture). The rate of accidents (full damage) would therefore be:

$$A(BC)$$

where A = challenge rate

B = failure probability of AFW

C = failure probability of HPI

Hence, "failure" means insufficient functioning to cool the core, and involves consideration of performance, timing, and reliability. Given A=2 per reactor-year, the product BC must be adequately low; numerical guidance is not currently available.

2.3.2 Source of Water

Table 2.1 shows the sources of water available to the AFW. Each plant has multiple sources, but in some older plants they are not seismic Category 1. Abundant quantities of water are available from these sources.

2.3.3 Pump redundancy

All plants except Oconee have redundant AFW pumps. All plants except Oconee and Davis-Besse have diverse prime movers - steam and electric.

Oconee has one steam-driven pump per unit. The three pumps for the three units can be interconnected through normally closed valves (remote manual control); two pumps are stated to be sufficient in capacity for all these units. The potential redundancy in this arrangement has not so far been exploited. Davis-Besse has two identical steam-driven AFW pumps.

2.3.4 Valves and Piping

Table 2.1 does not list the valve arrangement. In general, separate valves are provided to control AFW to the two steam generators. We have not yet evaluated whether a single failure - control, valve or pipe break - could inhibit all AFW; this was not a requirement when these old plants were licensed. In some plants, common pipes and relief valves exist whose failure could inhibit all AFW.

2.3.5 Controls

In all plants except Davis-Besse, the Integrated Control System actuates the AFW flow control valves. On some plants, these control valves can be bypassed (remote manual control) to allow AFW flow in the event of control system failure.

B&W was unable to state whether failures in the Integrated Control System could initiate a LOFW event and also inhibit AFW via the flow control values. We have asked B&W to analyze this question promptly. If this common-mode failure can occur, and we see no reason why it is impossible, then the combined frequency AB (see Section 2.3.1) could be high because, for these events, B = 1.

2.4 Conclusions regarding AFW

2.4.1 Performance

AFW performance in operating B&W plants appears marginal, in that dryout would occur rapidly (1-2 min) unless AFW is initiated at its design time of 40 seconds after a LOFW.

2.4.2 Reliability

AFW reliability in operating B&W plants varies widely among different designs. The older plants are not in conformance with SRP 10.4.9, for example, by requiring redundancy, diversity, and single failure criterion, etc. Improvements are needed in some plants.

2.4.3 Dependence on HPI

Successful recovery for most LOFW events appears to require HPI even if AFW functions as desired. This requirement to use HPI for an anticipated transient, and its failure modes and consequences of failure, should be analyzed in this context of use as inventory control.

3.0 TRANSIENT ANALYSIS

3.1 General

In general, the loss of feedwater transient analyses performed and reported in the Final Safety Analysis Reports for B&W reactors considered the event to be a loss of main feedwater only. A loss of all (i.e., main and auxiliary) feedwater has not been considered in the course of a usual case review. This is consistent with current and past practices because it was believed that a total loss of all feedwater could only occur after multiple and unlikely equipment failures. Operator error to lock-out a system had not been considered. Single failures were generally considered to be a loss of a redundant component to establish minimum system performance requirements.

An evaluation of a feedwater transient was performed for Three Mile Island Unit 2 as reported in the SAP and the results are typical for all B&W plants. However, feedwater transient analyses that take the lessons learned from TMI-2 have not yet been provided.

During a LOFW transient, the loss of main feedwater reduces the capability to dissipate heat-flow from the primary to secondary system. The primary system heats up, the power operated relief valve is actuated, and the reactor trips on overpressure in the primary system. [There are safety valves installed on the pressurizer to limit the pressure excursion to code design limits.] The emergency feedwater system refills the steam generator and dissipates the decay heat. The reactor core remains covered, no fuel damage occurs and calculated offsite radiological doses are well within the guidelines of 10 CFR 100. The actual analysis presented in the SAR spans a time period of about one minute. In this time, it indicates that core power and primary sytem pressure are moving in a safe direction relative to fuel damage and system overpressure.

- The SAR analysis that was performed did not include delay of AFW or failure of the power operated relief valve to reclose when the pressure decreased further. Further long term cooling aspects were not addressed. However, the Standard Review Plan (SRP 15.2.7) indicates that there should be no loss of function for any barrier other than the fuel cladding for such a feedwater transient, even when accompanied by a single failure.

The analyses of situations involving a release of reactor coolant from the system through a failure of a relief valve were based on small break ECCS studies and not as a consequence of an operational transient.

3.2 Small Break Analysis

The models that are used for small breaks analysis are usually Appendix K type with the emphasis on conservatism; e.g., loss-of-offsite power, minimum core cooling and no short term operator actions. More realistic studies of the reactor plant dynamic response are needed to ensure proper tracking and understanding of the event being analyzed.

The blowdown codes used by B&W are CRAFT and TRAP. CRAFT has been approved by the NRC for ECCS analysis of large and small breaks in the primary system. TRAP is a modified version of CRAFT with a detailed secondary model and a simplified primary model and is used for steam and feedwater line break analysis. TRAP is currently under review by the NRC.

The transient codes used by B&W are NATURAL, CADD and POWER TRAIN. CADD has been approved by the NRC for ATWS analysis. NATURAL, which would be used for natural circulation calculations, has not been submitted and POWER TRAIN is under review.

In response to staff requests, the Duke Power Company (Oconee Nuclear Station, Units 1, 2 and 3) provided (April 21, 1979) the results of an evaluation of small break events in conjunction with the loss of emergency feedwater flow for 20 minutes.

Operator actions are assumed to initiate HPI and restore emergency feedwater flow to the steam generators. The analyses indicate, in the licensee's opinion, acceptable results. The core uncovery is not predicted to occur and therefore adequate core cooling was available. The analyses covered various small break sizes of 0.07 ft^2; 0.02 ft^2 and 0.01 ft^2.

At a meeting held on April 24, 1979 the staff indicated its need for additional information for its review concerning the analyses; e.g., the ability of a HPI to provide adequate core cooling without short term operation of the AFW, break locations such as in the pressurizer should be considered; the analyses should extend into the long term cooling mode, and the systems effects of a stuck-open relief valve need to be discussed.

At this meeting the B&W representatives stated that further small break analyses had been performed that covered some of the staff's concerns. B&W agreed to provide the results of such analyses to the staff in two weeks. The analyses would include sensitivity studies on the delay of AFW, one and two HPI pumps in operation, and long term cooling capability.

Table 3.1, obtained from B&W, states those analyses done or a process that is relevant to transient analyses.

TABLE 3.1

CRAFT-II ANALYSES

		STATUS	RESULTS
1.	PORV stuck open; 2 HPI; RC pumps on + autofeed	Done	OK
2.	PORV stuck open; 1 HPI; RC pumps on + auxiliary feed	Done	OK
3.	PORV stuck open; 200 gpm; RC pumps on + auxiliary feed	Done	Melt
4.	TMI-II actual transient best estimate prediction	1/2 done we have it to one hour we will finish it to core uncovery	Melt
5.	.07, .02, + .01 Small breaks; no RC pumps, no auxiliary feedwater no 20 min.;.2 HPI	Done	OK
6.	Zero break with manual actuation of 2 HPI @ 20 min.; no RC pumps	Reconfirm old analysis	OK
7.	Small break in steam space of pressurizer 1.05 in^2. PORV break treated as normal small break; no RC pumps; auxiliary feedwater, 1 HPI	Done	OK
	Note: Additionally all analyses previously submitted in support of our FAC evaluation model. These make use of the three forms of natural circulation described.	Done	

CADDS SENSITIVITY STUDIES

		STATUS	RESULTS
1.	TMI-2 incident benchmark (~6 min.)	Done	
2.	Best Estimate Model Studies	Done	
	• AFW Actuation delay (40 sec.; 120 sec + delay)	Done	
	• Reactor trip coincident with LOFW/turbine trip	To do	
	• Studies supporting changes recommended in high RC pressure trip setpoint and PORV setpoint.	Done	

3.3 Natural Circulation Cooling in a B&W Plant

For most B&W plants, the safety analyses are carried out in time only long enough to indicate that pertinent parameters relative to core damage or overpressurization are proceeding in a safe direction. Analyses are seldom pursued out in time to evaluate operator actions, inactions, or error in judgment, or the course of natural circulation cooling in the event of a loss-of-offsite power. The concerns on natural circulation cooling have been raised by the ACRS and C. Michelson, a consultant to the ACRS.

A report entitled, "DECAY HEAT REMOVAL DURING A VERY SMALL BREAK LOCA FOR A B&W 205-FUEL-ASSEMBLY PWR," by C. Michelson (January 1978) has recently been provided to the staff. In this report Mr. Michelson described concerns regarding small breaks ($\sim .5$ ft^2 range) and the ability of the plant's heat removal systems to remove adequate decay heat to prevent system repressurization in the event of a loss-of-natural circulation or break isolation by operator action. He has also discussed concerns on slug or two-phase flow through a PORV. This report is presently being reviewed by the staff and B&W. The staff is pursuing with B&W and the owners of B&W plants those aspects of concern raised in this report.

Studies by B&W indicate that natural circulation should not be significantly affected due to the formation of steam spaces in the upper portions of the hot leg piping and upper plenum of the reactor vessel.

B&W has conducted tests to determine the amount of natural circulation. The tests are normally done during startup testing from an initial power level of about 20-25%. The reactor is scrammed, the RCPs are tripped, the emergncy diesel generator comes on, the steam and motor driven AFW pumps start, the ICS raises OTSG level to the 50% value, and the plant is verified to be operating on natural circulation, without any operator action.

These tests have been conducted at Davis-Besse and Oconee. Also, Arkansas-1 suffered a loss of offsite power from 100% on 7/25/75 and natural circulation was established, without any operator action. We were not provided with these data. TMI-2 also had two (2) unscheduled events in their startup testing program which resulted in natural circulation.

The staff requested as much detail and description as possible on all the natural circulation tests and events. B&W has agreed to provide the requested information to the staff including verification of its computer code to calculate natural circulation cooling. Such studies will include recent TMI-2 results.

While the staff believes that natural circulation cooling is effective, further evaluation of the B&W analyses and test information will be necessary to confirm the adequacy of this cooling mode.

4.0 ALTERNATIVES

We have briefly considered the pro-and-con of three alternatives related to the safety of continued operation of the B&W plants. They are listed below.

4.1 Further Bulletins

Pro

1. Bulletin process is simple for NRC, and has not proved a burden to industry (according to industry)
2. Temporary improvements can be implemented quickly.
3. We need more information of FMEA of ICS and plant transient behavior in order to make an informed decision; the bulletin is a fast and effective way to obtain information.

Con

1. Multiple bulletins on some subject poses potential for overloading operator.
2. Technical merits of revised designs not subject to usual thorough scrutiny of staff and applicant.
3. Needed information may take 1-2 months; delay in decision-making is not the most cautious thing to do.
4. Plant responses to bulletins are varied in substance.

4.2 Immediate Remedial Measures

Pro

1. Faster implementation of needed safety measures reduces the likelihood of another TMI in the interim.

Con

1. May not be enough time or adequate information for careful staff consideration.

4.3 Plant Shutdown

Pro

1. Conservative course of action.
2. Gives time for staff and industry to work in more orderly fashion.

Con

1. Difficult to enumerate the restart criteria.

September 14, 1978

Note to: Darl Hood, LPM, LWR-4, DPM

From: R. Martin, Technical Assistant for Reactor Safety, DSS

Subject: MIDLAND SCHEDULE AS OF 9/13/78

To be sure we are communicating accurately I would like to provide you the following understanding that I, RSB, and AB have of the Midland schedule attached to your Schedule Change Request dated 9/1/78 and your note to S. Varga dated 9/7/78.

1. RSB and AB currently have what essentially amounts to Q-2 packages in the concurrence chain. Both of these packages are expected to be signed out by about 9/15/78, your scheduled date for this milestone which you term Q-1 1/2.

2. We understand the applicant will be submitting responses on 10/20/78 and 12/11/78 to the still outstanding Q-1s and the 9/15/78 questions; then all three branches will be largely prepared to write their SER by your scheduled 3/1/79.

3. RSB is expected to have additional questions as a result of the ICSB review being done out of phase (later than) the RSB review. These questions will be provided at about the same time as I&CSB Q-2s, 3/1/79. In addition any other questions as a result of the three branches review of the future responses to Q-1s and Q-2s will be provided by the 3/1/79 date. Therefore you will get SERs from the three branches and most likely some additional questions by 3/1/79.

4. This get well plan of course assumes a high degree of success in resolving issues with responses received by 12/11/78.

With these clarffications we concur in the September11, 1978 schedule change request.

R. Martin

cc: D. Ross
RS B/Cs and S/Ls
M. McCoy
W. Jensen
S. Salah
C. Long
T. Sullivan

February 9, 1979 UNITED STATES SECY-79-106

NUCLEAR REGULATORY COMMISSION

INFORMATION REPORT

For: The Commissioners

From: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

Thru: Executive Director for Operations

SUBJECT: REVIEW OF USE OF WASH-1400 IN LICENSING ACTIONS

Purpose: To provide the Commission with the results of a further review of the extent to which licensing or other regulatory actions relied upon WASH-1400, including those issues raised by the Union of Concerned Scientists (UCS) in a press release dated January 26, 1979.

Discussion: On December 11, 1978, I forwarded the results of a preliminary survey of the use of the Reactor Safety Study (RSS) in the licensing process (SECY-78-637). The documents identified in that survey were categorized into six groups by the manner in which WASH-1400 was used. That initial survey revealed no general disposition toward primary reliance on the RSS, with only three items identified as warranting reconsideration. Since that time, I have critically reexamined those findings in light of the Policy Statement and the supplementary guidance in the Secretary's memorandum of January 18, 1979. Also, I met with individuals who commented on the first survey.

To better understand the context of the identified uses of the RSS, I and the Division Directors met with individuals responsible for those documents which appeared to involve at least partial reliance on the RSS in justifying either the status quo or a relaxation of requirements. Approximately 40 uses were reviewed in this effort (see Enclosure 2). The discussions focused on the role the RSS actually played in the analysis described, whether its use was the determinative factor in the decision, and whether a different conclusion would have been reached today. The results of this effort are summarized in the enclosed synopses, (Enclosure 1) which replaces and updates that in SECY-78-637. My views remain basically the same as stated in SECY-78-637. There is no pattern indicating a causal bond between the RSS and licensing decisions.

Contact:
D. F. Bunch, NRR
49-28041

Four additional instances have been identified where the staff's use of the RSS would not now be in conformance with the Commission's recent guidance. Reconsideration is not considered necessary for these because primary reliance was not placed upon WASH-1400. These were discussions of the health effects of alternative fuel cycles (1-2), the discussion of fire protection in safety evaluation reports (3-5), the treatment of accident risks in environmental impact statements (3-16), and the report on the RESAR-414 integrated protection system (5-3). Future actions on these matters will reflect the new guidelines.

Our reviews of the correspondence did reveal numerous references to WASH-1400 estimates of failure rates, principally the probability of a pipe break or a large loss of coolant accident. The written record generally does not contain a discussion of the uncertainties associated with those estimates. However, my discussions with the responsible individuals indicate that these uncertainties were considered by the staff and were an element of the decision making process, albeit an implicit one. I intend to provide specific guidelines to assure the explicit treatment of uncertainties in the future.

The review also identified several instances where the staff provided recommendations on matters decided by the Commission. These are:

- UCS Petition on Fire Protection (3-6)
- Containment Inerting Rule (3-43)
- PRM 50-19 on Vacuum Containments (3-64)
- Big Rock ECCS Exemptions (4-12)

Our review leads us to conclude there is no compelling reason to alter our previous recommendations on these matters.

The UCS, in its press release of January 26, 1979, appears to have a different view of the use of WASH-1400. As requested, we have given special attention to the UCS comments regarding fire protection, qualification of electrical equipment, and Class 9 accidents. We conclude that the record has been mischaracterized by the UCS and that the UCS recommendations to require the shutdown of a number of operating facilities are not warranted. Our views are provided in synopses 3-4, 3-5, 3-6 and 3-65 and an expanded discussion of these UCS issues is included as an Appendix to Enclosure 1.

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I continue to view the record as a whole as showing an ancillary use of the RSS in licensing actions. Its principal application has been to supplement or confirm the main stream of analyses and judgments reached by the staff. Past and present regulatory decisions depended on traditional engineering analyses rather than any assumed finality of the RSS. Another view by a staff member is given in Enclosure 3. In those three cases where primary reliance was placed on WASH-1400 or used WASH-1400 in a way that appears inconsistent with the Policy Statement, (1-1, 3-2, and 3-11) only one, D.C. power reliability, requires active reconsideration. I have found no actions which, because of their reliance on the RSS, should now be overturned.

Item number 3 of the guidance memorandum instructs the staff to review Commission, ACRS and licensing board actions and statements. The views of the ACRS are given in Enclosure 4. These actions will be the subject of a later paper.

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Synopses of Items Identified in Staff Survey
2. Items Discussed with Staff
3. Memorandum from D. L. Basdekas to H. R. Denton dated February 5, 1979
4. ACRS Views on Application of WASH-1400

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ENCLOSURE I

SYNOPSES OF

ITEMS IDENTIFIED

IN STAFF SURVEY*

*A cross-index of the items identified in this Enclosure to those identified in SECY-78-637 follows Category 5.

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Synopses of Category 1 Items

Definition

Includes those actions in which an absolute value of accident risk as set forth in WASH-1400 was relied upon in the licensing process to make a specific licensing decision. Included in this category would be any reliance on an overall probability for core melting or on the probability of a given event sequence leading to core melt. A possible example is the use of the RSS to develop quantitative estimates of health risk from the coal and nuclear fuel cycles.

Summary

The two items identified in this category include the example in the definition and the use of the numerical estimates of core melt probability from WASH-1400 to derive proposed safety objectives for ATWS. In both of these instances, either the final report or the planned supplement have or will include use of WASH-1400 in a manner consistent with the Review Group recommendations.

1. Synopsis: Using the results of WASH-1400, regarding the probability of core melt, the staff recommended in NUREG-0460, that the safety objective for ATWS events be changed from 10^{-7}/RY to 10^{-6}/RY. The staff further recommended that systems to be used to mitigate ATWS events be safety grade or that they could be shown to be reliable using RSS estimates or an updated data base. Other portions of the ATWS study where WASH-1400 is addressed fall into Categories 2, 3 and 4.

   In deliberations before the Regulatory Requirements Review Committee the propriety of basing the decision heavily on the RSS was questioned. It was recommended that these actions be reconsidered. Volume 3 to NUREG-0460 takes an approach which is intended to be consistent with the Review Group's recommendations.

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2. Synopsis: Health Effects Attributable to Coal and Nuclear Fuel Cycle Alternatives, Draft NUREG-0332, includes references to WASH-1400 data. Somatic health effects have been considered in numerous forums including hearings and impact statements. Although the format of the documents involved has varied slightly, the method of incorporating WASH-1400 has been the same as in NUREG-0332 (draft). The draft report recognized the controversy over the probabilities of serious nuclear accidents and discusses a 20-fold increase in WASH-1400 accident risks and its affect on the health effects assessment of the fuel cycle. As a result of the Commission policy statement all future environmental impact statements that reference NUREG-0332 draft will, in its discussion of health effects, describe: the Lewis Report conclusion regarding uncertainty in the RSS, note publication of the policy statement, and discuss the implications of a core-melt probability of up to a factor of 100 greater than that in the RSS. (The first such analysis has been performed and the staff conclusion has remained unchanged.) No reconsideration of previous licensing actions appears necessary. The final version of NUREG-0332 should include a range of mortality values for the uranium fuel cycle that includes a consideration of a broader range of accident risk estimates consistant with the Review Group's recommendation and the Commission's Policy Statement.

Synopses of Category 2 Items

Definition

Includes those actions in which the absolute values of accident risks of WASH-1400 were used in the licensing process, but when such use was restricted to relative comparisons of risks.

Included in this category would be any reliance on the overall probability of core melting of the RSS to draw comparisons between two design concepts. Possible examples are the use of the RSS to compare an FNP to a land-based plant and the use of the RSS to develop perspectives on overall ATWS risks.

Summary

There were 4 items in this category. Typically, items in this category utilize the numerical risk estimates of the RSS (such as a core melt probability of 5×10^{-5} pe eactor year) but only in a relative sense.

These assessments did not require that the values used be precise since they were used to compare the relative differences between two or more alternatives or concepts. In the staff's view, none of the items in this category warrant reconsideration.

1. Synopsis: The Safety Evaluation Report for Offshore Power Systems Floating Nuclear Power Plants 1 through 8, NUREG-0054, issued October 8, 1976, contained in its Appendix C a rationale for the selection of quantitative risk criteria. WASH-1400 was cited as a source of information on low probability events but no reliance appears to have been placed on the data. The SER acknowledged, in a qualitative sense, that there were a range of views as to the uncertainties associated with accident risk estimates. The Commission's Policy Statement would now require a more explicit indication of the wide range of uncertainty associated with accident risk estimates.

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2. Synopsis: The "Estimation of Safeguard-Related Risk Associated with Continued Operation of Existing SNM Processing Facilities" by J. H. Conran in late 1976 and other related earlier documents, compared a safeguards-related risk to safety-related risk (as given in WASH-1400), in an attempt to show that NRC safeguards approach should be more conservative. Safeguards requirements for both reactors and safeguards facilities have been upgraded since 1976, however, it is unclear the extent the aforementioned analyses were used, if at all, in developing the upgraded requirements.

3. Synopsis: Liquid Pathway Generic Study, NUREG-0440, February 1978 and Offshore Power Systems, OES, Part III, NUREG-0127 (Revision 1) uses a variety of results from WASH-1400 and follow-on studies to compare risks of a floating nuclear plant to a similar land-based plant. The final impact statement was issued after the Lewis Report was issued and includes several comments on the great uncertainties associated with the RSS core melt or risk estimates. (See also Category 3 - item 36).

4. Synopsis: Commissioner Action Paper, SECY 78-137, March 7, 1978, Assessments of Relative Differences in Class 9 Accident Risks, provides an evaluation of alternatives to sites with high population densities. WASH-1400 consequence models were modified to provide site specific information and then were used to perform analyses of the differences between

the Perryman site and other alternative sites from the standpoint of accident risks. The paper addresses the limitations of the RSS in such applications and contains a number of cautionary remarks about the uncertainty of the estimates derived from the RSS consequence model. The results were used to gain additional perspective about the characteristics of the sites and thus supplement other parts of the site review. The recently issued Policy Statement provides specific guidance on use of the consequence model. The use in this instance is believed to be consistent with the Policy Statement.

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Synopses of Category 3 Items

Definition

Includes those actions in which the quantitative estimates of fault tree/event tree analyses of WASH-1400 were used in the licensing process to illustrate or confirm staff conclusions on the disposition of a potential safety issue or to aid in selecting the preferred of several alternate regulatory requirements. One possible example is the NUREG-0138, "Treatment of Non-Safety Grade Equipment in Postulated Steam Line Break Evaluations."

Summary

There were 66 identified topics in this category. The predominant use of WASH-1400 was to further support or buttress a staff conclusion and not as the principal basis for the staff action. For example some of the items contained in NUREG-0138 and NUREG-0153 utilized information from WASH-1400 to help respond to the concerns raised by some individuals that the priority or progress of resolution of certain issues was not proceeding satisfactorily.

Since the values or techniques were only used in a supportive role or to help select a preferred of several alternatives, all but two do not require any reconsideration. One of those two, CRBRP design criteria will be reconsidered if the review is reactivated. The other (reliability of d.c. power supplies) is being reconsidered as a part of generic issue A-30.

During the course of the review of these items, a few were identified that appeared to be significant reliance on WASH-1400, but where it was evident that the action taken would not have changed even if the values used had changed substantially (e.g. denial of PRM 50-19, item 3-64).

1. Synopsis: Testimony presented at the Beaver Valley, Unit No. 1 hearing used a figure of 1×10^{-4} per reactor-year as the base value for probability of pipe rupture. A table on p. 15 of the testimony provides ranges of failure rates from various sources including WASH-1400. WASH-1400 estimates were similar to those from other sources and no reconsideration is necessary.

2. Synopsis: In the CRBRP FES (NUREG-0139, Section 7.1.2) the staff compared a number of selected CRBRP accident sequences with the results of similar sequences analyzed in WASH-1400 in order to provide an additional basis for gaining perspective on risks of very severe accidents in CRBRP.

3. Synopsis: Certain Westinghouse Topical Reports rely upon absolute values of probability of accident events as set forth in WASH-1400. These reports currently are under staff review. Certain of these reports (WCAP-8966, WCAP-8976 and WCAP-9283) are referenced in RESAR-414, and the remainder are expected to be referenced in other applications. The recent NRC policy statement and supplementary guidance to the staff will be implemented in reaching a final staff position on these topicals.

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4. Synopsis: In discussing the interpretation of General Design Criterion 19, we noted in NUREG-0138 that the analysis of the Browns Ferry fire in the WASH-1400 supports the staff position, whose primary basis did not rely on WASH-1400. For an event in the control room to lead to serious consequences it would need to involve damage of redundant equipment in the control room (or anywhere else) in such a way that operations at the secondary control stations could not accomplish long-term cooling of the reactor. The fire damage experience at Browns Ferry involving (among other things) the loss of control of a number of systems helps to verify that many redundant means are available

to resourceful reactor operators to maintain a reactor in safe condition. The staff concluded that a serious accident resulting from damage to the control room is of sufficiently low probability as not to warrant revision of the current design basis. (Fire issues are discussed in greater detail in an Appendix to this Enclosure.)

5. Synopsis: The Conclusion section of some Fire Protection Safety Evaluation Reports such as Amendment 60 to the Hatch Unit No. 1 operating license contains a quote from the Review Group report on the fire at Browns Ferry (NUREG-0050). The quote is in part, "the study (WASH-1400) concludes that the potential for a significant release of radioactivity from such a fire about 20% of the calculated from all other causes analyzed." The Review Group conclusions has been part of the staff's bases for allowing continued operation of the facilities until implementation of facility modifications.

This statement has been used only to support the staff's overall technical judgment. However, an additional paragraph is being added to the SERs to further clarify the staff's bases for allowing continued operation. (This item is discussed in greater detail in an Appendix to this Enclosure.)

6. Synopsis: In their responses of December 15, 1977 and July 6, 1978, to the Commission on the UCS petition for emergency and remedial action, the staff utilized the work of the Browns Ferry Review

Group as reported in NUREG-0050. This group utilized the models of WASH-1400 to provide additional support to its conclusion. (This issue is discussed in greater detail in an Appendix to this Enclosure.)

7. Synopsis: An analysis of the type done in WASH-1400 was used as a partial basis for recommending thatonly manual portions of fire protection capability in new plants need have seismic qualifications and operating plants or plants under construction need not have any seismic qualifications for seismic events. This item was initially included in Category 5; however we have now determined some WASH-1400 data were also used in conjunction with this analysis. Thus, it is included herein. The primary concern of the Lewis Committee, the possible inaccuracy of probabilistic estimates, was considered when this issue was decided. The estimate was charactertized as "rough" and recognized to be based o. a limited amount of data concerning the frequency and consequences of earthquakes. We accepted the results of the study as valid because the conservative assumptions used in making the estimates are adequate to compensate for the limited data. One conservatism is the assumed value of the probability that a fire would occur given that an earthquake occurred and caused damage. Data from the 1971 San Fernando earthquake, which was not used in the estimate, indicate that the probability of a fire resulting from an earthquake is lower than used in the estimate. A second conservatism, is the assumption

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that the probability of failure of a non-Seismic Category I automatic fire protection system, given an earthquake occurs, is one A third conservatism is that the estimate did not consider fire suppression by manual means which is available at each site. A fourth conservatism is that the potential for fires in safety related areas of nuclear power plants is lower than in other structures because the nuclear power plant structures, even if non-Seismic Category I, are generally of massive reinforced concrete that are not likely to be damaged by an earthquake. Finally, in a nuclear power plant as opposed to other industrial facilities, combustibles and ignition sources that could cause fires are limited and not located in safety related areas.

8. Synopsis: The staff practice of not requiring that a passive mechanical valve failure be considered as a single failure following a postulated design basis accident is based on our judgment that such failures have an acceptably low likelihood of occurrence during both the injection (short-term) and recirculation (long-term) phases of a loss-of-coolant accident. Further, analyses of ECCS performance in WASH-1400 indicate that passive mechanical failures of valves are unimportant contributors to ECCS unavailability during both the injection and recirculation modes of operation. Thus the staff does not consider that changes in safety criteria are warranted at this time but studies will seek to compile a more rigorous data base on passive valve failures. These

studies (TAP's B-4 and B-58) are not scheduled for resolution in the near-term, due to the allocation of resources to higher priority tasks.

9. Synopsis: An Information Report on the Sin : Failure Criterion (SECY-77-439) was sent to the Co... ssioners on August 17, 1977. This report describes current practice on application of the single failure criterion to LWR electrical and fluid systems. It draws upon WASH-1400, in part, to support the conclusion that the single failure criterion, as it is currently applied, leads to a generally acceptable level of hardware redundancy in most systems important to safety. It also points out that methods such as those used in WASH-1400 will gradually come into increasing use as a supplement to the Single Failure Criterion.

10. Synopsis: In considering loss of offsite power subsequent to normal safety injection reset following a LOCA, the staff implied in NUREG-0138 that the analyses in the Reactor Safe Study, WASH-1400, were relied upon in the interim period while actions were taken in operating facilities to resolve the concern. The likelihood of a LOCA was estimated to be about one chance in 1000, per reactor year. This estimate is supported by experience and other studies independent of the RSS. The probability of a LOCA was combined with the probability of the loss of offsite power in a one-hour period following a LOCA (about one chance in 50,000 from information in WASH-1400 and from grid availability data) to obtain a combined probability of this sequence of events which was very low.

On the basis of our review of this issue as redefined in NUREG-to review the emergency diesel loading for operating PWR's to assure that all safe shutdown loads (which include cooling to the diesel generator) are automatically picked up following an operator action to reset SIS. I&E inspectors also were to examine emergency procedures to be followed in the event of a LOCA to assure that these procedures do not permit SIS reset by operator action earlier than 10 minutes following the accident signal, unless it can be shown that such action is required in the interests of safety. These actions have either been completed or are being addressed as manpower becomes available. It is the staff's judgment that operating facilities are safe and no reconsideration is needed during the time needed to complete staff efforts.

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11. Synopsis: On July 15, 1977, it was stated by staff to the ACRS regarding DC power reliability that, "...a conservative probabilistic assessment of the likelihood of occurrence of Mr. Epler's postulated scenario which is the basis for the concern regarding DC system reliability has been performed."

"The probability for occurrence of unacceptable consequences, i.e., core melt, as a result of this postualted sequence is 5×10^{-9}. A comparison with the WASH-1400 core melt prediction of 5×10^{-5} indicates that the contribution to core melt of this particular sequence is a fraction of one percent. Furthermore, this would not change significantly even if it were assumed that there would not be any capability for manual action to restore core cooling; i.e., if this number were one instead of 5×10^{-1}."

"A similar conservative assessment has been made for the postulated sequence initiated by simultaneous loss of both redundant DC divisions and predicts a core melt probability of $<5 \times 10^{-7}$. Comparison with the WASH-1400 prediction again shows that the contribution to core melt of the common mode sequence is negligible."

"In the staff's judgment, on the basis of the probabilistic assessments cited, core melt resulting from the simultaneous and independent failure of two redundant DC power divisions is so unlikely as to be incredible; and core melt resulting from common mode failure of these systems is very low in likelihood. We conclude, therefore, that adequate protection of the public health presently exists. However, additional technical studies over the next year should and will be performed to add confidence to this judgment." This issue should be reconsidered in association with the completion of Task Action Plan A-30 .

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12. Synopsis: In evaluating a postulated steam line break inside containment, the staff has accepted operation of certain "non-safety grade" isolation values (steam line and feedwater line) needed to limit both the resultant blowdown to a single steam

generator and the consequences of the postulated accident. Credit is taken for the closing of these "non-safety grade" valves only if a single failure is postulated in a redundant safety grade isolation valve in one of these lines.

In discussing this position in NUREG-0138, the staff considered the reliability of these valves and the probability of certain accident scenarios. The staff concluded that it is acceptable to rely on these "non-safety grade" components in the steam and feedwater systems because their design and performance are compatible with the accident conditions for which they are called upon to function. This justification does not rely on any probability arguments and therefore the position does not depend upon WASH-1400.

13. Synopsis: In a document transmitted to the ACRS in February 22, 1977, regarding grid availability, the staff stated:

"The data base used in the analysis is that provided in WASH-1400. The symbology, WASH-1400 numbers with specific references, sample calculations and tabulated results are attached. The conclusions reached is that the improvement in unreliability of offsite power the emergency buses provided by a second immediate access circuit is not significant. This is true even if the unreliability of the grid, which is the governing factor, were reduced by a factor of 10." Thus WASH-1400 was used as a source of information on grid reliability, but great accuracy in the estimates of WASH-1400 was not essential to the staff conclusion.

14. Synopsis: The Supplement No. 2 to the Staff Safety Evaluation Report on the OPS case states regarding accident evaluations that WASH-1400 results confirm that accident risks are roughly proportional to population density. Thus WASH-1400 was used as an additional source of information to support other analyses by the staff.

15. Synopsis: In the development of a draft paper on Current Accident Evaluation Practices, dated October 3, 1977, it was proposed as an interim position that no change in the safety or environmental regulations pertaining to nuclear power plants is warranted until a detailed evaluation is made of the draft study. WASH-1400 statements were used in a confirmatory manner.

16. Synopsis: Section 7.1 of staff Environmental Impact Statements on CP applications include a discussion of accident risks as set forth in the proposed Annex A to 10 CFR Part 50 Appendix D. This discussion routinely includes two paragraphs relating to WASH-1400 . One notes the existence of the study and the exchange of correspondence with EPA on the study. The second paragraph states, "As with all new information that might have an effect on the health and safety of the public, the results of these studies will be assessed within the Regulatory process on generic or specific bases as may be warranted."

While these EIS's do not rely on WASH-1400, future statements will be modified to reflect the NRC Policy Statement on its use.

17. Synopsis: The Final Environmental Impact Statements on LWR CP and OL applications include a response to comments received on the draft. Routinely comments are received by EPA, DOI and interested members of the public dealing with accident risks and/or WASH-1400. Frequently the responses make note of results of WASH-1400.

18. Synopsis: In the Three Mile Island, Unit No. 2 hearing, staff witness responses to cross examination in transcript regarding aircraft crash hazards made various references to WASH-1400 during testimony. However, reliance was not made on WASH-1400 in connection with the staff analyses of aircraft risks.

19. Synopsis: In a memo dated October 31, 1978 regarding San Onofre Unit No. 1 the probability of a propane explosion was discussed to determine if it should receive greater priority than other tasks in the SEP. The event was determined to be of low probability; WASH-1400 estimates were used as a point of comparison to illustrate that propane explosions could be considered on the same time scale as other SEP items.

20. Synopsis: Testimony of C. Vernon Hodge and Donald J. Kasum related to radioactivity released as a result of sabotage during shipment of radioactive material (Starling and Pilgrim

hearings) indicated that no credit is given for protection afforded by buildings or for evacuation of the endangered area. WASH-1400 is referenced to indicate that there actually would be a range of mitigating factors.

21. Synopsis: Page A-2/6 of Revision 1 to Task Action Plan A-2, Asymmetric Blowdown Loads on Reactor Primary Coolant System, cites pipe failure probability estimates from WASH-1400. This information was used to support the staff's engineering judgment for continued operation of the affected plants. Since the information was used only to support the staff's engineering judgment, NRR believes no reconsideration is necessary. This issue will be studied further as part of TAP A-2 which has been designated an "unresolved safety issue (USI)." Resources will be expended on this task on a priority basis.

22. Synopsis: The Safety Evaluation Reports on steam generator operation for Surry Unit No. 1 dated February 8, 1977, Turkey Point, Unit No. 4 dated February 8, 1977, and Surry, Unit No. 2 dated April 1, 1977 used pipe failure probability estimates from WASH-1400. This information was used to support the staff's engineering judgment for continued short-term operation. These three reactors which were experiencing steam generator tube failures were granted continued operation for 60 days.

In addition TAP's A-3, A-4 and A-5 have been designated as "USI's." Resources will be expended on these tasks on a priority basis.

23. Synopsis: For the extension of the ECCS exemption for Dresden, Unit No. 1, the staff constructed simplified fault trees of selected ECCS equipment and derived numerical probability estimates using failure rates from WASH-1400. In an October 28, 1977 note to I. Wall, Mr. Taylor sent the results of some probabilistic assessments pertaining to an ECCS single failure exemption for Dresden 1. This was done in response to a request from DOR. The results of the probability logic were not used in the December 29, 1977 SER. The information was used to support the staff's engineering judgment. The exemption from 10 CFR 50.46 was extended from December 31, 1977 to October 31, 1978.

24. Synopsis: In the May 9, 1978 RSB input to the Safety Evaluation for the Haddam Neck Overpressure Protection System, the staff tentatively accepted the results of a quantitative fault-tree analysis. The fault-trees were constructed to determine scenarios that could lead to an overpressurization. Possible human errors were considered to be the principal causes of such events and human error rates were taken from WASH-1400 (Appendix III, Table II-3). This analysis was used as a portion of the supporting basis for omitting inadvertant

water injection into the primary system through the high pressure safety injection pump as a design basis event.

25. Synopsis: The February 13, 1976 Safety Evaluation for Vermont Yankee authorized continued operation for 30 days until hold-down devices were installed on the torus. The licensee presented as supporting information pipe failure probabilities from WASH-1400. The staff, with more conservative failure estimates, effectively endorsed probability values as supporting information to the staff judgment in granting continued operation. Other factors affecting staff judgment were the ΔP mode of operation, recent inservice inspections of affecting piping, and short period of time (30 days).

26. Synopsis: RSB's October 18, 1977 Safety Evaluation granted a one cycle exemption from the Appendix K single failure criteria applied to the Big Rock Point Nozzle Spray System (NSS). The exemption request was made since the licensee could not substantiate the ability of the Ring Spray System alone to provide adequate core cooling in light of recent test data. The staff evaluated the probability of a non-refloodable LOCA and the failure of the NSS, and the probability of a LOCA in the NSS (refloodable LOCA) and the failure of the feedwater system using the WASH-1400 fault tree techniques. The staff's recommendation that the one cycle exemption be granted was not

based on these probability assessments alone. Several other factors related to the BRP ECCS performance and reliability were considered by the staff, and our conclusions reflect an integrated assessment. (See also Category 4, item 12)

27. Synopsis: The April 1, 1977 Safety Evaluation granted a six month exemption from the ECCS single failure criteria to San Onofre. Component failure rate data from WASH-1400 were used as a portion of the supporting bases for granting the exemption.

28. Synopsis: Pages A-12/3,4 of Revision 1 to Task Action Plan A-12, Fracture Toughness and Potential for Lamellar Tearing of Steam Generator and Reactor Coolant Pump Supports, cites pipe failure probability estimates from WASH-1400. This information was used to support the staff's engineering judgment for continued plant operation. This issue will be further studied as part of TAP A-12 which has been designated an "USI." Resources on this task will be expended on a priority basis.

29. Synopsis: To achieve a level of safety for CRBRP comparable to that for LWRs as far as residual risks associated with core melt accidents, the staff utilized WASH-1400 analyses of the times to containment failure to aid in establishing CRBRP containment integrity requirements. If the CRBRP review is

reactivated, this decision should be reevaluated. In light of the current inactive status of the project, no further action on reconsideration is recommended at this time.

30. Synopsis: Reference to WASH-1400 was made by the licensee in providing the justification for not removing the catwalks from the Nine Mile Point, Unit No. 1 containment torus for a period of five months. To the best of our recollection, the licensee's position was accepted as the basis for continued operation. However, the catwalks have since been removed.

31. Synopsis: WASH-1400 is occasionally used to support reviews of events considered for reporting as abnormal occurrence. For example Davis Besse Unit 1 diesel generator automatic load sequence failure was reviewed by the licensee using probabilistic techniques.

32. Synopsis: In periodic updating of the IE reactor inspection procedures, a cross-check has been made to determine that WASH-1400 high risk event related procedures and equipment receive appropriate inspection attention. Although the specific values stated in WASH-1400 were used in this evaluation, they were used to make subjective comparisons and to confirm previous conclusions.

33. Synopsis: IE is studying ways of using risk analysis to improve the inspection program to make resource allocations and to categorize risk related procedures with emphasis on human factors.

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34. Synopsis: Some accident sequences taken from WASH-1400 were made the basis for scenarios in developing procedures for the Incident Response Center.

35. Synopsis: While none of the results or models of WASH-1400 were used in licensing reviews, the consequence model computer code (CRAC) has been used by NMSS in NUREG-0194, a special study of transportation sabotage, and some data from WASH-1400 has been used in generic environmental statements on transportation of radioactive materials (NUREG-0170 and SAND 77-1927). However, no new regulatory actions or changes to rules have resulted from these efforts. Thus, no regulatory actions or staff positions have been affected by WASH-1400 material.

36. Synopsis: In a memo from S. Levine to B. Rusche dated August 9, 1976, WASH-1400 results were used to support comments on the draft liquid pathway generic study (See also Category 2, item 3).

37. Synopsis: Studies were performed by Batelle Columbus (BMI-2002) and Sandia (SAND 77-1344) for RES on alternate containment concepts. These studies used the methodology of WASH-1400 to determine the potential risk reduction from various containment designs. This information has been considered in establishing the priorities for research on improved safety

concepts, as reported in NUREG-0438, and were not used as the basis for a licensing action on LWR applications.

38. Synopsis: In an October 14, 1977 memo, I. Wall sent J. Stolz comments on PAB's review of Diablo Canyon Amendment 52. The analyses in support of the Amendment and therefore these comments refer to component failure probabilities, and consequence models and results from WASH-1400. A December 30, 1977 memo from Wall to Stolz provides a draft SER input supporting Amendment 52. Memos from Buhl to Stolz dated September 8, 1978, and November 6, 1978, provide a reassessment of the Diablo Canyon analysis of the risk to the public from a seismic event in light of the comments of the Lewis Committee. Methodology and absolute values of risk from WASH-1400 were compared to the applicant's recommendations. These assessments were to support a request for interim licensing. This request for interim license has been withdrawn and the risk assessment approach abandoned.

39. Synopsis: Task Action Plan A-37, "Turbine Missiles" (Revision 1) in Section 3, "Basis for Continued Plant Operation and Licensing Pending Completion of Task," states:

"The basis for allowing continued operation of the existing LWRs, pending completion of this task is the low probability of unacceptable damage to an essential system by turbine missiles. WASH-1400 assessed the turbine missiles accident risk and concluded that LWR designs have a considerable degree of protection provided by plant design and layout such that the public risk associated with large turbine missiles is insigifnicance compared to risks from other accident causes."

Two memos that preceded this version of the Plan (October 14, 1977 from I. Wall to S. Pawlicki and M. Taylor to S. Pawlicki dated September 3, 1976) provide comments on turbine missiles based on WASH-1400 analyses.

As reflected in the balance of the Plan, the staff has not relied upon WASH-1400 in its case-by-case reviews of turbine missile risks. The reference to WASH-1400 is corroborative or supportive and is not the principal basis for licensing. The information in WASH-1400 on turbine missile risks is largely taken from other sources (e.g., an analysis of turbine failures by Bush). The degree of reliance on WASH-1400 was questioned by the UCS in connection with the North Anna Review. In the NRC's Staff Response to UCS Brief Amicus Curiae dated November 15, 1978 at page 4, the staff notes that "WASH-1400 does not form a basis regarding turbine missiles contained in the Staff's Response and in §10.7 of Supplement 2 of the Safety Evaluation Report for North Anna, Units Nos. 1 and 2.

40. Synopsis: In an August 3, 1977, memo I. Wall sent J. Knight comments on Task A-18, Pipe Rupture Design Criteria. The comments were based in part on the results of WASH-1400. It noted that "The Reactor Safety Study (WASH-1400) assessed much of the available pipe rupture data and found that it was necessary to carry a large uncertainty (factor of 100) on the pipe rupture probabilities."

41. Synopsis: In response to an ACRS request, the June 20, 1977 and August 11, 1977 memo from S. Levine to R. Fraley transmitted calculations performed by PAB (with assistance from NRR:AAB) on Control Room Doses for Postulated Core Meltdown Accidents. The doses were calculated for two accidents as characterized in WASH-1400.

42. Synopsis: The March 28, 1977 memo from Mat Taylor to Ian Wall transmitted viewgraphs on three generic issues (PWR Pump Overspeed during LOCA, Installation of Seismic Scram, and Turbine Missiles) which were to be used in an informal presentation by RES:PAB to NRR. The viewgraphs used results and insights from WASH-1400 and made no conclusions and recommendations for actions on each issue.

43. Synopsis: In a memo from I. B. Wall to V. M. Panciera dated July 9, 1976, regarding the estimated impact upon public risk associated with a non-inerted BWR containment, it was noted that, based on a scoping risk assessment, non-inerted BWR Mark I containment risks associated with a LOCA would be somewhat greater than for an inerted containment but "would not significantly change the magnitude of overall public risk estimated in the Reactor Safety Study for the BWR." This letter was noted and considered in connection with a proposed rule on inerting (SECY-78-290).

44. Synopsis: Battelle -Columbus prepared a report on the effect of engineered safety features on LMFBR risk due to accidents. WASH-1400 accident event trees were used in the analyses. This information was not relied upon in any licensing reviews.

45. Synopsis: In a memo (Buhl to Mattson) dated September 21, 1978 provided (RES) comments on Supplement 1 to NUREG-0460. Methodology and insights from WASH-1400 were used in the recommendation to NRR. (See Category 1, item 1.)

46. Synopsis: The March 21, 1977 memo from W. Vesely to R. Baer, C. Berlinger, S. Israel, and J. McGough transmitted information copies of a description of the allowed downtime calculational approach used by PAB. Accident probabilities are used in the calculations.

47. Synopsis: The February 25, 1977 memo from S. Levine to B. Rusche and R. Minogue transmitted Research Information Letter No. 10,

sure Vessel Failure Probability Prediction. The draft report compared the new failure probabilities with those predicted in WASH-1400. The report was only a draft and no licensing action was taken based upon it.

48. Synopsis: Two memos [I. B. Wall to File, dated April 5, 1976, Subject: Minutes of Meeting Held on April 2, 1976, and I. B. Wall and W. E. Vesely to H. J. C. Kouts, dated March 16, 1976, Subject: Comments on "Reliability Assessment of CRBRP Reactor Shutdown Systems" (WARD-D-0118 Rev. 1), November 1975] discussed the role of probabilistic analysis in the licensing of the Clinch River Breeder Reactor Plant. The discussion in the memoranda relied on WASH-1400 insights, data and analyses of similar LWR systems to assess the feasibility of the CRBRP Control System to meet the numerical goals set for it by the applicant. Based partially on the Wall/Vesely evaluations the staff decided not to accept the applicant's position of using numerical techniques and criteria in determining the reliability of the system, but instead utilize the deterministic approach in conjunction with the utilization of insights gained from WASH-1400 during the design, bench testing and preoperation testing. (See Category 4, item 4 for additional discussion.)

49. Synopsis: Memo from Edison to Novak dated November 7, 1978 provided an assessment, using WASH-1400 techniques, of changing the test frequency of the containment spray recirculation pumps. This assessment was used by the staff in its consideration on alternate testing scheme for the Surry pumps. Results indicated that less frequent dry starting improves pump availability. Thus, less frequent testing should result in improved safety.

50. Synopsis: In September 1976, the Director, RES testified in a court proceeding related to the constitutionality of the Price-Anderson Act. His testimony covered what WASH-1400 was and its results. RES considered this as a Category 1 item. Since absolute values of risk were not relied upon to make any specific licensing decision in this instance, NRR has classified it as a category 3. It should be noted that the court ruled against the NRC in this instance. On May 17, 1977, a memo from I. Wall to S. Eilperen transmitted comments on Judge McMillan's decision. These comments were developed using WASH-1400 methodology and results. However, the lower court decision was overruled by the Supreme Court. Further, as we understand it, the Supreme Court decision did not depend on the numerical risk estimates of WASH-1400.

51. Synopsis: Memo from Buhl to Mattson dated May 18, 1978 comments on proposed NRR study of missile impact effects on structural barriers (TAP A-32). Memo compares proposed study with an attached event tree and concludes proposed study only covers a small part of total accident sequence probability. Memo uses WASH-1400 analyses to confirm RES conclusion on utility of NRR study. The comments were directed to assuring that the scope of the proposed study was broad enough to encompass the significant contributors to risk.

52. Synopsis: As part of staff efforts regarding Seismic Scram, UCRL performed a study (UCRL-52156, "Advisability of Seismic Scram") which relied upon some WASH-1400 data regarding accident probabilities as a means of evaluating relative accident probability with and without seismic scram. The staff has informed the ACRS that it considers this item to be of low priority. The staff has, as yet, taken no final action regarding this matter and will consider the latest policy guidance before action is completed.

Memo from I. Wall to V. Panciera, dated April 15, 1976, provides comments on the second status report of the UCRL study. These comments were based on WASH-1400 insights and results.

53. Synopsis: The proposed revision to SRP 5.4.7 (March 9, 1976 memo to RRRC) argued for increased requirements for RHR systems based on insights from WASH-1400. It was stated that WASH-1400 "shows for PWRs, that the inability to remove decay heat from the reactor following a normal shutdown has a higher probability of resulting in a core melt than does a large LOCA...for BWRs, the report shows that the inability to transfer decay heat from the reactor following a normal shutdown is the largest contributor to the core melt probability..." Neither the numerical data nor the methodology of WASH-1400 was used.

WASH-1400 was examined for justification of the staff's proposed RHR Shutdown position (single failure/safety grade/ seismic, etc.) to see if it did reduce the probability of core melt. It was found that the RHR position would not affect the WASH-1400 results since hot standby was considered to be a success path in WASH-1400. In a January 19, 1978 memo, NRR concluded that: "No quantitative assessment was made of the reduction in risk that would result from the proposed improvement in the RHR system (SRP 5.4.7), and the effect of a loss of the RHR cooling on risk was considered small and hence not evaluated." In conclusion, the staff recommended implementation of the "RHR shutdown position."

Subsequent to issuance of the Branch Technical Position, arguments were used in the justification of the need for Regulatory Guide 1.139, "Guidance for Residual Heat Removal." Additional bases for the regulatory position of Regulatory Guide 1.139 are provided in the discussion, and it is the view of the staff that the position would be unchanged if the WASH-1400 results had not been considered.

54. Synopsis: Two contracts were issued to investigate the feasibility of extending WASH-1400 techniques to the quantification of allowable outage times for ECCS components. The results of these studies formed part of the basis for slight increases in out-

age times. It is the staff's judgment that WASH-1400 data was not the principal bases for modification of these technical specifications.

55. Synopsis: In 1976, the staff concluded that the likelihood of a pump overspeed resulting from a LOCA was sufficiently small as not to warrant any action on the part of licensees at this time. This was based on an analysis performed in 1973 (before WASH-1400). Subsequently, in rereviewing the priority for staff efforts on this subject, probability values and analyses were used to determine quantitative estimates of the probability of core melt resulting from a PWR reactor coolant pump flywheel missile impacting on an ECCS line due to pump overspeed following a cold leg break. This task is now TAP B-68 and is not a high priority effort. The use of WASH-1400 appears to have been used to confirm the earlier analysis and no reconsideration is necessary.

56. Synopsis: The probability of an SSE was extracted from WASH-1400 for use in an enclosure to the RRRC working paper on overpressure protection while operating at low temperatures. This estimate of the frequency of earthquake induced overpressure events was used as the basis for the staff position that the system for overpressure protection during startup and shutdown need not be designed to Seismic Category I requirements.

The current limits of Appendix G were developed for relatively frequent events, i.e., anticipated transients. Higher, but not defined, limits would be acceptable for the lower frequency events such as earthquakes. Thus, while the current limits are sufficient, they may not be necessary for overpressure events resulting from earthquakes.

RRRC concluded that some protection against earthquakes was appropriate and specifically that the overpressure protection system must be designed to withstand the operating basis earthquake (OBE) While the data from WASH-1400 was considered when determining the seismic requirements, it was not used as a significant factor in the RRRC decision.

57. Synopsis: In evaluating Diablo Canyon with respect to long term residual heat removal following a safe shutdown earthquake and extended loss of offsite power, the staff accepted an RHR design where a mechanical failure of an isolation valve in the suction line could preclude activating the system.

In discussing this issue in SER Supplement No. 7, the staff considered the availability of the steam generators and the auxiliary feedwater system to provide long term heat removal and the probability of a mechanical valve failure combined with an earthquake. The staff concluded that the steam generator/auxiliary feedwater system is an acceptable backup to the RHR system for complying with the functional requirements of General Design Criterion 39. The justification of this position does not rely on any probability arguments, and therefore, is not compromised by the concerns with WASH-1400.

58. Synopsis: In the Shoreham review, the staff reviewed the positions for a recirculation pump trip in the event of a turbine trip or generator load rejection, to determine whether such a trip would not invalidate the pump coastdown conditions assumed in the ECCS calculations (which assume a coincident loss of offsite power at the time of the large loss-of-coolant accident). In the staff's evaluations, WASH-1400 was cited in arguing that this scenario "is extremely unlikely (with a median value on the order of 10^{-7} per reactor year). Therefore, based on this low probability of occurrence, we concluded that the applicant's response regarding recirculation pump trip was acceptable. The analysis for Shoreham is consistent with analyses previously accepted on other boiling water reactors."

In a memo of November 13, 1978 from RES to the staff on the same subject, data from WASH-1400 and a recommended rewording of the staff position (with the same conclusion) were provided. A discussion of the uncertainties associated with the WASH-1400 estimates was included.

59. Synopsis: WASH-1400 is referenced twice regarding BWR rod drop accidents in a June 17, 1975 memo from H. Richings to D. Ross. In the first reference, the absolute values of accident probabilities for severe BWR accidents were used in a relative way to support the choice of a probability criterion such that the occurrence of the accident need not be considered a design basis event. It should be pointed out, however, that the primary basis for the choice of the criterion was WASH-1270 (ATWS) and was made without use of WASH-1400 data or methods. The use of WASH-1400 was only supplementary in character to assess the selected probability criteria for rod drop accidents.

The second reference to WASH-1400 was with respect to the probability of human error. Again the reference was supplementary in character and primary reliance for the estimate of the probability for human failure was not based on the reference to WASH-1400. (see also Category 4, item 10).

60. Synopsis: In considering grid frequency decay, we stated in NUREG-0138: "Considering the likelihood of occurrence of excessive frequency decay and the release to atmosphere that would result from release of a portion of the total gap activity to the primary coolant system, an accident such as that postulated would represent a negligible portion of the reactor accident risk predicted in the Reactor Safety Study (WASH-1400)."

The risk from a grid frequency decay event was found to be a negligible portion of the WASH-1400 reactor accident risk. TAP A-35, a high priority task, will determine the maximum credible decay rate. Results of TAP A-35 will be used in TAP B-70 to determine whether additional requirements are needed.

61. Synopsis: In establishing the requirements for the design of the containment purge system, the staff considered the subject of allowable time that purging could occur during normal operations. The staff established this time by assuring that the probability of an accident with the calculated site boundary dose in excess of guideline values is reduced to an acceptable level. WASH-1400 estimates for probability of a LOCA plus staff estimates for the probability of an iodine spike were used in the analysis to conclude that large containment purge systems should not be used during normal plant operating modes more than about 1% of the time.

62. Synopsis: WASH-1400 estimates for fission product gap activity (Appendix VII) were used to affirm the use of Regulatory Guide 1.25 source terms in Regulatory Guide 1.89 to determine the radiation environment for qualifying electrical equipment. The more conservative source term of Regulatory Guide 1.25 was used in developing Regulatory Guide 1.89.

63. Synopsis: WASH-1400 was used to provide an estimate of the consequences of sabotage. However, the decisions to implement reactor sabotage regulations were not based on the WASH-1400 results but rather on the knowledge that sabotage could cause releases that would be harmful to the public. WASH-1400 is referenced in:

(1) "Safety and Security of Nuclear Power Reactors to Acts of Sabotage," SAND 75-0504 Sandia Laboratories, March 1976;

(2) Memo R. B. Minogue thru L. V. Gossick to B. Huberman, Director of Policy Evaluation transmitting a discussion of design threat levels entitled, "Basis and Rationale for Selection of a Design Threat Level for Power Reactors Sabotage Protection" prepared by SD staff, January 3, 1977;

(3) Transcript of the public hearings on the Material Access Authorization Program - "Rulemaking in the Matter of 10 CFR Parts 11, 50 and 70, Docket Rm-50-7, July 10, 11 and 12, 1978."

64. Synopsis: In denial of PRM 50-19, the calculated accident pressures in PWR and BWR reactors were compared to containment design and failure pressures from WASH-1400 to estimate the potential effectiveness of an evacuated containment to mitigate the effects of a Class 9 and lesser accident. Risk assessment results from WASH-1400 (i.e., probability of the events) were not used. Based on this comparison, it was shown that heavy vacuum containments would not substantially mitigate the effects of a Class 9 accident. It should be noted that in the staff's view that consideration of substantial uncertainties in the numerical estimates used would not effect the conclusion.

65. Synopsis: In supplemental testimony, the NRC staff addressed Contention I-7 and Contention I-2g, in the matter of Union Electric Com..y (Callaway Plant, Units 1 and 2), Docket Nos. STN 50-483 and STN 50-486. Contention I-7 alleged that the staff's analysis of the environmental impact for the proposed facility is inadequate because a loss-of-coolant accident followed by failure of ECCS are dismissed without

detailed analysis, in spite of the probabilities for such an incident being one in 17,000 per reactor year (WASH-1400). The staff testimony concluded that the draft WASH-1400 report did not present any information concerning the frequency of occurrence of the accident sequence described in Contention I-7 that alters the conclusion that the environmental risk of such an accident can be considered to be negligible and need not be considered further. (For more detailed discussion of this type of item see an appendix to this enclosure).

66. Synopsis: In supplemental testimony Darrell Eisenhut addresses Contention I-10, in the matter of Kansas Gas and Electric Company and Kansas City Power and Light Company, (Wolf Creek Generating Station Unit No. 1), Docket No. 50-482, Janaury 6, 1976. The Contention is similar to the Callaway Contention in Item 65 above. The conclusion regarding the draft WASH-1400 report is also the same as in the Callaway testimony.

Synopses of Category 4 Items

Definition

Includes those actions in which values of WASH-1400 were modified by the staff to reflect different data base or experience and were then used in the licensing process.

A possible example is the adjustment of the RSS estimates of scram unreliability in NUREG-0460.

Summary

There are 12 items in this category. Typically, the issues identified used WASH-1400 data as modified or supplemented by the staff to reflect added experience or a different data base before using the more complete information in the licensing process. For example, WASH-1400 data on pipe ruptures was considered along with data obtained by the staff during its review of water hammer events at operating plants.

While the additional failure rate information gathered from operations provided a more complete data base, the decision to proceed with water hammer as a generic issue was based principally on other considerations. In the staff's view, none of these items warrant reconsideration.

1. Synopsis: With regard to water hammer, there is no specific reference to WASH-1400 in Section 3 "Basis for Continued Plant Operation and Licensing Pending Completion of Task" of TAP A-1. However, the WASH-1400 estimates of pipe rupture probabilities have been considered along with data on pipe cracking or rupture obtained during the staff review of water hammer events. In view of the low probability of piping failure due to water hammer and the corrective actions being taken with respect to water hammer in PWR steam generators, continued operation and licensing of plants can proceed while Task A-1 (high priority task) is being conducted.

2. Synopsis: Evaluations of core debris behavior following meltdown have been performed for the CRBRP, FFTF and FNP. These have utilized the molten core-concrete penetration evaluation models, data and results of WASH-1400 modified as appropriate based on more recent analyses or experimental results.

3. Synopsis: With regard to intersystem LOCA, WASH-1400 identified the intersystem LOCA in a PWR as a significant contributor to the risk resulting from core melt. The staff has analyzed this and other similar scenarios using the general methodology and the data of WASH-1400. Memo dated July 3, 1978 from Buhl to Novak provided minor comments on NRR Intersystem LOCA Analysis. Minor changes in terminology and definition of terms were recommended. The staff analyses are limited to those sequences which are significant contributors to risk in relation with the WASH-1400 results. Using these analyses, the staff plans to determine leak testing frequency.

4. Synopsis: With regard to the use of probabilistic assessments of reliability, we stated in NUREG-0138 that:

"The staff agrees that present technology does not permit a rigorous demonstration of the WASH-1270 objective of 10^{-7} per reactor year. As shown by the

Reactor Safety Study (WASH-1400), however, the use of a reactor protection system with a low unavailability, plus additional capability provided by other systems to limit transients, prevents anticipated transients without scram (ATWS) from being the predominant contributor to core melt probability for light water reactors (LWRs). The conclusion supports the staff position that an acceptable level of safety can be achieved by use of reliable transient-limiting systems in conjunction with a highly reliable reactor protection system." (See also Category 1, item 1).

5. Synopsis: With regard to protection against single failures in reactivity control system, we stated in NUREG-0138 that:

"The release to the environment resulting from such release of gap activity to the primary cooling system would represent a negligible contribution to the reactor accident risk predicted in the Reactor Safety Study (WASH-1400). An in-depth review of the analyses has not been carried out since the transients have not been generally judged to be a Condition II event and the reviews have been commensurate with the apparently small safety significance of the event. The analyses which have been submitted, however, have been reviewed and none have been found unacceptable."

Conclusion of the staff in NUREG-0138 was that single failure of reactivity control system was a small contributor to overall accident risks. Comparison with WASH-1400 values was supporting in nature. Staff judgement was made independent of WASH-1400.

6. Synopsis: The RSS consequence model (CRAC code) was used to calculate consequences of a core melt at the GETR. Results were transmitted informally to and at the request of PSS/NRR. Not documented and approach abandoned.

7. Synopsis: As part of evaluation of Diablo Canyon for interim license (which has not been used) the Probabilistic Analysis Staff prepared a summary evaluation of the risk of operation of Diablo Canyon for a range of probabilities of a seismic event. (See also Category 3, item 38).

8. Synopsis: Memo from I. B. Wall to E. G. Case, dated June 29, 1976, Subject: Proposed Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants." This memo provides comments on the proposed Regulatory Guide from the standpoint of overall public risk based on diesel generator unavailability. It concludes that the position advanced in the Guide "appears unnecessarily stringent from the standpoint of public risk." The guide was subsequently [illegible] as proposed. (See also Category 5, item 22).

9. Synopsis: In Exhibit A, Section 6, Part IV of the Nuclear Energy Center evaluation an accident risk analysis is provided utilizing the WASH-1400 consequence model with data modified by the staff to reflect the specific design considerations of a nuclear park.

10. Synopsis: Memo from M. A. Taylor and W. E. Vesely to I. Wall, dated August 6, 1975, Subject: BWR Rod Drop Accident. This memo provides comments on the June 17, 1975 memorandum from H. J. Richings (See Category 3, item 59). This probability study by Richings was made without use of WASH-1400 data or methods. The use of WASH-1400 was supplementary to assess the selected probability acceptance criteria for rod drop accidents.

11. Synopsis: The staff is presently reevaluating the effectiveness of existing transportation regulations in protecting the health and safety of the public. To a very great extent, that reevaluation is depending on quantitative risk assessment. There is, of course, little in common between reactor accident probabilities and transportation accident probabilities. But there is some similarity in accident consequences and post-accident cleanup between the two. Therefore, the staff is using the consequence analysis portions of WASH-1400 in the transportation analyses. These uses are documented at this time in NUREG-0170 (Vol. 1) and a Sandia contractor report SAND 77-1927.

The Sandia report is a precursor of a staff environmental statement.

The staff use of quantitative risk assessment in general, and WASH-1400 material in particular has been cautious and critical. Some aspects of the staff's questions on the validity of this risk assessment are addressed specifically in the overall summary and conclusions of NUREG-0170 (Vol. 1, p. ix). No rulemaking action has yet been taken on the basis of these risk assessments.

12. Synopsis: In May 1976, Big Rock Point was granted exemptions from ECCS single failure criteria. These included two cycle No. 14 exemptions, which were short term in nature and have expired. These applied to a LOCA followed by a failure in the ring spray system and to the failure of the onsite diesel generator in the absence of off-site power for ECCS long-term cooling. These exemptions were granted subject to the licensee making several plant modifications prior to returning to power.

In May 1976, Big Rock Point was also granted a lifetime exemption from the single failure criteria as applied to a LOCA caused by a break in either core spray system.

The staff based its decision on technical judgements of the overall performance capability of the entire ECCS and design margins at Big Rock. The staff also performed in a supplemental manner, a reliability assessment of selected systems at Big Rock using methodology similar to that in WASH-1400. WASH-1400 failure rates were modified on a judgemental basis to better reflect specific conditions at the plant. (See also Category 3, item 26.)

Synopses of Category 5 Items

Definition

Includes those actions in which the event tree/fault tree methodology of WASH-1400 was used in the licensing process, but no reliance was made on the specific numerical estimates of WASH-1400.

Summary

There were 41 items identified in this category. The items in this category used the evaluation techniques of WASH-1400. An example of this use is in the evaluation of vendor proposed computer protection systems. In these reviews, the staff performed preliminary reliability assessments using WASH-1400 methodology. These results aided the staff in their deliberations. None of the items in this category warrant reconsideration.

1. Synopsis: The staff utilized the event tree/fault tree methodology of WASH-1400 to evaluate the reliability of the CRBRP Shutdown Heat Removal System. This evaluation was used in parrallel to the staff's deterministic approach (i.e., diversity, redundancy, etc.) and provided additional insight on design changes and their contribution to achieving the required diversity and redundancy to meet the applicable General Design Criteria.

2. Synopsis: A study of comparative risk evaluations utilizing event and fault trees for advanced reactors is being done utilizing WASH-1400 type methodology. The objective of this work is to

provide early guidance on the licensability (i.e., conformance with the well-established regulatory criteria and practices) of a given advanced reactor relative to the present generation of LWRs.

3. Synopsis: Section 7.1.2.5 of the Report to ACRS on RESAR-414 describes the Westinghouse design verification program for the Integrated Protection System (IPS). The program will include a system reliability analysis based upon techniques similar to those in WASH-1400. Staff reviewers have been instructed to consider the recommendations of the Lewis Committee in its evaluation of this proposal.

4. Synopsis: A review of the methods associated with the analysis of systems interactions and common mode failures was performed at Brookhaven (BNL-NUREG-23815). This report uses event and fault trees and involves an evaluation of methods and techniques available for a qualitative and quantitative study of systems interactions and common mode failures. This was only a draft report and was not used in any staff licensing actions.

5. Synopsis: In response to ALAB-444, the staff has provided testimony on generic issues and plans for their resolution (Yellow Creek, Black Fox, Perkins, etc.). These discuss the task action plans and the bases for continued operation.

In some instances, WASH-1400 has been noted in connection with a risk-based evaluation of the priorities for resolution of these generic issues. (The draft of this risk based evaluation was included in the paper on unresolved safety issues, SECY-78-616, and will be included in a "for comment" document on NRR's priorities for generic issues to be issued in the near future.)

6. Synopsis: WASH-1400 methodology was used for a preliminary analysis of the ANO-2 core protection calculator system. The analysis was not used in the final decision on ANO-2. Similar methodology was used in evaluation of reliability of B&W RPS-II and Westinghouse IPS. None of these analyses has been used or referenced in a licensing action.

7. Synopsis: In the evaluation of the acceptability of the BSAR-205 Decay Heat Removal (DHR) Suction Relief Valves as Overpressure Protection while Operating at Low Temperatures, operator error data was extracted from WASH-1400 to assist in evaluating the potential for an overpressurization event to occur while the DHR relief valves were isolated. Use of the WASH-1400 data was not the basis for accepting this design. The primary bases for determining the above scenario to not be a concern was:

   (1) Normal plant operating procedure is to maintain RCS temperature and pressure well below the Appendix G limit.

   (2) The very slow rate of pressure increase for the loss of DHR transient (~ 6 psi in 10 minutes).

(3) Sufficient indication to alert the operator.

(4) Staff requirement for additional alarms.

8. Synopsis: In the staff response to a Board question (North Anna, Units Nos. 1 and 2), reference was made to Regulatory Guide 1.120 which includes the following statement:

"Although WASH-1400, Reactor Safety Study, An Assessment of Accident Risks in U.S. Commerical Nuclear Power Plants, dated October 1975, concluded that the Browns Ferry Fire did not affect the validity of the overall risk assessment, the staff concluded that cost-effective fire protection measures should be instituted to significantly decrease the frequency and severity of fires and consequently initiated the development of this guide." (See also Category 3, items 4, 5, 6 and 7).

9. Synopsis: In the development of SRP's 3.6.1 and 3.6.2, probability was used as a rationale to:

1) justify break exclusion for "super pipe,"
2) determining failure mode difference between high and moderate energy piping i.e., breaks vs. cracks, and,
3) justify exemption of single active failures for certain piping systems.

Probability was also used as a partial basis for excluding certain primary piping breaks from consideration as CDA initiators in Clinch River and FFTF.

10. Synopsis: In the development of SRP 10.4.9 diesel generator reliability operating experience was used as a probability data base coupled with probability of loss of offsite power to support the staff position on requiring diverse power supplies for auxiliary feed systems.

11. Synopsis: NRR comments on the working paper for Regulatory Guide 1.63 (Rev. 2) regarding electrical penetrations for pump power supplies in containment included the following statement:

"We have performed a probabilistic analysis using the above failure data (failure rate calculated at the 95% confidence level); the established LOCA probability of 10^{-4} per reactor year; and conservative assumptions regarding the time intervals during which the pump penetrations would be subject to failure (while energized) given that a LOCA occurred first, or during which a plant is subject to a LOCA (while not a cold shutdown) given that a pump penetration failure occurred first. Our detailed calculations are shown in Enclosure 2. The results of this analysis indicate that the probability of a LOCA concurrent with a pump penetration short circuit failure is less than 3.6×10^{-9} per year. This is considered to be an insignificant risk to the public health and safety. In our opinion a regulatory requirement directed toward reducing this risk cannot be justified, and may in fact have a negative impact on safety by diverting both applicant and staff resources from matters of greater safety significance."

12. Synopsis: In the description of Generic Issue Task Action Plan A-25, the following statement is included:

"The approach selected for problem resolution is that of a reliability analysis of typical plant onsite Class IE power systems."

This is an on-going program and Commission guidance on use of the RSS will be considered.

13. Synopsis: In our study to assess the effects of postulated event devices (snubbers) on normal piping system operation, an analysis will be performed to quantify the probability of deleterious interaction of such devices with the piping system.

14. Synopsis: During the period in which generic activity on Task Action Plan A-2 regarding asymmetric loads on RV supports was progressing, several plants were licensed prior to the completion of our complete evaluation based on scoping calculations, design conservatisms and the low probability for pipe rupture. (See also Category 3, item 21.)

15. Synopsis: During a general review of the turbine missile problem, we performed a risk assessment review of the valves which are part of the turbine control system. Based on data which was available, a failure probability as a function of valve inspection frequency was determined for use in the overall turbine missile study. (See also Category 3, item 39.)

16. Synopsis: In a talk by Dade Moeller of ACRS, presented at the 15th DOE Nuclear Air Cleaning Conference, LER data for Containment Spray System Failures was compared to WASH-1400 failure data. No licensing actions are involved with this item.

17. Synopsis: LASL under technical assistance contract to the NRC is using fault tree and event logic in analyzing nuclear plant vital areas as part of the security plant review. Fault trees from WASH-1400 have been used as part of the overall logic structure. No numerical estiamtes from WASH-1400 have been used. The results of the evaluation are transmitted from LASL to RSLB in a letter report that is withheld from public disclosure in accordance with 10 CFR 2.790(d). The site specific fault trees/event trees are classified as Confidential NSI and are kept in approved security repositories at either LASL or RSLB.

18. Synopsis: In SECY-77-388A, the staff proposed guidelines for the preparation of Value-Impact analysis. In an example of where further action may be needed, WASH-1400 techniques were referenced as the type of analysis that could be conducted.

19. Synopsis: Memos from I. Wall to G. Arlotto dated June 30, 1975 and July 3, 1975. Subject: IEEE/NPEC/P577, Draft 1, "Reliability Requirements in the Design and Operation of Nuclear Power Generating Stations." These memos present detailed comments on the above cited draft. The comments relied on insights from WASH-1400.

20. Synopsis: Memo from S. Levine to V. Stello dated June 24, 1976, Subject: DOR Re-review Program for Operating Nuclear Power Plants. This memo discusses the difficulty of applying risk assessment to the re-review program. The memo relied on WASH-1400 insights and recommended that "engineering insights from the Reactor Safety Study as opposed to partial risk assessments be used to supplement the standard licensing reviews."

21. Synopsis: Memo from S. Levine to H. Lowenberg dated July [illegible]3, 1976, Subject: Review of GESMO Chapter IV, Section C. This memo provides comments on the environmental risks associated with Class 1-9 accidents and recommended deletion of reference to WASH-1400 (on pg. IV-C-159) since the RSS did not consider Pu recycle.

22. Synopsis: Memo from I. Wall to T. Wilson dated December 13, 1974, Subject: Statistical Analysis of Diesel Failure Data. This memorandum encloses a report on statistical tests performed on data obtained on diesel generator performance. The methods used are similar to those that were used to evaluate data in WASH-1400.

23. Synopsis: Memo from W. Vesely to A. Thadani dated September 23, 1976, Subject: Review of EPRI Report "ATWS Reappraisal" (EPRI NP-251). The memo relies on techniques similar to those in WASH-1400 to criticize the EPRI report.

24. Synopsis: Memo from S. Levine to R. Boyd dated October 8, 1976, Subject: Responses to NRDC et al Fourteenth Set of Interrogatories in CRBRP proceeding. This memo relies on insights from the Reactor Safety Study to respond to interrogations. (See Category 3, item 48.)

25. Synopsis: In a January 19, 1977 memo, S. Levine sent comments to G. Arlotto on the Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes. In the memo reference was made to the risk assessment contained in the EIS. Also, the memo indicated that the use of data from WASH-1400 for latent cancer fatalities instead of from the BEIR report was criticized by EPA.

26. Synopsis: Memo from I. Wall to S. Smiley, dated July 30, 1976, Subject: Review of "National Security and Accident Recovery Considerations of Nuclear Energy Center (NEC) Siting," by G. A. Cristy, C.V. Chester, and R. O. Chester, ORNL-5036. This memo provides comments on the above cited report and relied on insights from WASH-1400.

27. Synopsis: The June 16, 1977 memo from S. Levine to E. Case and R. Minogue transmitted RIL-12, Modifications to Pressure Vessel Failure Probability Prediction. The draft reports contained sensitivity studies on the effects of the new modifications and updated failure probabilities.

28. Synopsis: In a June 14, 1977 memo I. Wall sent to D. Skovholt the results of PAB's review of the Study of NRC QA Programs by Sandia Laboratories. The comments dealt with the reliability analysis and probabilistic techniques used in the study.

29. Synopsis: The November 9, 1977 memo from S. Levine to E. Case transmits RIL-18 on the FRANTIC Computer Code. The code calculates system unavailability.

30. Synopsis: In a November 17, 1977 memo I. Wall sent I. Roberts comments on ANSI-N635, Draft 3, Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites. PAB criticized the probability and risk assessments used in the draft Standard.

31. Synopsis: In a July 26, 1977 memo M. Taylor sent S. Pawlicki comments on a paper by S. Bush titled, "A Reassessment of Turbine Failure Probability." No specific mention of WASH-1400 is made. (See also Category 3, item 39.)

32. Synopsis: In a July 27, 1977 memo I. Wall sent R. Moore comments on a proposed contract with Control Analysis Corporation. The study would furnish methods for predicting the probability of the coincident occurrence of several natural or man-made hazards to nuclear power structures, systems and components.

33. Synopsis: In an August 23, 1977 memo W. Vesely transmitted information on probabilistic analyses of test interval effects to V. Nerses. The information addressed system unavailability and relied on WASH-1400 insights.

34. Synopsis: Memos from Buhl to Mattson dated February 3 and March 20, 1978 provided comments on Draft III of the NRR report on ATWS. Specific comments related to the scram failure synthesis models and dealt with the conservatisms used in the analysis as well as models used. (See also Category 1, item 1.)

35. Synopsis: Memo from Buhl to Kahnemuyi dated April 20, 1978 provides comments on criteria contained in ANSI-N658 on single failures. Comments discuss the use of probabilistic technology and recommend concurrence in proposed ballot.

36. Synopsis: Memo dated January 23, 1978 from S. Levine to E. Case providing RES comments on the draft working paper of the Liquid Pathway Generic Study. Principal comments related to WASH-1400 methods used in the LPGS. (See also Category 2, item 3.)

37. Synopsis: Memo from W. Vesely to G. Vissing dated December 18, 1975, Subject: Regulatory Guide "Periodic Testing of Diesel Generators Used as Onsite Electrical Power Systems at Nuclear Power Plants." Evaluations were performed to determine the reliability and risk implications of the proposed testing scheme. Analytical techniques were used that are similar to those used in WASH-1400.

38. Synopsis: Memo from I. Wall to R. Minogue dated March 4, 1976, subject: Minutes of Meeting Held on March 1, 1976 to Discuss Degree of Conservatism in the Draft Environmental Impact Statement on the Transportation of Radioactive Materials. Comments were based on techniques and insights from WASH-1400. (See also Category 4, item 11.)

39. Synopsis: Memo from S. Levine to R. Heineman dated March 26, 1976, Subject: Examination of the Seismic Design Basis for Fire Protection Systems. This memo provides an analysis directed to the question of whether fire protection systems should be

designed to seismic Category I systems. Improved data obtained since publication could modify results and widen error bounds but the general conclusions would be expected to remain valid. (See also Category 3, item 7.)

40. Synopsis: Risk assessment has been indirectly considered in the Mark I Short Term Pool Dynamic Program (NUREG-0408). The conclusion of the Short Term Program (STP) was that, based on the demonstration of a minimum safety factor of two against failure, the Mark I plants could continue to operate during an interim period of about two years while a methodical and comprehensive Long Term Program is conducted. This conclusion was based on the use of most probable loads for the postulated LOCA and without an evaluation of Safety Relief Valve Loads. This approach was found acceptable on the basis of the low probability of a LOCA during the nominal two years needed to complete the Long Term Program. Consideration was also given to the low probability of a LOCA in establishing the Mark I technical specification related to ΔP operation which imposes a positive pressure in the drywell relative to the wetwell so that in the event of a LOCA the pool dynamic loads are reduced.

The conclusions of the Mark I STP are only valid for Mark I plants under ΔP operating conditions. Plants are allowed to operate in a non-ΔP mode for the limited periods specified in the Technical Specifications based on the expected low probability of a LOCA during this time limited period. (Although this item was originally included under Category 3, it now appears that no use of WASH-1400 was made and thus is included herein.)

41. Synopsis: Memo from Vesely to Staley, DSE, from Vesely to Ayer and from Vesely to Burkhardt dated June 7, 1978 providing an analysis of flood frequency of the Kiskimentas River using risk assessm . methodology (no use was made of the log-n. rmal method commented on by the Lewis Report).

Synoposes of Category 6 Items

Definition

This category was added after the survey responses were received. Issues were placed in this category when they could not be considered to fit into any other categories. Included here are instances when the staff considered using WASH-1400 in the licensing process but dismissed it and staff reviews of WASH-1400 information used by other agencies in their evaluations. Also included here is correspondence with members of Congress, public and other Federal Agencies which were identified by staff during this survey. This correspondence does not involve specific licensing decisions.

1. Snynopis: "Report on the TVA Seismic Issue by NRC Staff Working Group" considered, but recommended against, use of WASH-1400 as an aid in determining seismically-induced core melt sequences. The use of WASH-1400 was considered, but rejected.

2. Synopsis: Additional remarks by ACRS member Dr. Okrent in the Committee's Report on Perkins/Cherokee (April 14, 1977) included a critical comment about the estimates of the contribution of earthquakes to overall nuclear reactor safety risk, as given in the Reactor Study (WASH-1400). . The Hearing Board then requested written material that addresses the reservations of ACRS member Okrent. Written material pertaining to quantification of inherent safety margins on seismic design was provided. During the hearing, the Board pursued the question of how the staff rationalizes

their position on setting the design basis earthquake against the probabilities. A staff witness, stated that the staff did review a draft of WASH-1400 and did make comments, but that the staff had not then (July 21, 1977) adopted that report or any similar procedure on its licensing review actions.

3. Synopsis: In the rulemaking hearing for 10 CFR 11 held in Washington, D.C., on July 12, 1978, the staff referred to the "consequence tables" in WASH-1400 during presentation of testimony. The references to WASH-1400 were in response to questions about the worst possible consequences of sabotage. The consequence tables are in terms of releases from the plant and do not rely on the consequence model.

4. Synopsis: Basic data referenced in the draft WASH-1400 concerning natural gas pipe line failure rates was used in the preparation of the environmental statement on the Bear Creek Project of Rocky Mountain Energy Company, Docket No. 40-8452. However, such data would have been available and might have been used by the NMSS staff whether or not it had also been used in WASH-1400.

5. Synopsis: Draft input in the Seabrook alternative site review contains results of limited studies that led the staff to conclude that population density is a sufficiently crude indicator that relatively large differences in population densities between two sites would be required before significant differences in residual risks at these sites could reasonably be expected. The final alternative site analysis report was issued in December, 1978. In the process of final editing the reference to population densities and its relationship to residual risks was deleted. The deletion was the result of a major attempt to reduce the size of the document and was not related to the comments provided by the Lewis Committee on the RSS.

6. Synopsis: Commissioners information cards contain information related to risks from various non-nuclear and nuclear accidents. Data used was compared to WASH-1400.

7. Synopsis: The Annual Reports for 1975, 1976, 1977 and 1978 discuss WASH-1400 and some uses of the results.

8. Synopsis: An extract from the November 18, 1978 issue of National Journal discusses the Rasmussen Report.

9. Synopsis: A December 8, 1978 memo from Levine to Denton provides three additional items identified by RES that utilized the insights of WASH-1400. They are a letter to Senator J. Glenn dated December 9, 1976 and copies of NUREG-0138 and NUREG-0153. The letter to Senator Glenn provides responses to questions about the discussions by NRR of issues in NUREG-0138. Specific issues of NUREG-0138 and NUREG-0153 are discussed in other synopses.

10. Synopsis: Letter to G. Paulson, Assistant Commissioner for Science, Department of Environmental Protection, State of New Jersey, on minutes of a meeting in New Jersey on March 21, 1977, regarding staff studies of the FNP. The releases associated with a steam explosion at a floating nuclear plant were compared to an analysis of steam explosions in WASH-1400. (The meeting minutes refer to the staff's use of WASH-1400 to draw comparisons between the FNP and similar land-based plants -- see Category 2, item 3).

11. Synopsis: In responding to W. D. Rowe's (EPA) request for further studies of nuclear accident risks, a letter dated November 18, 1976 notes

the staff's safety reviews and their objectives and follows with this statement that "the Reactor Safety Study indicates that the approach to safety as set forth in the Commission's regulations has been successful and that the safety and environmental risks from accidents are lower than the risks from most other natural and mancaused events." This language is patterned after the 1974 Interim General Statement of Policy. The letter ends with agreement on the need to pursue this matter further.

12. Synopsis: Letters from S. Levine to G. Paulson, New Jersey Department of Environmental Protection dated November 9, 1976, and June 20, 1977 provide comments on investigation conducted for the State of New Jersey of the probability of hypothetical catastrophic accidents in the Oyster Creek Nuclear Power Plant. The use of certain results in the Reactor Safety Study by the author of the Oyster Creek study is questioned in this letter. The critique includes a discussion of how the results in the Reactor Safety Study were generated. In addition, the extrapolation of failure probabilities over a 30-year time period is discussed and compared to the 5-year time period extrapolation in the Reactor Safety Study.

13. Synopsis: Memo from Buhl to Vollmer dated June 6, 1978, provides comments on GSA's DES regarding disposal of Charlestown site. Material used in the GSA's DES regarding WASH-1400 was critiqued. This critique was included in comments provided to GSA on their DES.

14. Synopsis: The AAB input to the proposed response to Congressman Pattison's letter of April 2, 1976 describes generally the NRC's siting criteria, and the relationship of Class 9 accident risks to the NRC's preference for sites in areas of relatively low population density. WASH-1400 is mentioned as a source of information (the Executive Summary and the Main Report to WASH-1400 were enclosed).

15. Synopsis: A letter to Ms. Phyllis Taber dated May 20, 1976 regarding the siting and safety of nuclear power plants discusses Commission's regulations and safety requirements, transmits the Main Report of the Reactor Safety Study (and notes that the Report "will help to allay your concerns about the safety of power reactors").

16. Synopsis: The letter to Lash and Cotton, NRDC, dated October 4, 1976 relating to an EPA proposed generic evaluation of risk acceptability quotes former Chairman Anders statement on the overall assessment of the Reactor Safety Study but notes that more work remains to be done on this matter.

17. Synopsis: Note from Emergency Planning Branch to J. Lafluer commenting on some EPA studies, dated May 28, 1976. NRC comments note that EPA utilized WASH-1400 dose conversion factors although some discrepanices were identified (and NRC staff recommended that the discrepancies be eliminated).

18. Synopsis: Letter to W. D. Rowe (EPA, dated April 5, 1977) regarding staff's intent to extend the WASH-1400 methodology to more likely events. This letter states that the NRC intends to extend the RSS methodology to more likely events (Class 3-8 accidents). This program is currently in progress in connection with Task Action Plan A-33. The NRC's policy statement and supplementary guidance to the staff will be considered in further efforts on this task.

19. Synopsis: Letter to John E. Ward (AIF) dated September 1, 1978 re: SECY 78-137 and the staff's intended use of Class 9 accident considerations. The letter states that we believe that the Reactor Safety Study consequence model can provide useful insights into a few situations but we are aware of the need to be cautious in the direct application of any such analyses.

20. Synopsis: Response (June 1, 1977) to Congressman Moorhead provides information for responding to concerns of a constituent

regarding the U.S. District Court decision that the Price-Anderson Act was unconstitutional. The letter explains the major provisions of the Price-Anderson Act. Reference is made to the RSS to show that the risk of accident in excess of $560 million is extremely remote.

21. Synopsis: Response (June 12, 1975) to Mr. Murphy, JCAE, discusses the legislative status of the Price-Anderson Act and a vetoed version of the bill (H.R. 15323) which would have extended its expiration date. The letter also indicates that Dr. N. Rasmussen testified that he considered the present $560 million limit on liability was a reasonable value. The basic conclusion of the Draft WASH-1400 report stating reactor risks are smaller than other man-made and natural risks is also stated in the letter.

22. Synopsis: Response (June 2, 1978) to Congressman Hamilton provides information for responding to a constituent concerns that electrical generating costs are subsidized by providing liability insurance for nuclear plants. The response provided an average annual loss from nuclear power plant accidents based on WASH-1400. The response indicates that WASH-1400 had been the subject of controversy. The staff estimated the uncertainty in the estimate to be about a factor of 10.

23. Synopsis: In a November 11, 1976 letter from W.J. Dircks to Hon. L.M. Hamilton regarding decontamination processes, reference to the probability and consequences of a core melt as stated in WASH-1400 was made. Since no licensing action was taken no reconsideration is necessary.

24. Synopsis: Memo from I. B. Wall to R. DeFayette dated August 23, 1976, Subject: Draft Responses for California State Energy Resources Conservation and Development Commission. This memo uses results from the Reactor Safety Study to illustrate the distinction between the design basis accident used for preparation of emergency plans and the Reactor Safety Study. In addition, further clarification was provided regarding evacuation and relocation as used in the Reactor Safety Study. A copy of this memo was transmitted to R.W. Houston by I.B. Wall on September 14, 1976.

25. Synopsis: Mar. 15, 1977 memo from S. Levine to R. Ryan discussed the Program Plan being developed by Sandia Laboratories on Emergency Planning and Response Evaluation. This work is based in part on the models and methodology of WASH-1400.

The NRC/EPA Task Force has used information in the RSS as a basis to perform calculations which illustrate the likelihood of certain offsite dose levels given a core melt accident.

The results derived from the RSS based work served to confirm the Task Force judgment that offsite planning for a generic distance around nuclear power plants is prudent and useful.

Memo from Levine to Ryan, SP, dated May 22, 1978 provides comments on draft NUREG-0396.

26. Synopsis: Memo from S. Levine to R.G. Ryan, dated October 7, 1976, subject: Comments on EPA Draft Publication Concerning the Technical Bases for Dose Projection Methods to be Used as a Basis for Protective Actions for Nuclear Incidents. The Comments in the memo use results of the Reactor Safety Study to illustrate points made in the review.

27. Synopsis: Letter from S. Levine to H.B. White, Sacramento County, California, dated June 30, 1976. This letter provides some clarifying information regarding WASH-1400 in terms of establishing an appropriate basis on which to formulate emergency plans. (See also item 24 above).

28. Synopsis: Letter from H.J.C. Kouts to W.D. Rowe, EPA dated July 7, 1975, regarding Emergency Response Protective Action Guides. This letter forwards comments to EPA on the Protection Action Guides. The comments relied on insights from WASH-1400.

29. Synopsis: A staff member in response to this survey noted, "It is expected that our future work dealing with responses to dynamic loadings will use probabilistic techniques for combination methods, or as the rationale for decoupling."

30. Synopsis: Letter to Senator Case dated October 2, 1978 provided information responding to a constituent's concern regarding consequences of a core melt. Since the letter was prepared subsequent to the Lewis Committee Report no references to WASH-1400 were made.

31. Synopsis: An April 12, 1978 report to Congress on research to improve LWR safety utilized the methodology to help establish what research should be accomplished to improve reactor safety.

32. Synopsis: Letter S. Levine to Dr. J. Baroff, dated May 13, 1976 responding to several questions directed toward bounding some calculations in the RSS. Human errors, sabotage and complete loss of off-site and on-site AC power are discussed in this response. This information was provided to assist Mr. Baroff in preparing for a presentation to the Governor's Conference.

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APPENDIX TO ENCLOSURE 1

DISCUSSION OF CONCERNS RELATED TO WASH-1400 RAISED BY THE

UNION OF CONCERNED SCIENTISTS

DISCUSSION OF CONCERNS RELATED TO WASH-1400 RAISED BY THE

UNION OF CONCERNED SCIENTISTS

The staff's use of the RSS was commented on in a Union of Concerned Scientists (UCS) press release issued on January 26, 1979. The first two paragraphs of that release stated:

> "The government may have to require shutdown of at least 16 of the country's nuclear power stations, the Union of Concerned Scientists said today.
>
> The continued operation of these 16 plants, despite officially acknowledged safety defects, has been allowed based on theoretical risk estimates from the so-called Rasmussen Report. (See Attachment 1 and 2 for plant names and locations.) In an unprecendented act, the government repudiated the Rasmussen Report last week."

Attachment 1 of the UCS press release listed 12 operating plants that the NRC Staff had previously identified as "Plants Requiring an Alternate or Dedicated Shutdown System" in a table following page 4 of our memorandum to the Commission of July 6, 1979, concerning the UCS Petition for Reconsideration of May 2, 1978. As discussed in our memorandum to the Commission, this Staff finding was based on our ongoing fire protection reviews of operating plants.

Attachment 2 of the UCS press release listed 6 operating plants in which "defective equipment was discovered as a result of the Union of Concerned Scientists' Petition to the NRC of November 1977." These plants had been previously identified in various NRC Staff memoranda to the Commission on the UCS petition as plants in which electrical equipment in safety systems had been replaced or requalified because of the inability of licensees to demonstrate, on the basis of available information, that the equipment was environmentally qualified for appropriate accident conditions.

Two of the plants listed in UCS's Attachment 2 were also listed in Attachment 1. The combination of these two lists apparently provided the basis for the number of plants specified in the title and lead of the UCS press release.

In another attachment to the press release, UCS listed and discussed "Three serious safety hazards which the NRC has not acted upon because of past NRC reliance on the Rasmussen Report." These three items were "Safety system electrical cables will fail in fire (discussed above); Safety system equipment cannot withstand accident it is designed to control" (also discussed above); and "Catastrophic accidents for which there is no protection required."

Based on the foregoing, it appear, that the principal UCS concerns related to the use of WASH-1400 that formed the basis for its press release of January 26 are fire protection measures in operating plants, performance of electrical equipment in safety systems of operating plants, and class 9 accidents in operating plants. UCS concerns over staff use of WASH-1400 in response to the UCS petition concerning fire protection and electrical connectors were also expressed in its letter to the Commissioners of October 16, 1978. UCS concerns over the role of WASH-1400 in defining Class 9 accidents were also expressed in its letter to the Commissioners of November 1, 1978.

The fire protection, electrical equipment and class 9 issues are discussed in more detail in the enclosures to this report.

FIRE PROTECTION MEASURES IN OPERATING PLANTS

NRC regulations require, in General Design Criterion 3 (10 CFR Part 50 Appendix A), that structures, systems and components important to safety of nuclear power plants be designed and located to minimize, consistent with other safety requirements, the probability and effects of fires. Prior to the fire at the Browns Ferry Station that occurred on March 22, 1975, NRC staff requirements for fire protection were minimal. Since that time, considerable staff effort has been devoted to developing comprehensive guidance to assure that nuclear plants have adequate fire protection systems. This effort has produced the guidance set forth in Section 9.5.1 of the NRC Standards Review Plan and in Regulatory Guide 1.20. These review procedures are used to evaluate fire protection systems for all facilities. And, although they contain considerably more stringent requirements than were imposed prior to 1975, they are being backfitted on all existing operating plants.

During this backfitting period, the safety of the nuclear plants during the time interval taken to implement these requirements has been questioned. In fact, this question has been of paramount importance and the focus of staff attention since the Browns Ferry Fire. This question was raised during Congressional Hearings in September 1975 and was most recently resurrected by the Union of Concerned Scientists (UCS) in the January 26, 1979, press release announcement. The basis for the most recent questioning is the premise that the reliance of plant safety was based mainly on the results of the Reactor Safety Study (WASH-1400). This premise is incorrect.

The basis for the staff conclusion that these plants can continue to operate while these modifications are being made does not depend on the conclusions of the WASH-1400 study. No reliance has been placed by the staff on quantitative analyses derived from WASH-1400 in the licensing reviews of fire protection systems for operating plants. The NRC's Special Review Group Report on the Browns Ferry Fire (NUREG-0050), and WASH-1400 Appendix XI indicate the limited applicability of the Reactor Safety Study calculations on fire risks. These reports state that rather straightforward measures can be used to improve fire protection and fire fighting capability and that these improvements would significantly reduce the likelihood of a core melt accident caused by fire. As will be discussed further, the staff's conclusions on plant safety is primarily dependent on these measures and not on quantitative assessments of risk caused by fire.

The staff's current judgment that the probability of occurrence of serious fires is low in operating plants is based upon the plant conditions in 1975, the subsequent positive and effective actions taken to improve the fire protection programs at nuclear plants since that time and the plant conditions observed during our site visits. This judgment is independent of and corroborates with similar judgments arrived at previously by the Browns Ferry Special Review Group and Reactor Safety Study.[1]

This staff conclusion on continued plant operation pending the implementation of additional fire protection measures was expressed in a memorandum of December 15, 1977, to the Commission on the UCS petition which presented the overall staff conclusion and recommendations concerning the petition. Page 33 of that memorandum includes the following statements:

> "6. Basis for Continuation of Plant Operation and Licensing
> The staff has previously indicated its basis for the continued operation of licensed plants pending completion of the full implementation of our current fire protection guidance in the November 9, 1977 report (pp. 8-9) and the November 22, 1977 report (pp. 12-18). This basis includes, 1) the actions taken as a result of the I&E inspections and subsequent follow-up actions by licensees; 2) the conclusions of the Browns Ferry Fire Special Review Group Report (NUREG-0050) that the probability of fires

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[1] It should be noted that the analysis presented in WASH-1400 showed that the potential for a significant release from a severe fire was about 20% of that calculated from all other causes analyzed. This analysis could be interpreted to infer that no further actions were needed for fire protection. This clearly was not the approach recommended by the Special Review Group or adopted by the staff after the Browns Ferry Fire.

of a large and disruptive nature of the magnitude of the Browns Ferry fire is small and that 'there is no need to restrict operation of nuclear power plants for public safety', and 3) improvements made since that time by licensees in fire prevention measures and fire brigade capability and training that have been noted in the plants visited to date and are expected to exist in the remaining plants, which further reduce the probability and consequences of fire."

A similar staff conclusion was expressed in a July 6, 1978, staff memorandum to the Commission on the UCS petition for reconsideration. Page 4[2/] of that memorandum includes the following statements:

"For those plants not yet evaluated, and those plants for which the staff has required enhancement of the fire protection system, the staff believes that the probability of occurrence of severe damaging fires is acceptably low for the interim period until staff evaluations and licensee enhancements are completed. This conclusion is based upon the information discussed by the Browns Ferry Fire Special Review Group in NUREG-0050 and upon

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2/ This page immediately precedes the Table identifying the 12 plants requiring an alternate or dedicated shutdown system which was included as an attachment to the UCS press release.

the additional defense-in-depth protection provided by the staff's overall fire pr'tection upgrading program which provides (1) controls ovedr ignition sources, combustibles and access to the areas, (2) physical separation and use of flame retardants to delay or prevent propagation, and (3) fire detection, fire suppression and trained fire brigades to effect prompt manual suppression of fires."

The Commission's Memorandum and Order on the UCS Petition of April 13, 1978, on pages 36 through 40 discusses the Commission's conclusion concerning the need for immediate action with respect to fire protection of operating plants and the bases for its conclusion that no immediate action is necessary.[3/] The bases discussed in this Order are generally consistent with those advanced by the staff discussed previously, but additional details concerning the view of the Browns Ferry Fire Special Review Group are provided by a quotation from its Report (NUREG-0050). Within the quotation is a paragraph describing the results of the RSS probabilistic assessment of the risks of fires. It is this particular paragraph that apparently provides the basis for the UCS statement in its press release that "the Commission adopted this Reactor Study finding ...." as a basis for allowing these plants to continue to operate.

3/ The UCS press announcement quotes only 1 paragraph of this multi-page discussion, and omits the footnote applicable to the specific portion it did quote.

As previously mentioned, since 1975, there has been a significant improvement at each operating plant in the fire prevention program, and in the capability of plant equipment to detect and extinguish fires promptly. These improvements have provided the primary basis for the staff conclusion that a severe fire in operating plants is not likely. To provide a better understanding of these improvements, we review the accomplishments achieved since the Browns Ferry fire:

(a) As a result of the Office of Inspection and Enforcement's special bulletins to all licensees of operating power reactors on March 24, 1975, and April 3, 1975, directing controls over ignition sources, a review of procedures for alternate shutdown and cooling methods, and a review of flammability of materials used in floor and wall penetration seals, some of the changes and improvements at operating plants are:

(1) modifications of plant administrative procedures for work permits to assure consideration of the safety significance of electrical cables and piping in the work area;

(2) incorporation of the control of combustible materials into plant administrative procedures;

(3) improved plant administrative procedures for the control of ignition sources:

(4) development of new procedures and guidelines covering the use of water on electrical cable;[4/]

(5) study and development of procedures for a variety of means to provide decay heat removal;

(6) addition, upgrading and repair of cable penetration fire stops; and

(7) addition of fire suppression equipment.

(b) As a result of the special inspections by the NRC Office of Inspection and Enforcement completed at all operating power reactors in April and May 1975 covering the installation of fire stops on electrical cables and penetration seals, any inspection findings which reflected noncompliance with then current NRC requirements resulted in prompt corrective action by licensees. Follow-up I&E inspections have confirmed that licensees implemented the required corrective actions and that administrative control procedures were in place.

(c) More detailed procedures for inspection of fire prevention and protection measures have been incorporated in the NRC Operating Reactor Inspection Program. Since September 9, 1975, the Office of Inspection and Enforcement has been conducting detailed annual inspections of licensees' fire protection

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4/ This improvement alone tends to preclude the development of a fire in plant areas in the proportions of the Browns Ferry fire.

programs as one of the routine I&E inspection modules. In addition, a plant tour is conducted quarterly, during which the inspector looks for conditions that might contribute to fires. These inspections include review of fire insurance inspection reports. The results of these inspections show that the licensees' fire protection programs have improved and that the licensees have an increased appreciation for the need for effective fire prevention measures and improved fire-fighting capability. Some of the more significant improvements include:

(1) The licensees have completed sealing the safety-related cable penetrations and have instituted controls to ensure that penetrations are not subsequently degraded.

(2) The licensees have surveyed their plants, identified the possible fire hazards, and are in the process of eliminating or reducing the identified hazard. In cases where the hazard could not be eliminated, additional automatic fire protection systems have been or will be installed. In other cases in which fire protection of safe shutdown systems was uncertain, modifications to provide alternate shutdown methods have been or will be installed or additional fire protection measures taken. Periodic tests of fire protection systems are now being performed.

(3) Fire brigades have been established or enlarged and formally organized with duties and defined responsibilities. Formal training and requalification programs including periodic drills, have been or will be implemented.

(4) Formal agreements with local fire departments have been established and joint participation in some fire drills is now taking place.

(5) Administrative controls have been implemented to limit the use of combustible materials within the plants. The control of ignition sources has also been improved by limiting the use of open flames in the nuclear plant. When open flames are required, such as welding or burning, trained fire watches are used to monitor the operation and to take any necessary corrective action.

(d) Quality assurance inspection procedures have steadily improved over those in effect at the time of preoperational QA inspections of Browns Ferry Units 1 and 2.

(e) As a result of implementing the staff's improved guidelines on administrative controls for fire protection, the licensees' fire protection programs now provide the following:

(1) Control use and storage of combustible materials;

(2) Limit use of ignition source and provide protective measures where ignition sources are used, including fire watches in critical areas;

(3) Establish training programs for fire brigades to include classroom instructions, "hands-on" practice, and periodic drills;

(4) Establish fire-fighting procedures to include notification of operators, offsite fire departments, fire brigade and any other required personnel, and procedures that include fire-fighting strategies for fires in specific safety-related areas.

(f) Technical Specifications for fire protection systems have been incorporated into the license of each plant to assure the operability of fire protection systems. For example, the specifications require the periodic testing, inspection, and surveillance of fire protection systems and equipment (i.e., fire protection instrumentation, fire suppression systems, firehose stations, penetration fire barriers) and require compensatory actions when such equipment is deemed inoperable. These specifications establish the minimum shift strength of the onsite fire brigade for each individual plant. The specifications also require a periodic independent fire protection and loss prevention program inspection and audit that uses either qualified offsite licensee personnel or an outside fire protection firm to assure that the fire protection program is being properly carried out.

All operating plants have been visited by the staff's fire protection review teams, and improvements in fire prevention and fire fighting continue to be made as a result of the ongoing evaluations. During these visits, we have also observed an increased awareness on the part of the utility management and plant personnel that fire prevention and fire-fighting readiness are important elements in the fire protection of their plants. Correction of deficiencies in fire prevention are being made on a schedule that is commensurate with the concern expressed by both the licensees and NRC staff. We have also found that, at every facility reviewed to date, the licensees have established administrative controls that substantially conform to staff guidelines.

In the overall conclusions of our fire protection Safety Evaluation Report (SER's) on individual plants, we have also referenced certain comments made by the Special Review Group report on the Browns Ferry Fire (NUREG-0050). The Special Review Group concluded that there was no need to restrict operation of nuclear plants based on (1) WASH-1400 conclusions that fires contribute negligibly to the overall risk of nuclear plant operation; and (2) the Special Review Group's conclusion that, based on its evaluation of events occurring before, during and after the Browns Ferry Fire, the probability of disruptive fires of the magnitude of the Browns Ferry event is low. The staff's bases for not restricting operation pending completion of our reviews and implementation of all modifications were the recommendations of the Special Review Group as well as the actions already being taken at operating plants to further reduce the likelihood of disruptive fires. These other actions include the administrative controls, the fire brigade staffing and training, and the previously discussed technical specifications. At plants where the staff review found that smaller fires may affect safe shutdown, immediate preventive actions were taken to establish methods to safely shutdown the reactor if such fires were to occur.

The Reactor Safety Study was one element considered by the Special Review Group in making its recommentation; however, the RSS is not a primary element of the staff's basis to allow continued operation pending implementation of all facility fire protection modifications. The RSS calculations did not consider the effects of actions taken in the plants since 1975 and, therefore, are not indicative of the risk of fire in any presently operating plant.

The UCS press release states that fire tests sponsored by NRC at Sandia Laboratories in 1977 and at Underwriters' Laboratories (UL) in September 1978 showed that plant designs meeting current standards do not provide adequate protection against fire. The "current standards" implied by the UCS statement are Regulatory Guide 1.75 and IEEE-384, which deal with separation of redundant safety systems, and IEEE-383, which specifies cable fire-retardant criteria. With regard to these standards, the staff agrees with the UCS that these standards alone do not provide adequate protection against fires. Since the Browns Ferry Fire, the staff has taken a position that sole reliance should not be placed on these standards for fire protection of nuclear power plants.

The staff considers that the Sandia tests confirm the validity of the staff position; namely, that Regulatory Guide 1.75 cable separation criteria and IEEE-383 cable flame-retardant criteria by themselves are not sufficient to protect against exposure to fires and that additional fire protection measures are required. These additional measures include fire barriers, fire-retardant coating on cable, automatic fire detection and extinguishing systems, backup fire extinguishing capability (fire hoses and portable extinguishers), administrative procedures and controls to minimize fire hazards due to poor housekeeping or to plant maintenance activities, and plant fire brigade staffing and training to assure adequate response to fire emergencies. This staff position was taken more than a year before the 1977 Sandia cable fire tests. Thus, the test results confirmed the staff position that additional fire protection measures beyond Regulatory Guide 1.75 and IEEE-383 were necessary as a safe and conservative basis for the plant fire protection evaluation program that is now being implemented.

The UL tests referred to by UCS were generic separate effects tests that did not test a specific fire protection configuration in an operating plant.

Vertical cable trays (unbarriered) have been identified during the course of the operating plant review that are grouped in a manner similar to the tested cable tray configuration. Licensees have proposed various systems of fire protection for vertical cable trays that include the use of fire barriers. The types of barriers proposed to protect redundant divisions of safety-related cables include cable tray covers, ceramic wool blankets with tray covers, insulating board material (Marinite), and fire-retardant coatings. Representative barrier and suppression systems will be, or have been, tested in NRC-sponsored or licensee-sponsored test programs. The particular barrier configuration chosen for the UL test is that currently recommended by the barrier material manufacturer to protect cable trays. The UL test provided data with which the staff can evaluate such barrier systems.

The UCS press release also includes a memorandum from Mr. Cohn of Gage-Babcock dated September 30, 1977. The press release alleges that Gage-Babcock agrees that fire protection is inadequte in many existing plants. The staff response to Mr. Cohn's concern is discussed at length on pages 5, 13 and 14 in the staff response to the Commission dated July 6, 1978, on the subject of the UCS petition for reconsideration. A brief summary of the staff response follows:

(a) At a meeting with the staff on October 20, 1977, subsequent to the September 30 memorandum, Mr. Cohn agreed that no changes in the NRC fire protection guidelines were necessary.

(b) In a letter dated July 1, 1978, Mr. Conn reiterated his position on fire safety in nuclear power plants as follows:

> "It is my belief, based on my knowledge and experience of conditions in the plants I have either personally visited or discussed in depth with our engineers, that sufficient precautions have been taken and that operation can continue in the interim period during which additional measures are implemented to fully meet NRC guidelines."

The Nuclear Regulatory Commission staff reiterates its previous conclusion that the fire protection features in operating plants are adequate to permit operation during the interim period until certain additional fire protection features are installed. This conclusion, previously explained, is primarily based on the many improvements in fire protection systems already accomplished in operating nuclear plants as a result of staff review which began immediately following the 1975 Browns Ferry Fire and does not rely on WASH-1400 results or calculations.

Performance of Electrical Equipment in Safety Systems of Operating Plants

NRC regulations require, in General Design Criterion 4 (10 CFR Part 50 Appendix A), that structures, systems and components important to safety of nuclear power plants be designed to accommodate the effects of the environmental conditions associated with postulated accidents. The staff review procedures and acceptance criteria for the environmental qualification of safety equipment were first developed on a case-by-case basis in the late 1960s and are now contained in the Standard Review Plan issued in 1975 and in several national standards and Regulatory Guides, principally Regulatory Guide 1.89, also issued in 1975. These review procedures and acceptance criteria are used in the review of all new CP and OL applications. Prior to 1975, earlier versions of related national standards were used for CP and OL reviews. Thus, the plants now in operation have been reviewed against detailed acceptance criteria that have changed (in fact, grown more stringent) with time. All plants, however, must meet the same overall requirement of General Design Criterion 3 of having safety equipment qualified for service in an appropriate accident environment.

In November 1977, the UCS filed a petition with the NRC regarding the effects of fires and the validity of environmental qualifications of a certain type of electrical connector used in safety systems.

Actions by the NRC in connection with the petition identified a number of plants which had insufficiently qualified electrical connectors or insufficient documentation of environmental qualifications. The questionable connectors have been either replaced or requalified in all operating plants. In the course of acting on the UCS petition, the staff identified several other types of electrical connections that were used in safety systems and also had questionable qualifications. These too were either replaced or requalified by the operators of the plants in which the equipment existed.

The UCS and the staff have also raised the question of whether this experience with the special class of electrical equipment, namely electrical connections, is indicative of a general inadequacy of environmental qualifications of electrical equipment in safety systems of operating nuclear power plants. The staff, on its own accord and in response to the Commission's April 1978 Memorandum and Order on the UCS Petition, has ongoing activities in the inspection, licensing and research areas to confirm its present judgment that electrical equipment in safety systems of operating nuclear power plants, previously considered to be acceptably qualified, remains acceptable in light of today's knowledge. These actions are described in a number of staff filings with the Commission on the UCS Petition and are publicly available. Two useful summaries are NUREG-0413 published in February 1978 that describes the evolution of environmental qualification criteria, including the IEEE-323, 1971 standard criticized by the staff as referenced by the UCS, and NUREG-0458 published in May 1978 that contains a short term safety assessment of the environmental qualifications of safety-related electrical equipment in eleven of the oldest operating reactors.

The NRC staff has not relied upon the Reactor Safety Study for its conclusion that plant operation could safely continue pending resolution of questions about the continued acceptability of the earlier qualification of electrical qualification of electrical equipment used in safety systems of operating plants.

The basis for the staff conclusion on continued operation is described on pages 34-36 of Appendix B of a staff memorandum to the Commission dated December 15, 1978, in connection with the UCS Petition (later published as NUREG-0413). In reaching a conclusion on the environmental qualification aspects of the petition, the staff stated that:

> "In reaching the judgment that no immediate action is required on operating reactors, the staff, as discussed elsewhere in this report, considered the following:
>
> 1. Nuclear power plants include provisions, such as redundancy and diversity, to cope with equipment failures without affecting the public health and safety.
>
> 2. Operating experience indicates that electrical equipment has performed adequately under both normal operating environmental conditions and on the few occasions where severe environmental conditions have existed.

3. Even the older operating reactors used conservative design and construction practices and many improvements have been made in the area of environmental qualification.

4. A preliminary audit of the environmental qualification of electrical connectors and penetrations in operating reactors has indicated that there is reasonable assurance that this equipment would perform its safety function under accident conditions even though complete documentation is not readily available in all cases. It is the staff's belief that these findings would be essentially the same for other safety-related equipment.

5. The likelihood that essential safety-related equipment or other non-safety equipment would not perform the necessary safety function prior to failure due to environmental reasons coupled with the likelihood of a major accident requiring the performance of this equipment is very low.

6. The regulations have included requirements for environmental qualification and a comprehensive quality assurance program since 1971. The requirement for environmental qualification was included in initial versions of these regulations in the mid-1960s. The NRC compliance effort by the Office of Inspection and Enforcement has emphasized review of environmental qualification test results for safety systems in its routine inspection program."

One part of the six reasons relied on by the staff is that "the likelihood of a major accident requiring the performance of this equipment is very low." This statement was quoted in the UCS press release and apparently provides the only basis for the UCS concern that the Rasmussen Report was relied on for the staff conclusion that no immediate action is required for operating plants because of environmental qualification concerns.

The staff did not rely on the results of the Reactor Safety Study in reaching its overall conclusion nor this specific conclusion on this issue, nor has it conducted a specific, independent analysis of the probability of a major accident in arriving at this judgment. Rather, the staff meant by this statement that in deciding whether immediate action should be taken to further reduce risk to the public, the existing level of protection provided in the facilities to prevent the occurrence or loss-of-coolant accidents was considered. That is, the staff considered the past experience in commercial power reactor operation that is sufficient to demonstrate that the likelihood of such events is low. Data developed from similarly designed high pressure piping systems in other industries is in agreement with this experience. This engineering experience is sufficient to support a judgment by the staff on the low likelihood of occurrence of such an accident, which is, in turn, a part of the overall basis for requiring no immediate action where the technical data also provides reasonable assurance that the safety-related electrical equipment will provide its intended accident mitigating function.

As initially outlined in the staff memorandum of December 15, 1977, and subsequently on July 6, 1978, in Item 11 of Enclosure 1 of another staff memorandum to the Commission concerning the UCS Petition, the scope and timing of staff programs to provide additional confidence that adequate environmental qualification exists for safety equipment in operating plants is based on several factors, including the likelihood of a major accident requiring the performance of this equipment. The degree to which this factor has shaped the staff's actions is difficult to quantify, but other licensing actions taken by the staff serve to illustrate the partial reliance placed upon the low probability of an accident. In cases in which the licensing staff has low confidence that equipment important to safety would function in an accident, plants are required to shutdown and remedy the problem (e.g., see description of staff actions on D. C. Cook Unit 1 in the memorandum to the Commission on November 18, 1977). Such decisions flow from the staff's view that the low probability of an accident within the design basis does not, by itself, provide a sufficient basis to permit continued operation in the face of significant related questions regarding the safety of a plant. That is, the low probability of a severe accident is not considered to be sufficient justification, by itself, to allow continued operation in light of a staff judgment that safety equipment provided to mitigate the consequences of that accident is not likely to function under the accident conditions expected. In other cases, the staff has judged that the available technical information was sufficient to conclude that the equipment was likely to perform adequately or could be demonstrated to be qualified and that reasonable time should therefore be allowed to complete the demonstration of the qualifications.

The ti‹e frame allowing for confirming or further documenting qualifications was chosen, in part, based upon the generally understood low likelihood of occurrence of an accident that would environmentally challenge this equipment during that time frame.

In continuing to recommend that no immediate action need be taken, the staff relies neither on the Reactor Safety Study nor solely on the low likelihood of a major accident, but rather is guided primarily by the judgment, as discussed in Commission memoranda of December 15, 1977 and May 12, 1978, and in NUREG-0458, that the electrical equipment in safety systems of operating plants will not fail before performing its safety function when exposed to expected accident conditions, and there is ongoing work by the staff to confirm this conclusion for these plants.

CLASS 9 ACCIDENTS IN OPERATING PLANTS

NRC regulations require, in Paragraph 50.34 of 10 CFR Part 50, that a determination be made of the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The accidents that are required to be considered under the Commission's regulations are identified on the basis of engineering analyses and a review of operating experience. The basis for the Staff's approach has been set forth in a number of statements and documents including, for example, the Interim General Statement of Policy regarding the Rasmussen Report of August 21, 1974:

> "The Commission's safety regulations set forth a comprehensive three-level approach ... First, nuclear power plants are required to be designed and constructed with a high degree of reliability so that failures or malfunctions that could lead to accidents are highly improbable. An essential part of this first level of safety is the requirement for a comprehensive quality assurance program for plant design, construction, and operation. The second level of safety is the required provision for measures to forestall or cope with incidents and malfunctions that could occur notwithstanding the assurance offered by careful plant design, construction, and operation. For example, plants are required to be equipped with reactor protection systems to terminate the nuclear chain reaction quickly and reliably if plant conditions should require such action, and provision is made for leak detection systems to provide indication of incipient fuel cladding failures or degradation of the reactor coolant system pressure boundary well before leaks become safety problems.
>
> "The third level of safety is unique to nuclear power plants. A series of highly unlikely major failures of plant components is postulated as a set of design basis accidents, and safety systems are required to be installed to control all such postulated events."

and

> "In the approach to safety reflected in the Commission's regulations, postulated accidents, for purposes of analysis, are divided into two categories - 'credible' and 'incredible.' The former ('credible') are considered to be within the category of design basis accidents. Protective measures are required and provided for all those postulated

accidents falling within that category, and proposed sites are evaluated by taking into account the conservatively calculated consequences of a spectrum of severe postulated accidents. Those accidents falling within the 'incredible' category are considered to be so improbable that no such protective measures are required."

There are many accidents which are required to be prevented from occurring and there are many accidents whose consequences must be shown to be acceptable. These are set forth in the staff's Standard Review Plan and in the various Regulatory Guides (see particularly Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Chapter 15 of which lists representative accidents to be considered).

There are a number of criteria related to the prevention of accidents. For example, General Design Criterion 14 requires that the reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Because of the provisions made to prevent a gross failure of the reactor vessel, no special provisions have been required to accommodate the consequences of such an event. The staff has addressed arguments that gross vessel rupture is credible, and shown that its probability is so low as to pose a negligible risk to the public (see, for example, "Technical Report on Analysis of Pressure Vessel Statistics from Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service," WASH-1318, May 1974).

Similarly, the staff has shown that there is an extremely small likelihood of occurrence of a loss-of-coolant accident accompanied by failure of emergency core cooling systems to cool the core to the degree necessary to breach containment (see staff testimony on Callaway, Zion, Prairie Island, Monticello, etc.).

On the other hand, some events have been added to the list required to be considered in the design basis, because of new information that showed that their probability was not acceptably low. The staff has taken and continues to consider such actions. Examples of actions where the need for increased protection is now under consideration are anticipated transients without scram, and station blackout.

On November 1, 1978, the UCS, in providing to the Commission a suggested policy statement on the RSS, stated:

> "...the profound implications for nuclear power plant licensing posed by withdrawal of the RSS. These implications were noted on pages 136-139 of the 1977 UCS review of the final RSS (footnote omitted) and are amplified in this letter."
>
> ****************
>
> "The withdrawal of NRC's endorsement of the RSS and its findings leaves the NRC with no technical basis for concluding that the actual risk is low enough to justify continued plant licensing and operation."

In addition, following issuance of the NRC policy statement on the RSS the UCS press release of January 29, 1979 stated:

> "...with the NRC repudiation of the quantitative probabilities set forth in the Reactor Safety Study, the Commission must reassess all plants to determine what accident scenarios were not examined and determine what additional safety features must be added to adequately protect the health and safety of the public."

The Rasmussen Report has not been a principal basis used by the Staff to determine what events are credible and thus explicitly considered in the design of nuclear power plants.[4/] The staff will continue to use engineering analyses, including statistical analyses as appropriate, to reach reasoned determinations regarding those accidents having severe consequences that are sufficiently likely that they should be considered in the design of nuclear plants.

4/ This point is apparently well known to the UCS in that they stated in the 1977 UCS review cited above: "Yet for all practical purposes, NRC makes essentially no use of the immense body of safety analysis that went into the RSS."

ENCLOSURE 2

Items* Discussed With the Staff

3-1	Beaver Valley Testimony
3-3	Westinghouse Topical Reports
3-4	Control Room Design
3-5	Fire Protection SER's
3-6	Response to UCS Petition for emergency and remedial action
3-8	Passive Valve Failure
3-10	Loss of offsite power post-LOCA with SIS reset
3-11	D.C. Power Reliability
3-12	Steam Line Break - use of non-safety Grade Equipment
3-13	Grid Availability
3-16	CP EIS accident risk discussions
3-17	Response to DES Comments
3-21	TAP-2 (Asymmetric Blowdown Loads)
3-22	Steam Generator SER's
3-23	ECCS Exemption for Dresden
3-24	Haddam Neck Overpressure Protection
3-25	V-Y SER - Hold Down Device
3-26	Big Rock Appendix K Exemption for one cycle operation
3-27	San Onofre SER - ECCS - 6 months exemption
3-28	TAP-2 Fracture Toughness for Vessel Supports
3-29	CRBRP Containment Integrity
3-30	NMP Catwalks
3-38	Diablo Canyon Amendment 52
3-49	Containment Spray Pump Frequencies for Surry
3-52	Seismic Scram
3-54	ECCS Outage Times
3-56	SSE Caused Overpressure at Low Temp.
3-57	Diablo Canyon RHR Valve Failure
3-58	Shoreham Post LOCA loss of offsite power
3-59	BWR Rod Drop Accident

* These items appear to involve at least partial reliance on the RSS in justifying either the status quo or a relaxationof requirements.

3-60	NUREG-0138 - Grid Frequency Decay
3-61	Containment Purging
3-64	PRM 50-19 Evacuated Containments
4-1	Waterhammer TAP A-1
4-5	NUREG-0138 - Reactivity Control System Single Failure
4-10	Rod Drop Accidents for 10 Oldest BWR's
5-40	Mark I Short Term
6-3	10 CFR 11 - Rulemaking on access clearance

ENCLOSURE 3

Enclosure 3

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

FEB 5 1979

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Demetrios L. Basdekas
Experimental Fast Reactor
Safety Research Branch

SUBJECT: REVIEW OF REGULATORY ACTIONS AND STAFF POSITIONS WHICH RELIED ON WASH-1400 RESULTS

I appreciated the opportunity to have met with you last Thursday to discuss your activities regarding the further consideration of items identified and categorized in SECY-78-637 as a result of the subject review.

Your invitation to have me attend as an observer another meeting later that afternoon to hear scheduled presentations on the subject matter was very useful. You were kind enough to ask me to comment to you and your Division Directors on the presentations after they were concluded. In response to your request I made the following points:

- There is more common ground than not between what I perceived to have been the consensus at the meeting and what I believe is a correct interpretation of the Lewis Committee's discussion of WASH-1400 deficiencies and its use in the regulatory process. I reiterated the major points I had raised in recent communications I prepared responsive to the Commission's desire to receive comments on the matter.

- I was not familiar with the details of all the issues presented at the meeting. However, based on my overall impression derived from the twelve or so presentations, there seemed to have been an attempt on the part of those who made the presentations to convey to you their honest perception of the extent to which they had relied on WASH-1400 results in the decision making process in each case. However, I expressed the opinion that this might not have been as successful because "they knew what you [all] wanted to hear."* Therefore, I pointed out that you would have to be mindful in your decision making of the associated uncertainty in your receiving a reasonably realistic assessment of the use of WASH-1400 results in specific regulatory matters.**

- In response to a comment on the Clinch River application I pointed out that heavy reliance was placed on WASH-1400 in drawing up the May 6, 1976 letter to the applicant and NUREG-0139, CRBRP Final Environmental Statement, Section 7.1, involving, among other things, the Reactor Shutdown Systems and the Decay Heat Removal Systems.

- The fact that certain safety issues might have been treated only partially or not at all on the basis of WASH-1400 results, does not necessarily mean that their treatment by the staff was technically sound. (This question and its relation to the safety issues of NUREG-0138, NUREG-0153, and NUREG-0410 was not discussed at the meeting).

After the meeting, and according with your preference; I offered to give you privately my comments on the specific issues presented. You indicated you might want them at some later time. I will be glad to do so at your convenience.

Thank you again for the opportunity to participate in this exchange of views.

Demetrios L. Basdekas
Demetrios L. Basdekas
Experimental Fast Reactor
Safety Research Branch

cc: S. Chilk, SECY
S. Levine, RES
R. Budnitz, RES

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*If there was any doubt in anybody's mind about that it was certainly removed by the frequent and almost explicit reminders offered during the presentations by some senior members of your management team.

**This is closely related to the problem of conflict of interest considerations, when the individual that made a potentially improper use of WASH-1400 results is called to give an assessment of it. Although I strongly believe that this should be done, I also believe that it should be supplemented by some form of independent assessment as I have advocated in my December 26, 1978 memorandum to R. Budnitz.

ENCLOSURE 4

ENCLOSURE 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

JAN 11 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: R. F. Fraley, Executive Director, ACRS

SUBJECT: USE OF WASH-1400 BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Office of the General Counsel has notified this office that the Commissioners have requested information regarding the use of the Reactor Safety Study, WASH-1400 by its advisory committees, boards and panels.

Attached for your information and use is a brief summary of the manner in which the ACRS has been making use of WASH-1400 in its activites.

Members of the Committee have contributed to and the ACRS Chairman has concurred in the attached.

R F Fraley
R. F. Fraley
Executive Director

Attachment:
Applications of WASH-1400 Methodology or Conclusions by ACRS dated 12/11/78

cc:
S. Chilk, SECY, w/att.
W. Shields, OGC, w/att.
M. W. Carbon, ACRS, w/att.

Contact:
R. Fraley, ACRS
4-3265

APPLICATIONS OF WASH-1400 METHODOLOGY OR CONCLUSIONS BY ACRS

1. The Committee observed in its reports of April 8, 1975, July 14, 1976, and December 16, 1976 (attached) that the methodology of WASH-1400 is useful for purposes of identifying important accident sequences and for attempting to develop comparative and quantitative risk assessments for low probability high-consequence accidents. It noted, however, that the methodology cannot guarantee that all major contributors to risk will be identified and a considerable element of judgment is required in assigning many of the input parameters. The Committee concluded that a substantial effort would be required to develop and apply dependable methods for quantitatively accounting for the very large number of multiple correlated or dependent failure paths and to obtain the necessary failure rate data bases.

   WASH-1400 did not cause the Committee to alter its judgment that reactors under construction or in operation do not represent an undue risk to the health and safety of the public nor did it result in any relaxation of ACRS conclusions or practices concerning Reactor Safety.

2. WASH-1400 provided increased insight into containment failure modes following a postulated core melt and provided an improved basis for evaluation of the possibility of Class 9 accidents and the range

of consequences. This reinforced ACRS interest in the Generic Liquid Pathway Study which compared floating and land-based nuclear facilities. It led also to ACRS interest in the possible development of the filtered vented containment as an improved safety system.

3. More recently the Committee endorsed further development of the CRAC code for use in certain site evaluations. CRAC is a computation model developed for, and used in WASH-1400 to evaluate the consequences of serious accidents but does not directly involve the basic fault-tree/event-tree technique, nor the system reliability findings in WASH-1400.

4. Individual Committee members have sometimes used WASH-1400 as a point of departure for questions, comments, or suggestions regarding safety related matters. For example:
   a. It was suggested that the backfit decision-making process would be improved by using WASH-1400 methods to assess the reliability of alternate system designs.
   b. Preliminary comments on a recent staff ATWS study (NUREG-0460) were aimed at making direct comparisons between plant designs and the reliability goals in WASH-1400 (apparently one of the bases of the report) more direct.

5. The ACRS has used some results from WASH-1400 as a partial basis for requesting further evaluation of the adequacy of specific systems. For example, the ACRS has had a long-time interest in the capability of plants to survive safely a considerable loss of all AC power for an extended period. WASH-1400 indicated that the probability of a loss of all AC power was nonnegligible. The ACRS has asked the NRC staff for a comprehensive evaluation of the matter, including the possible need for design modifications. With the availability of WASH-1400 methodology and data, the ACRS was able to request an NRC Staff evaluation of the adequacy of the reliability of auxiliary feedwater and other systems of current design.

6. The consequence studies in WASH-1400 provided additional background information for ACRS consideration of emergency preparations.

Attachments:
1. Ltr. to W. A. Anders dtd. 4/8/75
2. Ltr. to M. K. Udall dtd. 7/14/76
3. Ltr. to M. K. Udall dtd. 12/16/76

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 14, 1976

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, DC 20515

Dear Congressman Udall:

At its 195th meeting on July 8-10, 1976, the Advisory Committee on Reactor Safeguards (ACRS) considered the points raised in your June 14, 1976, letter on the Reactor Safety Study (RSS, WASH-1400, NUREG 75/014). The ACRS reviewed the draft version of the Reactor Safety Study in late 1974 and early 1975 and submitted a report to the Nuclear Regulatory Commission on April 8, 1975. A copy of the ACRS report is attached.

Your letter identified eleven issues on which you requested comment and the Committee is pleased to respond to issues 1, 3, 4, 6, 8, 9 and 10. However, extensive time and effort would be required by the ACRS to respond adequately to the other topics and the needed effort would have to be factored into overall considerations of other ACRS functions, including mandatory review of applications for construction permits and operating licenses for commercial nuclear power plants.

The Committee's responses follow:

1. "The extent that the NUREG 75/014 fault-tree analysis adds to understanding of the likelihood of major nuclear reactor accidents."

The ACRS believes that the fault-tree methodology used in the Reactor Safety Study to develop comparative and quantitative risk assessments for postulated accident sequences represents a valuable contribution to the understanding of the likelihood of major nuclear reactor accidents.

3. "Adequacy of data base for NUREG 75/014 type fault-tree analysis."

As noted in our report of April 8, 1975, the ACRS believes that a better data base will be required to evaluate the validity of the RSS's quantitative estimates of the likelihood of low probability high consequence events, and recommends that current efforts to compile, categorize and evaluate nuclear and other applicable industrial experience be extended in breadth and depth to improve the data base for further studies of this type.

4. "Sensitivity of NUREG 75/014 conclusions to differences in reactor design, in site characteristics, in local meteorological conditions and in population distributions."

All of the factors noted above will have some effect on the probability or consequences of a serious accident. The Committee has recommended that the methodology of the Study be applied to other types and designs of reactors, other site conditions and other accident initiators and sequences. If this is done, it will provide greater insight into the sensitivity of differing reactor designs and safety features.

6. "Adequacy of NUREG 75/014 methodology to take account of gradual degradation of plant safety over plant lifetime."

The Committee believes the methodology is capable of taking into account wear out of components and degradation of equipment over the lifetime of the plant but an appropriate data base needs to be developed.

8. "Need for periodic updating of NUREG 75/014 to take account of new data."

The Committee believes that a continuing effort is desirable in the application of the methodology developed by the Reactor Safety Study not only to factor in new data but also to consider design variations and new concepts.

9. "Need for continuing analysis of NUREG 75/014 for purposes of delineating areas of research and data collection."

The Committee believes that the NUREG 75/014 methodology should be used to aid in delineating areas for further research. Special emphasis should be given to quantification of the initiators, probabilities, and consequences of core melting.

10. "The extent to which NUREG 75/014 can be used to aid development of regulatory policies concerning design, construction, and operations."

The Committee has recommended to the NRC that many of the techniques used in the Study can and should be used by the reactor designers to improve safety and by the NRC Staff as a supplement to their safety assessment.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachment:
Ltr. to Hon. W. Anders from D. W. Moeller, dtd 4/8/75 re: WASH-1400

NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 16, 1976

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, DC 20515

Dear Congressman Udall:

At its 200th meeting, December 9-11, 1976, the Advisory Committee on Reactor Safeguards (ACRS) continued its consideration of the points raised in your June 14, 1976, letter on the Reactor Safety Study (RSS, WASH-1400, NUREG 75/014). The ACRS had previously considered these matters at its 196th and 199th meetings and had responded to issues 1, 3, 4, 6, 8, 9 and 10 in its letter to you dated July 14, 1976. In its further consideration of the remaining four issues, the Committee had the benefit of meetings of its Reactor Safety Study Working Group with the Nuclear Regulatory Commission Staff in Washington, DC, on October 12, 1976, and November 10, 1976.

The ACRS is continuing to evaluate the considerable body of information presented in the RSS report, its appendices, and the comments received on it, giving primary attention to the potential implications of the report for the reactor licensing process. This letter provides the Committee on Interior and Insular Affairs a brief resume of current ACRS thought on issues 2, 5, 7 and 11.

"2. Adequacy and appropriateness of analysis used in NUREG 75/014 for purposes of estimating the likelihood of low probability, high consequence events."

The ACRS believes that the methodology of NUREG 75/014 is useful for purposes of identifying important accident sequences and for attempting to develop comparative and quantitative risk assessments for low probability, high-consequence accidents. However, the ACRS believes that considerable effort by more than a single group over an extended period of time will be required to evaluate the validity of the results in NUREG 75/014 in absolute terms. Among the matters which will warrant emphasis in such an evaluation are the following: improved quantification of accident initiators; the identification and evaluation of atypical reactors; the influence of design errors; improved quantification of the role of operator errors; improved quantification of consequence modeling; and the development of improved data for systems, components and instruments under normal and accident-related environmental conditions in a nuclear reactor.

The ACRS believes that NUREG 75/014 represents a very considerable contribution to the understanding of reactor safety and provides a point of departure for quantitative assessment.

"5. Adequacy of NUREG 75/014 methodology to take account of multiple, correlated errors in procedures, design, judgment, and construction such as those leading to the Browns Ferry fire."

The ACRS believes that the methodology of NUREG 75/014 is useful in accounting for that portion of the risk resulting from identifiable potential common mode or dependent failures, and can be used to search out the possibility of multiple correlated errors. However, the methodology cannot guarantee that all major contributors to risk will be identified, and a considerable element of subjective judgment is involved in assigning many of the quantitative input parameters. Both for nuclear and non-nuclear applications, for complex systems, where multiple, correlated failures or common cause failures may be significant, the record shows that investigators working independently will frequently make estimates of system unreliability which differ from one another by a large factor. At this stage of its review, the ACRS believes that a substantial effort may be required to develop and apply dependable methods for quantitatively accounting for the very large number of multiple correlated or dependent failure paths and to obtain the necessary failure rate data bases.

Whether multiple, correlated errors will dominate the overall risk, however, is subject to question, particularly if simpler postulated accident sequences are generally the dominant contributors to the likelihood of system failure.

"7. Extent to which the final version of NUREG 75/014 takes into account comments on the draft version."

The ACRS is in the process of reviewing the disposition of selected comments received by the Reactor Safety Study Group, particularly as they have implications for short or long-term improvements in reactor safety. The ACRS plans to continue this type of activity; however, it is beyond the scope or available working time of the ACRS to review in detail the extent to which the final version of NUREG 75/014 takes into account the comments received.

"11. Validity of NUREG 75/014 conclusions regarding accident consequences."

As stated in its report to you of July 14, 1976 and as indicated in its response to other questions in this group, the ACRS believes that considerably more effort on the part of various contributors is needed to evaluate the quantitative validity of NUREG 75/014 conclusions regarding accident consequences. Based on information currently available, the ACRS would assign a greater uncertainty to the results than that given in NUREG 75/014.

The ACRS believes that the past and current practice of trying both to make accidents very improbable and to provide means to cope with or ameliorate the effects of accidents has been the correct approach to nuclear reactor safety.

The ACRS review of the Reactor Safety Study has not caused the ACRS to alter its judgment that operation of reactors now under construction or in operation does not represent an undue risk to the health and safety of the public. The ACRS believes that NUREG 75/014 has suggested many fruitful areas for study and evaluation for potential improvements in light water power reactor safety. The ACRS also believes that the extension of such risk assessment methodology to the total spectrum of activities involved in the production of nuclear power and in the production of electric power by other means, as well as to other technological aspects of society, could add significantly to our overall understanding of risk.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 8, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: REACTOR SAFETY STUDY, WASH-1400

Since the release of the draft Reactor Safety Study, WASH-1400 (RSS) in August 1974, the Advisory Committee on Reactor Safeguards has been reviewing the considerable body of information presented in the report, its appendices, and the comments received on it, giving primary attention to the potential implications of the draft report on the reactor licensing process. In its review, the Committee has had the benefit of Subcommittee meetings held on October 9, November 22, and December 20, 1974, and March 5, 1975, and of full Committee meetings held on October 10-12, October 31-November 2, November 14-16, December 5-7, 1974, and January 9-11, February 6-8, March 6-8, April 3-5, 1975.

The ACRS believes that the RSS represents a valuable contribution to the understanding of light water reactor safety in its categorization of hypothetical accidents, identification of potential weak links for the two reactors studied, and its efforts to develop comparative and quantitative risk assessments for accident sequences examined. The Committee believes that a continuing effort and better data will be required to evaluate the validity of the quantitative results in absolute terms. Special emphasis should be given to quantification of the initiators, probabilities, and consequences of core melting.

The Committee believes that the methodology of the RSS should be applied to other types and designs of reactors, other site conditions and other accident initiators and sequences, and that the current efforts to compile, categorize, and evaluate nuclear experience should be extended in breadth and depth to improve the data base for future studies of this type.

The Committee believes, further, that the RSS can serve as a model for similar studies of the failure probabilities, consequences, and resulting risks of other hazards (both nuclear and non-nuclear) to the health and safety of the public.

The Committee believes that many of the techniques used in the RSS can and should be used by reactor designers to improve safety and by the NRC Staff as a supplement to safety assessment.

COPY

The Committee's review of the RSS has not caused the Committee to alter its judgement that reactors now under construction or in operation do not represent undue risks to the health and safety of the public.

The Committee will continue to review the RSS and will comment further on it in the future.

Sincerely,

Original Signed by:
W. Kerr

William Kerr

COPY

February 2, 1979

SECY-79-94

UNITED STATES
NUCLEAR REGULATORY COMMISSION

INFORMATION REPORT

D. ROSS
[redacted]
~~W. GAMMILL~~
~~D. ...ULT~~
~~L. CROCKER~~
~~H. BERKOW~~
L. Miller

For: The Commissioners

From: Saul Levine, Director
Office of Nuclear Regulatory
Research

Thru: Lee V. Gossick
Executive Director for Operations

Subject: The Status and Direction of Risk Assessment Research at the NRC

Purpose: To respond to the request by the Commissioners on January 25, 1979, for additional information on NRC's Risk Assessment Research Program*

Discussion: Quantitative risk assessment techniques can be used to identify the relative importance of various contributors to potential accident risks from the elements of the nuclear fuel cycle. This knowledge can be used to assist regulatory decisionmaking by identifying and understanding significant safety issues associated with these contributors. The objective of the risk research being performed by the Probabilistic Analysis Staff in RES is to support the regulatory decisionmaking process by application of existing tools and data as appropriate to current issues, by performing risk assessments, and by improving the existing situation in both data and methodology for future applications of risk assessment.

Risk assessment techniques are already being used in a limited way in the NRC regulatory process to help focus the relative importance of certain issues. The significant questions for planning future research are: (1) Should the use of risk assessment techniques continue to increase? and (2) How can their usefulness be improved? The Risk Assessment Review Group in NUREG/CR-0400 answered the first question affirmatively and provided substantial guidance on improving the usefulness of these techniques.

The Review Group Report recommended that fault tree/event tree analyses should be among the principal means used to deal with generic safety issues, to formulate new regulatory requirements,

Contact:
A. R. Buhl, RES
49-28528

SECY NOTE: Reference SECY Staff Requirements Memorandum to EDO, subject: "Briefin on Final Program Plan for the NRC FY 79 Budget, etc., dated January 29 1979.

to assess and revalidate existing regulatory requirements and to evaluate new designs. The Group's Report also calls for use of the methodology in guiding allocation of resources within NRC's research program. The report further points out that the NRC can make the licensing and regulatory process more rational by better matching its resources to those items which influence risk. Our research program has already begun to respond to these recommendations; indeed, many of our projects were actually underway before the Review Group's report was published.

The following sections in this document outline:

- the current risk assessment research program within PAS;
- the reprogramming of funds required for increased support of reactor licensing;
- increasing efforts to support NMSS;
- the impact of responding to the specific recommendations of the Review Group's Report;
- the improved safety program; and
- preparation for an update of the RSS beginning in FY 1982.

Risk Assessment Research Program

Risk Assessment research is organized into four major programs; Methodology Development, Data Analysis, Applications Research and Licensing Support. Each program is structured into the areas of research as shown in Figure 1. The major projects and their objectives are listed in Appendix A to this report. The ACRS reviewed these projects in great detail and their conclusions are enclosed as Appendix B. Table 1 provides a breakdown of the PAS budget for FY 1978 through FY 1982.

The Methodology Development Program is structured to permit more accurate assessments of systems performance and to develop methodology for evaluating problems which are beyond current capabilities. Development efforts are underway both within PAS and with contractor support to realistically assess the risk to nuclear facilities from fires and floods. These efforts involve characterizing the statistical behavior of fires and floods as initiating accident events and developing systems models to evaluate failure probabilities within the nuclear facilities. As part of the work in this program, common cause and common mode failure models are being developed to systematically evaluate common cause/common mode failure probabilities utilizing accepted statistical approaches. Other major efforts in the Methodology Development Program include propagation of uncertainties through complex risk models, updating the consequence model of the Reactor Safety Study, improvements in liquid pathway models and development of models for assessing the risk from disposal of radioactive waste.

The Data Analysis Program is structured to improve the data base and develop techniques to analyze these data. Efforts are underway to establish systematic methods of quantifying human reliability. Human factors analysis techniques are being codified for use in risk evaluations and plans are being made to use nuclear reactor simulators to collect human error data. Data analysis techniques are being used to analyze human errors, component failures, and system failures which have occurred and which are reported through the NPRDS or LER systems. Other major efforts in the Data Analysis Program include developing statistical techniques for analyzing reported common cause failures, determining uncertainties based on experience, and extracting time trends (e.g., wear-outs) from data.

The Applications Research Program is designed to apply existing quantitative risk assessment techniques to special regulatory areas to extend the utility of these techniques. For example, the methodology is being applied to examine the effects of design differences on risk between the two plants (Surry and Peach Bottom) examined by the RSS and four new commercial plants (Westinghouse Ice Condenser, B&W Dry Containment, CE Dry Containment, and GE BWR 6/MARK III). Other important elements of the Applications Program include an examination of the risk from Class 3-8 events, analyses of sensitivity of risk from core meltdown sequences to several important physical phenomena, emergency response planning, development of risk criteria, and an active training program conducted for the agency in the utility of risk techniques.

Licensing Support

Direct Licensing Support is provided on a priority basis to the Office of Nuclear Reactor Regulation. This support includes efforts to categorize the standard review plan according to safety importance from a risk perspective, ranking the generic safety issues based on risk significance, evaluating technical specifications, evaluating certain of the RRRC decision items as requested and performing the work required to resolve TAP A-30 (DC Power). This work was requested by NRR and provides direct support to the licensing decisionmaking process. RES is requesting reprogramming of \$1,400K in FY 1979 from RSR to fund licensing support. This \$1,400K provides \$1,000K to respond to new initiatives requested by NRR and \$400K to continue licensing support initiated in previous years. Budget requirements to continue these efforts are given in Table I.

Assistance to the Office of Nuclear Materials Safety and Safeguards

Efforts have been initiated over the past several years to support NMSS in the development of regulatory basis for high level radioactive waste management. This effort has been focused primarily on the development of analytical models and techniques for assessing the performance of a waste repository in deep geologic formations. The models appear promising as useful tools for identifying important contributors to risk and for supporting licensing. However, substantial improvements are needed in both the available models and the data base before we can have adequate confidence in their use.

PAS expects to devote over \$1,000K of its resource annually to support activities of direct benefit to NMSS. These funds are included in the FY 1980 Budget request to Congress.

Impact of Review Group's Report on Risk Research Program

Probabilistic Analysis Staff members appeared before the Risk Assessment Review Group during the past year and identified many of the deficiencies in the Reactor Safety Study which, along with others, were noted in the Review Group Report Work has already been initiated to correct these deficiencies. PAS is implementing many of the recommendations made in the report and has developed plans to modify existing programs and add new programs to respond to concerns raised by the Review Group Report.

Budget requirements to respond to the report are given in Table II. A detailed budget breakdown is given in Table III for each program addition or modification. Major efforts are underway to improve the data base by accelerating efforts to collect and evaluate all component failure rate data. Major efforts are underway to develop models and evaluate data on the effects of floods, fires and common cause failures. Analysis of human errors is being accelerated. The consequence model is being updated to incorporate more realistic dispersion characteristics and biological effects. Training efforts are being stepped up. Programs are being initiated on piping failure data analysis. Risk techniques are being developed to place priorities on other research programs.

Research to Improve the Safety of Light-Water Reactors

The Fiscal Year 1978 Budget Authorization Act for the NRC modified Section 205 of the Energy Reorganization Act to require that the NRC prepare a long-term plan for the development of new or improved safety systems for nuclear power plants. In April 1978 the Office of Nuclear Regulatory Research completed and the Commission submitted to Congress a "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants" (NUREG-0438). This report presented an evaluation of concepts proposed to improve safety and recommended a three-year, $14.9 million research program. The objectives are to determine the feasibility of achieving particular improvements in safety, to evaluate the safety significance of proposed changes and to propose regulatory requirements where implementation is determined to be desirable, without preparing detailed designs.

NUREG-0438 recommended that the following seven research topics be pursued: (a) alternate containment concepts, (b) alternate decay heat removal concepts, (c) alternate emergency core cooling concepts, (d) improved human interaction, (e) advanced seismic designs, (f) scoping studies of other concepts, and (g) improved evaluation methods.

The Fiscal Year 1979 Budget Authorization Act for NRC authorized $1,500,000 to implement the plan. Matching appropriations were not provided. A total of $800,000 was requested by reprogramming $400,000 from FY 1979 RES safety [illegible] research and $400,000 from FY 1978 unobligated carry-over. These funds will be used by the Probabilistic Analysis Staff to initiate research on alternate containment concepts, on alternate decay heat removal concepts and on improvement of methods to assess the values and impacts of proposed concepts. PAS will direct future research on human interaction and scoping studies, while RSR will primarily support any activities associated with alternate ECCS and advanced seismic design. The funds necessary to fully implement the research described in NUREG-0438 are shown in Table IV.

In response to the Congressional initiative, the Commission has expressed its belief that extension of its charter into research on the development of new or improved safety systems is very useful. It will permit the exploration and evaluation of the many suggestions that have been made for improving safety of nuclear power plants and may indeed lead to improvements in their safety.

Preparation for RSS II Update

Much of the ongoing work within PAS and the additional efforts responding to the Risk Assessment Review Group Report must be completed before a meaningful update to the RSS could be produced. In particular, the available data base has increased more than tenfold since the original RSS analysis and improvements are underway to calculate the risk from external events (floods, fires, seismic), human errors, and common cause failures. The available data should be thoroughly analyzed and the newly developed methodology well tested before attempting to update or perform a second Reactor Safety Study.

It appears that based on the amount of work to be done internally and the agency's difficulties in contracting, that from two to four years of work should be done prior to attempting an update. If an urgency develops, some of the required preliminary efforts could be accelerated by additional resources.

Summary

The original FY 1979 Risk Research budget request to OMB before the additional licensing support or increased efforts to respond to the Risk Assessment Review Group were identified was $3,400K and was included in the President's budget. The LWR improved safety research report was sent to Congress in April 1978 and the FY 1979 Authorization Act authorized $1,500K for this effort. The Senate Appropriations Committee reduced the $3,400K for Risk Research to $3,000K and did not specify funding for the LWR improved safety program.

Reprogramming has been requested of the Commission for $1,400K to reinstate the $400K reduction by the Appropriations Committee and allow $1,000K for the direct licensing support effort.

The $800K required to initiate the LWR improved safety program is being made available from two sources. The Commission has approved the use of $400K of FY 1978 unobligated balance and the remaining $400K is being requested as part of the FY 1979 reprogramming action. The FY 1980 LWR improved safety program request to OMB was $4,300K and after reclama OMB changed the budget from $0 to $1,000K. With this FY 1980 reduction, the LWR Improved Safety Program cannot be implemented consistent with our report to Congress.

After planning is finalized, funding necessary to respond to the Review Group Report will be identified and appropriate requests for this effort will be made in the near future.

Robert Budnitz

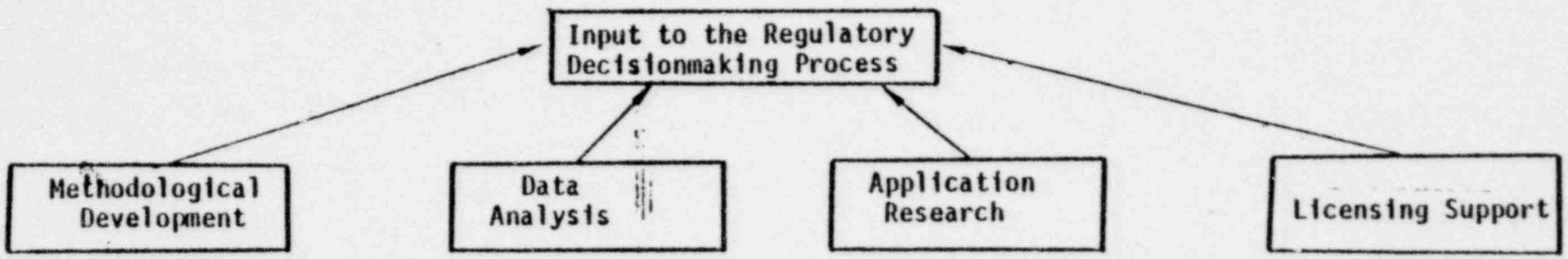
for Saul Levine, Director
Office of Nuclear Regulatory
Research

Enclosures:
1. Figure 1
2. Table I
3. Table II
4. Table III
5. Table IV
6. Appendix A
7. Appendix B

ENCLOSURE 1

FIGURE 1

Figure 1 - Overview of PAS's Risk Assessment Research



Fire Risk System Analysis	Plant Data Analysis	Core Meltdown/Heat Removal	Task Action Plans
Consequence Modeling	LER Statistical Trend Evaluation	RSS Methodology Appl.	RRRC Items
Fire Risk in Nuclear Plants	Human Error Rate	Hazards to Nuclear Plants	Standard Review Plan
Modular Approach to Fault Trees	LER Failure Rate	Acceptable Level of Risk Criteria	Changes to Tech Specs
Flood Methodology Dev.	Handbook of Human Reliability	Class 3-8 Accidents	Support to Systematic Evaluation Program
Liquid Pathways	Limiting Conditions for Operation		Tech Spec Review
Fuel Reprocessing			
Waste Management of Gases			
Dynamic Simulation of Waste/Rock Processes			
HLW Disposal Methodology			
Spent Fuel Isolation Alternatives			

Enclosure 1

ENCLOSURE 2

Table I

Table I - PAS Budget (Risk Assessment)* (thousands of dollars)

	FY78	FY79		FY80	FY81	FY82
Methodological Development	1492	1163		2000	2910	2550
Data Analysis	148	755	3400**	1778	1388	1478
Applications	1760	1482		1400	1102	1447
Direct Licensing Support	0	1000***		522	300	225
PAS Total	3400	4400		5700	5700	5700

*Does not include funds for Improved LWR Safety.

**This $3,400K is the $3,000K appropriated funds plus $400K being sought through reprogramming.

***This $1,000K is being sought through reprogramming.

ENCLOSURE 3

Table II

Table II - Impacts on PAS's Budget (thousands of dollars)

	FY79	FY80	FY81	FY82
Review Group Recommendations	940	1620	1180	580
Improved LWR Safety*	800**	1000	2900	2000
Reactor Safety Study Update	0	0	200	1300
TOTAL	1740	2620	4280	3880

*Total RES funds required to implement program in NUREG-0438 is $4300K in FY80, $4900K in FY81, and $3900K in FY 82.

**$400K has been approved from FY 1978 unobligated balance and $400K being sought through reprogramming.

ENCLOSURE 4

Table III

TABLE III: NEW PROGRAMS RESULTING FROM LEWIS REPORT -- K $

		FY '79	FY '80	FY '81	FY '82
1A.	FLOOD SYSTEMS ANALYSIS	200	100	50	
1B.	AUTO FAULT TREE ANALYSIS (SETS)	80	80		
1C.	HUMAN FACTORS SIMULATION DATA	50	70	100	70
1D.	COMMON CAUSE STATISTICAL MODELING	50	50		
1E.	PIPING FAILURE ANALYSIS	50	100	150	100
1F.	METHODOLOGY DEVELOPMENT	40	100	150	70
2.	HUMAN ERROR	70	150	150	30
3.	CONSEQUENCE MODELING	100	150	100	50
4.	TRAINING	20	80	60	20
5A.	INSPECTION EXPOSURE VS PUBLIC	30	150	100	40
5B.	LER INSPECTION FAILURES	30	50	80	100
6.	RESEARCH PRIORITIES	100	150	100	50
7A.	LWR SYSTEMS SURVEY	50	100	50	
7B.	ACCIDENT SEQUENCE PRECURSERS	70	210	50	50
8	PROBABILITY UNCERTAINTIES		80	40	
	totals	940	1620	1180	580

Enclosure 4

ENCLOSURE 5

Table IV

Table IV - Program Support Funds Necessary to Implement Fully the Research on Improved Safety Described in NUREG-0438 ($M)

FISCAL YEAR	PAS	RSR	TOTAL
1979	1.5	0.0	1.5
1980	1.8	2.5	4.3
1981	2.9	2.0	4.9
1982	2.0	1.9	3.9

ENCLOSURE 6

Appendix A

APPENDIX A

METHODOLOGY DEVELOPMENT

- CONSEQUENCE MODELING - NUC. REACTOR ACCIDENTS

  PROJECT OBJECTIVE:

  TO EXTEND AND IMPROVE THE CALCULATION OF REACTOR ACCIDENT CONSEQUENCE MODEL (CRAC) OF THE REACTOR SAFETY STUDY (WASH-1400) FOR SITE SPECIFIC PURPOSES. TO INVESTIGATE THE SENSITIVITIES OF THE VARIOUS PARAMETERS AND MODELS TO THE END RESULTS SO THAT ERROR BOUNDS CAN BE ASSOCIATED WITH THE RESULTS.

- FIRE RISK SYSTEMS ANALYSIS

  PROJECT OBJECTIVE:

  ANALYZE MORE THAN 300 REPORTED FIRE INCIDENTS TO OBTAIN PROBABILISTIC CHARACTERS OF FIRES, TO BE USED IN RISK EVALUATIONS OF NUCLEAR POWER PLANTS

- FIRE RISK IN NUCLEAR POWER PLANTS

  PROJECT OBJECTIVE:

  PROVIDE A PROBABILISTIC ASSESSMENT OF FIRE INITIATION, FIRE SEVERITY, AND FIRE EFFECTS ON SYSTEMS AND OPERATIONS

- MODULAR APPROACH TO FAULT TREE ANALYSIS

  PROJECT OBJECTIVE:

  1. TO EXTEND THE CURRENT VERSION OF THE PL-MOD FAULT TREE ANALYSIS CODE SO THAT IT CAN BE MORE EFFECTIVELY USED FOR THE QUALITATIVE AND QUANTITATIVE ANALYSIS OF NUCLEAR SAFETY SYSTEMS.
  2. TO ASSESS AND DEMONSTRATE THE CAPABILITIES OF MODULAR FAULT TREE ANALYSIS BY COMPARING ITS RESULTS WITH THOSE OF OTHER FAULT TREE CODES.

Enclosure 6

METHODOLOGY DEVELOPMENT (C O N T I N U E D)

- ANALYSIS AND PREDICTION OF MAJOR FLOODS

  PROJECT OBJECTIVE:

  INVESTIGATE AND DEVELOP PROBABILISTIC APPROACHES WHICH PREDICT THE PROBABILITIES AND MAGNITUDES OF OCCURRING FLOODS. IN ADDITION, NRR HAS REQUESTED THAT QUANTITATIVE MODELS FOR FLOOD PROBABILITY PREDICTIONS BE INVESTIGATED. THE MODELS DEVELOPED CAN POTENTIALLY BE USED FOR WASH-1400 EXTENSIONS AND UPDATES.

- Effect of Liquid Pathways on RSS Consequence Calculations

  PROGRAM OBJECTIVES:

  To establish the degree of radiation exposure to the population occurring from transport of radionuclides through liquid pathways after a core melt-down accident.

- FUEL PROCESSING METHODOLOGY DEVELOPMENT

  PROJECT OBJECTIVE:

  TO DEVELOP AND APPLY RISK ASSESSMENT METHODOLOGIES TO THE ANALYSIS OF THE CHEMICAL PROCESSING OF NUCLEAR FUEL; TO GAIN INSIGHTS INTO IMPORTANT CONTRIBUTORS TO RISK ASSOCIATED WITH CHEMICAL PROCESSING OF FUEL

- WASTE MANAGEMENT OF RADIOACTIVE GASES

  PROJECT OBJECTIVE:

  DEVELOP METHODOLOGY FOR ASSESSING RISK ASSOCIATED WITH ALTERNATIVES FOR MANAGEMENT OF C14, I129 AND KR85 GASEOUS WASTES FROM FUEL CYCLE FACILITIES; COST/BENEFIT ANALYSES TO BE PERFORMED; SENSITIVITY ANALYSES TO BE PERFORMED; FACILITY PERFORMANCE CRITERIA RECOMMENDED.

[redacted]

METHODOLOGY DEVELOPMENT (C O N T I N U E D)

• DYNAMIC SIMULATION OF WASTE/ROCK PROCESSES

PROJECT OBJECTIVE:

TO MODEL AND ANALYZE DYNAMIC GEOLOGIC PROCESSES WHICH RESULT IN FEEDBACK MECHANISMS.

• HIGH LEVEL WASTE DISPOSAL RA METHODOLOGY

PROJECT OBJECTIVE:

DEVELOP METHODOLOGY FOR ASSESSING RISK ASSOCIATED WITH HIGH LEVEL WASTE ISOLATION IN DEEP GEOLOGIC MEDIA (INITIAL EMPHASIS ON BEDDED SALT.) PERFORM SENSITIVITY ANALYSES TO GAIN INSIGHT INTO CRITICAL PARAMETERS/CHARACTERISTICS AFFECTING RISK

• SPENT FUEL ISOLATION ALTERNATIVES

PROJECT OBJECTIVE:

DEVELOP METHODOLOGY FOR ASSESSING RISK ASSOCIATED WITH SPENT FUEL ISOLATION IN DEEP GEOLOGIC MEDIA

[redacted]

DATA ANALYSIS

- PLANT DATA ANALYSIS

  PROJECT OBJECTIVE:

  TO EXPAND CURRENT DATA BASE FOR THE RELIABILITY DATA MANUAL TO INCLUDE MORE DATA FROM OPERATING PLANTS.
  TO COLLECT AND ANALYZE MAINTENANCE AND TEST DOWNTIME INFORMATION FOR INCLUSION IN THE NEXT EDITION OF THE RELIABILITY DATA MANUAL.

- LER STATISTICAL TREND EVALUATION

  PROJECT OBJECTIVE:

  INVESTIGATE THE CAPABILITIES AND LIMITATIONS OF BOTH THE BAYESIAN AND CLASSICAL STATISTICAL APPROACHES FOR OBTAINING THE PROBABILITY DISTRIBUTION FUNCTION OF VARIOUS COMPONENT FAILURES FROM MULTI-PLANT FAILURE DATA.

- HUMAN ERROR RATE ANALYSIS

  PROJECT OBJECTIVE:

  1. TO EVALUATE OPERATOR ERROR RATES BASED ON ACTUAL COMMERCIAL NUCLEAR POWER PLANT DATA
  2. QUANTIFY AND MODEL HUMAN PERFORMANCE FOR A RANGE OF OPERATIONAL CONDITIONS

- LER FAILURE RATE DETERMINATION

  PROJECT OBJECTIVE:

  DETERMINE FAILURE RATES FOR NUCLEAR PLANT COMPONENTS USING THE CURRENTLY AVAILABLE NRC LICENSEE EVENT REPORT (LER) FILE. DEVELOP AND USE COMMON CAUSE ANALYSES OF LER'S. PERFORM STATISTICAL ANALYSIS OF LER AND NPRDS DATA AND ASSESS POSSIBILITY OF MERGING THESE SYSTEMS.

- Handbook of Human Reliability Analysis for Nuclear Power Plant Operations

  PROGRAM OBJECTIVES:

  To prepare a Human Factors Handbook which can be used for the evaluation of engineered safety systems in nuclear power plants.

APPLICATION

- CORE MELTDOWN/HEAT REMOVAL & FP REL.

  PROJECT OBJECTIVE:

  SUBTASK 1: INVESTIGATE THE EFFECTS OF LWR PLANT DESIGN VARIATIONS ON THE RISKS ASSOCIATED WITH REACTOR MELTDOWN ACCIDENTS. SPECIFICALLY, DETERMINE THE EFFECTS OF PLANT DESIGN VARIATIONS ON THE PROBABILITY AND NATURE OF THE RADIONUCLIDE SOURCE TERM RELEASED DURING KEY MELTDOWN ACCIDENT SEQUENCES.

  SUBTASK 2: 1. DETERMINE THE AREAS OF GREATEST UNCERTAINTY IN THE CALCULATION OF MELTDOWN ACCIDENT RADIONUCLIDE RELEASES
  2. IDENTIFY PRIORITY AREAS FOR REACTOR MELTDOWN ACCIDENT RESEARCH WHICH WOULD REDUCE THESE UNCERTAINTIES IN ACCIDENT RADIONUCLIDE RELEASES
  3. EVALUATE THE EFFECT OF METHODOLOGY ASSUMPTIONS ON THE IMPORTANCE RANKING OF VARIABLES.

- SYSTEM ANALYSIS, RSS METHODOLOGY APPLICATIONS PROGRAM

  PROJECT OBJECTIVE:

  TO DETERMINE THE ACCIDENT SEQUENCES DOMINATING RISK FOR LWR PLANTS WHICH ARE REPRESENTATIVE OF THE CURRENT NUCLEAR INDUSTRY.

- HAZARDS TO NUCLEAR POWER PLANTS

  PROJECT OBJECTIVE:

  TO DEVELOP A METHODOLOGY FOR EVALUATING RISKS TO NUCLEAR POWER PLANTS FROM POTENTIAL NEARBY TRANSPORTATION ACCIDENTS INVOLVING HAZARDOUS BUT NON-NUCLEAR CARGO.

- ACCEPTABLE LEVEL OF RISK CRITERIA FOR NUC. POWER PLANTS

  PROJECT OBJECTIVE:

  (1) FORMULATION OF NEEDED RESEARCH NECESSARY TO SET ACCEPTABLE RISK CRITERIA FOR USE BY DECISION MAKERS AND REGULATORY GROUPS WITHIN NRC.
  (2) COMPARISON OF THE RISKS FROM OCCUPATIONAL, ROUTINE AND ACCIDENTAL RELEASES ASSOCIATED WITH THE NUCLEAR FUEL CYCLE AND THE COAL FUEL CYCLE.

APPLICATION (CONTINUED)

. Light Water Reactor Risk

PROGRAM OBJECTIVES

Provide a probabilistic analysis of a comprehensive set of non-core melt accident sequences in a manner consistent with the Reactor Safety Study such that when the results are viewed along-side the sets of conclusions of that study, they provide a complete description of the light water reactor (LWR) risk impact over a comprehensive spectrum of accident sequences. The project will also provide a basis for re-examining the class 3-8 accident sequences as utilized in the present licensing procedure.

LICENSING SUPPORT

• VALUE IMPACT ASSESSMENT OF REGULATORY SAFETY REVIEW UNITS

PROJECT OBJECTIVE:

ASSESS THOSE KEY ELEMENTS USED IN THE LICENSING SAFETY REVIEW PROCESS FOR ASSURING RADIOLOGICAL SAFETY OF LWR DESIGNS. THE PURPOSE IS TO ESTIMATE THEIR RELATIVE VALUES (FROM A RISK STANDPOINT) AND THEIR RESOURCE IMPACTS.

ENCLOSURE 7

Appendix B

APPENDIX B

NUREG-0496

1978 REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM

A Report to the
Congress of the United States of America

Manuscript Completed: December 1978
Date Published: December 1978

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Enclosure 7

11. RISK ASSESSMENT

11.1 Objectives

Risk assessment research, the responsibility of the Probabilistic Analysis Staff (PAS) of RES, has as its mission the development of methods for, and the promotion of the application of, quantitative risk assessment to assist the NRC staff in carrying out its various responsibilities. Its activities span a spectrum from research aimed at the developing and testing of new methods, to application of these methods to problems whose solutions are needed to reach decisions in a number of licensing, inspection, and program-planning areas.

11.2 Scope

The PAS has as a tool the fault-tree/event-tree methodology of the Reactor Safety Study (WASH-1400)* which can provide significant insights into the behavior of reactor systems from a probabilistic risk viewpoint; however, this methodology and the results obtained from it are only beginning to be used in the regulatory process. The PAS thus finds itself initiating new activities for which it sees a need, providing guidance and assistance to those divisions of NRC that are attempting to apply the methods already developed, and working on specific applications of immediate import to some NRC staff responsibility. Although a significant fraction of PAS activity is research, much of what it does is a direct application of earlier research to immediate problems. This situation is desirable, but requires continuing oversight to ensure that a proper balance is maintained between research and application. The present balance seems appropriate.

A significant fraction of the research and development for which PAS has responsibility is done by its own professional staff.

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*U.S. Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), October 1975.

Typical of research and development efforts are:

Development of a risk assessment method aimed at quantifying fire risks and consequences.

Development of computer codes dealing with fault tree manipulation, the effects of testing and maintenance on system and component reliability, and a systematic treatment of common cause failures.

Description and analysis of human errors observed in connection with operating reactors.

Efforts to define an appropriate program of research to examine the question of acceptable risk.

Typical of work that is primarily application of risk assessment to existing or anticipated problems are:

The application of the WASH-1400 risk assessment methods to four different LWR plant designs.

The development of criteria for outage times and surveillance intervals for systems and components.

The development of a model to predict flood occurrence probabilities, associated system failures, and resultant consequences.

Efforts to model the behavior of parameters important to safety in the behavior of a radioactive waste depository located in deep geologic media.

Development of a model for calculating risks to reactor plants due to transportation of non-radioactive hazardous materials nearby.

A study of emergency responses to reactor accident sequences.

11.3 Relation to the Needs of the NRC

The work being done by the PAS and that being planned appear to be relevant to the needs and responsibilities of NRC. The PAS is taking the initiative in defining and developing new areas of investigation. Computer code development programs, although eventually responding to NRC needs, is primarily in this category. The PAS work to collect, correlate, and evaluate performance data is also being done primarily as a result of PAS initiative.

Research on flood risk analysis, fire risk assessment, and the analyses of Class 3-8 accidents for use in environmental reviews is in direct response to requests from various other groups within NRC.

The recent increase in professional staff represents an increase in level of activity commensurate with increasing applications of risk assessment in the licensing and regulatory activities of the NRC. These applications are likely to increase. It is important that the PAS continue to recognize that risk assessment is not an end in itself and that, although the PAS will continue to be responsible for initiating and assisting in the development of new projects, methods must be taken over and used by other divisions as soon as feasible.

11.4 Progress and Results

Of special note are the activities of the PAS in improving the methods first developed in WASH-1400 for predicting consequences of the release of radioactive materials in reactor accidents. Various aspects of this part of the Reactor Safety Study have received serious criticisms, and a major effort is being made by the PAS to improve the method. The basic vehicle now being developed for consequence prediction is called the CRAC Code. It is designed to sample statistically a large population of atmospheric situations and to model a large number of atmospheric phenomena and site characteristics. Results are expected to predict consequences in some representative situations. Although progress is being made in improving the model, there are indications that it still has deficiencies that require further effort. This is an activity which should be pursued with diligence. The PAS is nearing completion of a study that extends the effect of liquid-borne activity on reactor accident consequences beyond that carried out in WASH-1400.

Another study extends the WASH-1400 study to light-water reactors of different designs. This new study includes a reanalysis of the dominant accident sequences using improved models. Special attention is given to analysis of systems designed to mitigate accident consequences and to accident analyses which provide a more advanced treatment to release magnitudes.

In addition, attention should be called to the beginning of a program to assess the risks associated with deep sea bed disposal of wastes.

This program is of special importance, both because of its possible long term implications and because it will require the international cooperation that is necessary for a permanent solution of the waste problem.

The ACRS believes that existing and planned programs of the PAS are responsive to the recommendations of last year's report.

11.5 Findings And Recommendations

Risk assessment is an expanding area and needs for both development of new techniques and applications of existing methods are likely to grow. The ACRS has not found any serious gaps in the existing program. However, a number of items deserve emphasis.

1) As the PAS and others have observed, and as the Risk Assessment Review Group (RARG) Report* emphasizes, accurate risk assessment requires a data bank of performance histories of components and systems. The PAS is working within the NRC and with others to collect and evaluate data. It should continue to emphasize this activity and also should provide guidelines to ensure that appropriate information is reported to those responsible for collecting reactor system performance information.

2) A point of continuing concern in connection with accident consequence prediction is the appropriate description of biological effects of radiation. The BEIR Committee is scheduled to release a report within a few months. The ACRS recommends that the consequence calculations be re-examined in light of the recommendations of that report when it is released.

3) Many of the PAS research projects result in sophisticated computer codes applied to specific systems with assumption about such items as failure modes and uncertainties on data. The ACRS believes that there is a need for quality assurance in the methodology and application of probabilistic analyses. The ACRS recommends that a systematic method of evaluation be developed which includes the necessary documentation of assumptions needed to enable peer review.

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*H. W. Lewis, et al., Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, NUREG/CR-0400, September 1978.

4) Many comments, including those in the RARG Report, have stressed the importance of further development of methods to evaluate more quantitatively the contribution of human error to risk. It is equally important that the contribution of operator adaptability be evaluated, because it may be a significant contributor in decreasing risk. An accurate evaluation may well provide insights into improvements in operator selection and training which could be implemented to further enhance safety of reactors.

5) After exchanges of correspondence with the EPA, the NRC agreed to undertake a study to determine acceptable levels of risk. This subject is of significance not only to the NRC but to virtually every organization making decisions that could affect the health and safety of the public. The ACRS believes that such studies are very important and there is a need for consideration of acceptable risk by each such organization. However, the ACRS believes that there is need for a comprehensive research program with the goal of defining potential criteria for societal risk acceptance, conducted with broad support from the many federal departments and agencies involved in such decisions, and conducted under the auspice of an organization not tied directly to the problems of any specific activity or regulatory decision.

6) Finally, the ACRS recommends that careful consideration be given to the recommendations of the RARG Report.

January 2, 1979 UNITED STATES SECY-79-8

NUCLEAR REGULATORY COMMISSION

INFORMATION REPORT

FOR: The Commissioners

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

THRU: Lee V. Gossick
Executive Director for Operations

SUBJECT: IMPROVING THE PROCESS FOR DETERMINING THE NEED FOR NEW REACTOR REQUIREMENTS

PURPOSE: To reply to the September 1, 1978 memorandum from Samuel J. Chilk to Lee V. Gossick on the subject matter.

ISSUE: The Commission requested the staff to consider a number of questions related to the process for determining the need for changes in reactor licensing criteria and requirements.

DISCUSSION: In a September 1, 1978 memorandum from S. J. Chilk to L. V. Gossick, the Commission requested the staff to consider a number of questions related to reactor licensing requirements and criteria. These questions dealt with the process for developing and approving new staff requirements, ways of better defining and documenting the criteria used in the decision process and improving the process by providing input from outside the staff. While each of the questions is addressed in the enclosure to this paper, a brief summary is provided below.

First, the criteria for deciding when a requirement is essential for safety depend on large measure on an interpretation of the broad language of the Atomic Energy Act of 1954, as amended. The statutory base from which our regulations flow contains such terms

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as "adequate protection," and "unreasonable risk." The regulations and staff criteria represent an attempt to quantify or at least give some dimension to these very general statutory criteria.

There has been a continued evolution towards increased specificity in reactor licensing criteria. To a large extent, changes in review requirements reflect the developing background, experience and interest of the staff in the different technical areas over the years, and the influence of changing interests and concerns expressed by the ACRS, the hearing boards, the industry, and the public. From time to time the staff has developed increasingly specific statements of its requirements and its review methods beginning with the General Design Criteria, and extending through a variety regulations, Regulatory Guides, and the Standard Review Plan (SRP) for nuclear power reactors.

The NRR Standard Review Plans for safety reviews were developed by an intensive staff effort over a period of about two and one-half years beginning in mid-1973. On August 12, 1975, the Director of NRR (NRR Office Letter No. 2), noted that the Standard Review Plans (SRPs) were complete and in use by the staff. The Director established a requirement that, except for clarification or corrections of errors, proposals to modify the plans in the future would be considered only for matters of major safety significance and only after review by the Regulatory Requirements Review Committee (RRRC) and authorization by the Director, NRR. In that letter, the Director of NRR stated:

> "The SRPs represent the integrated result of the hundreds of conscious choices made by the staff and by the nuclear industry in developing design criteria and design requirements for nuclear power plants. Now that the plans are published and in use, they represent the most definitive basis available for specifying NRC's interpretation of an 'acceptable level of safety' for light water reactor facilities."

Thus, the base for evaluating changes in regulatory requirements was established for CP safety reviews. The Director of NRR issued additional instructions requiring that all deviations from the acceptance criteria of the SRP be documented for all CP and OL safety reviews.

One of the principal objectives of the SRP was to provide stability to the licensing process. The SRP reflected the review process as it had evolved up to 1975 and was not based on a re-examination or a new determination and definition of the "appropriate" review. As a result, the original plans vary widely in completeness and specificity. Substantial improvements to the SRP are being made on a continuing basis but at the cost of of significant staff manpower.

It is against this backdrop that the RRRC and management of NRR must assess proposed new regulatory requirements brought forward to them. Although the SRP published in 1975 was viewed as "the most definitive basis available for specifying NRC's interpretation of an 'acceptable level of safety' for light water reactor facilities," it is important to recognize that there continue to be pressures to expand the areas of review, and to develop more stringent requirements. These pressures stem from two basic sources: (1) new technical information or unfavorable operating experience; and (2) attempts to improve safety, especially for new designs.

The staff's role is basically that of setting forth safety criteria and performing an audit or review of the utility's reactor design against these criteria. LWR designs and the staff's criteria have evolved in parallel. With each increment of adverse operating experience, or new piece of analysis that discloses some element of significant uncertainty, or marginal engineering practice, there is a tendency on the part of the staff, the boards, or interested members of the public to require yet more stringent safety standards. Similarly, as power reactors have grown in size and complexity, the staff has required additional safety features deemed appropriate for such applications. Beyond this, there are continued pressures to improve the safety of all reactors where practical means to provide substantial additional protection exist.

In part, this is reflected in a prevailing view that new classes of reactors should be safer than older reactors. Any proposed changes in regulatory requirements to achieve that goal must also be evaluated to determine their applicability to older plants, and it is here that the regulatory decisions are most difficult to make.

The criteria used in this decision-making process are difficult to quantify; however, the factors considered include improvement in safety and environmental protection balanced against the impact of the new requirements. Such a value-impact analysis should not be construed to mean that cost considerations must take precedence over considerations of health, safety, or national security. These factors are paramount. However, cost is an important factor which is to be considered particularly in evaluating alternative means for achieving the desired ends.* We view the purpose of a value-impact analysis to be the documentation of the logic used to develop and choose among alternative actions for achieving a needed safety goal. NRR Office Letter No. 16 has been issued to provide guidance to the staff for preparation of value-impact statements.

With respect to improving the decision process by providing opportunities for non-NRC participation, we find ourselves attempting to achieve conflicting goals. On the one hand, participation by non-NRC groups would provide additional input for value-impact analysis from the public and industry. On the other hand, such participation may result in lengthening the decision-making process. In addition, there will probably be cases that require more immediate action by the staff for safety reasons, thus not allowing time for such participation. Those matters considered by the technical staff to be of sufficient potential importance that they require prompt attention in on-going reviews are termed Category IV items.

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* See Attachment 2 & Response to Issue No. 2 for detailed discussion of consideration of economic impacts.

The issues identified by the Commission are difficult to address and warrant additional staff actions. These and related issues have been raised in other forums by the Commission (see memoranda from the Secretary to the EDO dated July 14, 1978 (ref: SECY-78-109), March 16, 1978 (ref: SECY-78-108) and November 15, 1977 (ref: SECY-77-561), by industry (see AIF letter to Chairman dated September 6, 1978 and GE letter to the Secretary dated September 5, 1978) and other groups (see June 5, 1978 letter from PIRG and our reply dated September 14, 1978).

The enclosure to this memorandum addresses each of the issues (numbered consecutively) identified by the Commission in the September 1, 1978 memorandum from S. J. Chilk to L. V. Gossick and provides an indication of our current perception of each issue. In some instances, staff actions aimed at providing more definitive responses to the issues or improving the current decision process for establishing new requirements are underway or being contemplated. The staff has also endeavored to respond to the issues raised in the March 16, 1978 (re: SECY-78-108) and November 15, 1978 (re: SECY-78-561) memoranda from the Secretary to the EDO.

Some of the actions underway include the following:

- Increased use of probabilistic risk analysis in establishing priorities for work tasks (e.g., RES review of generic technical activities).

- Increased use of probabilistic risk methodology in the value-impact analysis for new requirements (e.g., ATWS, NUREG-0460).

- Increased use of value-impact analysis for new regulatory requirements (e.g., NRR Office Letter No. 16).

- Changes in current RRRC procedures to provide earlier opportunities for public comments on proposed substantive changes in regulatory requirements (see staff reply to AIF letter of September 6, 1978).

Some actions under consideration include:

- Establishing an advisory group of senior technical staff to support the RRRC.
- Establishing procedures which would permit an earlier indication by RRRC of its views on proposed changes.
- Issuing an Office Letter establishing guidelines for Category IV matters including a time limit for consideration by RRRC.

While the completion of the actions discussed in this paper will permit additional marginal improvements in the licensing process, substantive gains can only be realized through new policy initiatives. As may be evident from the discussions in this paper, one of the major impediments to an efficient and effective licensing process is the lack of clearly defined criteria for making changes to existing staff criteria and practices. The Standard Review Plan was intended to establish a baseline of staff practices on new applications; however, there has been a tendency to apply these requirements to older facilities and to continually develop yet additional requirements for new facilities.

One mechanism that might add stability to the licensing process would be to establish a regulatory requirements "cut-off date" for each project in review. The establishment of a "cut-off data" would permit only those changes that are necessary to provide substantial additional changes to be implemented. All other chances (that improve safety) would be accumulated and implemented at some future date (perhaps on a yearly basis). Such a mechanism would afford the industry and the public to have some advance notice of new requirements and not require projects in review to be impacted by changes of marginal significance.

The feasibility of this concept and other measures to improve the process need to be examined further. I intend to submit additional papers to the Commission when our thoughts on this subject are better developed.

Coordination: This paper was coordinated with the Office of the Executive Legal Director. OELD prepared the Legal Memorandum in Attachment 2. Separate OELD comments are contained in Attachment 4.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Responses to Issues Identified in September 1, 1978 Memorandum S. J. Chilk to L. V. Gossick
2. Attachment 1 - Staff Response to March 16, 1978 Memorandum from Secretary to EDO
3. Attachment 2 - Legal Memorandum (prepared by OELD) - Consideration of Economic Impacts
4. Attachment 3 - Staff Response to November 15, 1977 Memorandum from Secretary to EDO
5. Attachment 4 - OELD Comments

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ENCLOSURE 1

RESPONSES TO ISSUES IDENTIFIED

IN SEPTEMBER 1, 1978 MEMORANDUM

S.J. CHILK TO L.V. GOSSICK

Issue No.1

How might the staff most expeditiously proceed to define in more explicit -- if not quantitative -- terms the criteria for deciding when a requirement is essential to safety, while still recognizing that judgment is an inherent part of such decisions?

Response:

To provide a broader perspective on this issue, the response has been divided into two parts. The first covers the general subject of how changes in regulatory requirements are presently controlled, including the criteria used. The second covers the results of efforts initiated toward developing improved methods to determine which changes should properly be backfitted since they fall within the definition of "substantial, additional protection which is required for the public health and safety or the common defense and security" (Section 50.109(a) of 10 CFR Part 50).[1]

Discussion of Regulatory Requirements Change Process and Criteria

The Standard Review Plan is the principal means for achieving a consistent and adequate review of license applications. As stated in NRR Office Letter No. 2 dated August 12, 1975: "The SRPs represent the integrated result of the hundreds of conscious choices made by the staff and by the nuclear industry in developing design criteria and design requirements for nuclear power plants.

1/ This discussion and Attachment 1 to this Enclosure are also intended to address a similar question posed in the March 16, 1978 memorandum from the Secretary to the EDO.

Now that the plan is published and in use, it represents the most definitive basis available for specifying NRC's interpretation of an 'acceptable level of safety' for light water reactor facilities."

In terms of changing safety requirements, NRR Office Letter No. 2 also requires that "adoption of a substantive increase or decrease in the safety requirements of the Standard Review Plans will require consideration by the Directors of Reactor Licensing and Technical Review (now the Directors of the Divisions of Project Management, Operating Reactors, Systems Safety and Site Safety and Environmental Analysis), review by Regulatory Requirements Review Committee, and authorization by the Director, NRR."

Thus, the Director, Office of Nuclear Reactor Regulation (NRR), must approve substantive changes to established licensing requirements before they are implemented. Prior to approving any licensing requirement change, the Director, NRR, obtains recommendations from the Regulatory Requirements Review Committee (RRRC) regarding the proposed changes. Since the RRRC is composed of senior management personnel, it is used by the Director, NRR, to evaluate proposed changes to licensing requirements referred to the RRRC after line-management review. The information provided for each proposed change includes the bases for the change, justification for the change (e.g., a value-impact statement prepared in response to NRR Office Letter No. 16), and a suggested implementation schedule.

The RRRC considers matters that could have a substantial impact on operating plants, plants undergoing an operating license review, plants under construction, plants undergoing a construction permit review, approved standard designs, standard designs under review, approved sites, sites under review for early approval and on new applications of future plants, designs and sites. Thus, the RRRCs's recommendations are to: (1) reject or defer the proposed changes; (2) implement the proposed change on new and future applications only (a Category I matter); (3) implement the proposed change on new and future applications and evaluate whether operating plants and plants with construction permits should be backfitted on a case-by-case basis (a Category II matter); and (4) implement the proposed change on all applicable plants, designs and sites (a Category III matter). A more complete description of the Category I, II and III matters is as follows:[2/]

The regulatory position of each approved proposed guide (or proposed guide revision) will be characterized by the Committee as to its backfitting potential, by placing it in one of three categories:

Category I - Clearly forward fit only. No further staff consideration of possible backfitting is required.

Category II - Further staff consideration of the need for backfitting appears to be required for certain identified items of the regulatory position -- these individual issues are such that existing plants need to be evaluated to determine their status with regard to these safety issues in order to determine the need for backfitting.

Category III - Clearly backfit. Existing plants should be evaluated to determine whether identified items of the regulatory position are resolved in accordance with the guide or by some equivalent alternative.

2/ Meeting Summary of the RRRC Meeting No. 31 dated July 11, 1975.

Category I requirements need be considered only as forward fit regulatory requirements for plants, designs and sites yet to be approved. Limiting actions actions to new applications (Category I) implicitly embodies a view that safety improvements should continue to be sought for future plants, even though the risks associated with previously licensed reactors are acceptably low. Consequently, many changes in regulatory requirements that are only incremental in nature, are justified on the basis of a value-impact assessment only for application to future plants.

The categorization of a requirement as Category II or Category III by the RRRC does not imply that such requirements must be implemented. The requirements approved by RRRC, e.g., Regulatory Guides or changes to SRPs, identify <u>one acceptable way</u> to satisfy the intent of a regulation. It is recognized that there may be other, equally acceptable, alternate solutions or, as in the case of Category II matters, justification may exist for requiring that no additional action be taken for plants under construction or in operation.

When it designates a requirement as Category III, the RRRC has determined that the proposed action will offer sufficient additional protection that is required for the public health and safety and the requirement or equivalent alternate actions should be implemented by the licensee or applicant. The recommendation of the RRRC is forwarded to the Director of NRR.

No criteria have been established that explicitly define the minimum increment in safety that constitutes substantial additional protection (see attachment 1 for additional discussion). However, there have been attempts to define the judgmental process through which a decision to take action is made. One example was provided in an ACRS response to a related question from the JCAE: 3/

(JCAE):

"Q. Is there any limit on the number or type of unresolved safety issues that should be permitted to remain unresolved at any one time before nuclear power plant operation should be curtailed?

(ACRS):

A. The important word in the preceding question is type rather than number. Most unresolved safety issues may be classified into the following categories of increasing significance beginning with those of low consequence;

(1) Conditions with potential for degrading system safety but for which it is judged that further theoretical and/or experimental evaluation will demonstrate no safety significance;

(2) Conditions of minor safety significance resulting from marginal engineering practice;

(3) Conditions having known safety significance but which have a low probability of occurrence and marginally acceptable consequences (approaching but less than 10 CFR 100 limits);

(4) Conditions that could lead to low probability accidents of serious consequences whose correction would require extensive evaluation or possibly substantial plant modifications, but where the delay in implementing correction can be justified on grounds of improbability for a limited period of delay;

3/ Response to questions regarding ACRS Testimony Before the Joint Committee on Atomic Energy, Hearing RE: "Investigation of Charges Relating to Nuclear Reactor Safety," February 18, 23 and 24, March 2 and 4 1976.

(5) Conditions leading to events having a high probability of occurrence and possibly serious consequences whose correction should occur prior to plant operation but where consequences can be acceptably mitigated by a decrease in power or other operational restrictions until corrective modifications are completed or where the occurrence likelihood is reduced by other means.

Instances of conditions falling into the first three categories can be numerous without creating significant jeopardy to public safety.

Only a few items in Category 4 would be tolerable at any one time because the cumulative effect would be unacceptable.

A limited number of items in Category 5 might be tolerable for for varying periods of time depending upon the degree to which (a) operational restrictions can effect a reduction in the event probability to a tolerable level or (b) surveillance can provide an acceptable means of mitigating risk.

A fully quantitative basis for making judgments regarding the the type and number of unresolved safety issues which are acceptable is difficult to develop but should be pursued In the current approach major dependence is placed upon reaching a conclusion through engineering judgment that the overall risk for the plant would not be significantly increased by the existence of the unresolved safety issues in question."

While the ACRS response was directed at "unresolved safety issues," it provides a useful beginning point for the broader process of evaluating the entire spectrum of proposed requirements for new and/or existing facilities.

The backdrop for this judgment process is set forth in the General Design Criteria (GDC), Appendix A to 10 CFR Part 50 which provide for a specific level of protection to be afforded. The GDC are further amplified and augmented by Regulatory Guides and by the acceptance criteria of the Standard Review Plan.

A first step towards defining, in more explicit terms, the criteria for deciding when a requirement is essential for safety is to set forth the rationale used by the staff in determining whether a requirement is to be a Category I, II or III matter (used by RRRC). Accordingly, we have attempted to define the implicit criteria used by the RRRC.

There are two sets of criteria involved:

Criteria Set A:

Those criteria used to distinguish the requirements to be considered for backfitting (Category II and Category III) from those that need not be considered for backfitting (Category I).

Criteria Set B:

Those criteria used to distinguish Category II requirements from Category III requirements.

Criteria Set (A) is a collection of characteristics which may be individually attributed to the proposed requirement under consideration for backfitting. For example, when one or more of the following Criteria Set A characteristics is determined to apply to the regulatory requirement, that requirement will be assigned either a Category II or Category III designation:

(1) Result of new information which indicates previous staff positions or practices are seriously non-conservative, i.e, safety margins to assure protection in light of uncertainties are determined to be unacceptably reduced. Actions are taken to restore safety margins to previous levels of protection.

(2) Results of a new or revised risk assessment which indicates that a design change may be necessary because (a) the probability of an occurrence exceeds an acceptable level and (b) the potential radiological consequences are a significant fraction of Part 100 guidelines.

(3) Result of value-impact assessments which indicate that the margin of public protection can be measurably increased without facility modification and at little overall cost, such as by augmented operating procedures, expanded QA coverage, or increased staff training.

(4) Result of staff assessment which indicates that an insufficient base exists to confirm the adequacy of licensee programs or action.

(5) Result of adoption of criteria or guides required by new or revised regulations.

(6) Result of an integrated assessment of previously licensed plants against current requirements considering alternative methods to achieve equivalent levels of protection, such as use of non-safety systems to perform safety functions, administrative or procedural changes, or augmented surveillance.

The Criteria Set B, those used to distinguish Category II requirements from Category III requirements, are as follows:

1. Category II requirements include:

   a. The cost of changes necessary to satisfy the requirement may vary significantly depending upon the construction status of a previously approved design, necessitating a plant specific value-impact analysis.

b. The complexity of the requirement is such that the extent to which changes in design or procedures are necessary will likely be affected by specific plant design features and characteristics.

c. The extent of changes necessary to satisfy the requirement will likely be affected by the degree of conservatism in analyses used to establish the existing design.

d. The changes necessary to satisfy the requirement are such that the schedule for their implementation should be compatible with plant operation, e.g., during refueling.

2. <u>Category III requirements include:</u>

a. The changes are readily discernible as required for all applicable plants and plant designs to assure public health and safety.

b. The changes are required for compliance with regulations.

Thus, there is a stepwise determination of whether a change shall be required on at least some plants, and whether the proposed changes meets criteria indicating that it should be required on all plants and plant designs.

Decisions to backfit are made only after careful evaluation and consideration by senior management. Generally, the bases underlying all the criteria and considerations discussed above are (a) the probability or likelihood of the event or situation occurring, and (b) the consequences in terms of potential offsite releases if the event does occur.

Where data or the type of situation do not permit this type of assessment (including such non-design-related items as QA, operator or personnel qualifications, reactor test programs, emergency planning), other inputs and methods are used. In all cases, however, decisions to backfit are associated with those items that are ultimately judged to have sufficient probability and consequences that the failure to implement them could adversely affect public health and safety.

While the current decision-making processes of the RRRC could benefit from additional quantitative criteria, such quantification is difficult. As discussed in the foregoing, subsequently, and in Attachment 1, we intend to continue to develop improved guidelines for determining when new requirements should be imposed. As such guidelines are developed, they will be considered by the RRRC and made available for comment. Other sections of this paper discuss other aspects having potential to achieve additional improvements in the decision-making process for new regulatory requirements.

Issue No. 2

What needs to be done to clarify the circumstances under which economic impacts associated with new requirements can and should be taken into account and to improve the quality of value-impact analysis of new requirements?

Response:

If there are several alternative ways of achieving the same safety objective or complying with a given safety standard, the applicant/licensee can always choose and the staff can accept the least costly. Exemptions from specific requirements of the regulations may be granted where the same level of safety can be achieved in a particular circumstance by use of less costly procedures or equipment. However, the more difficult case is where the level of safety is actually affected by the choice of alternatives. The critical question, discussed below, is whether economic impacts may be considered in making basic judgments that affect the level of safety presented by a given plant or standardized design or proposed site.

As the Legal Memorandum in Attachment 2 (prepared by OELD) indicates, the Atomic Energy Act of 1954, as amended, does not by its terms require "absolute" protection of the public safety or "zero risk," rather the Act calls for "adequate protection" and prohibits "unreasonable risk."

The definition of what constitutes "adequate protection" or "unreasonable risk" raises a policy question which may legitimately entail some form of balancing judgment by the Commission. In making this balancing judgment, we believe that the Commission may properly consider economic costs to the licensee or applicant and other factors, such as need for power, that bear some direct and logical relation to the balancing judgment and that are in the Commission's expertise.

However, the Commission may not decline to impose a safety requirement based solely upon general consideration of promotion of the national welfare, or improvement in the standard of living. These general policy concerns appear to go beyond the Commission's statutory mandate in the Atomic Energy Act of 1954, as amended.

Translating these general legal theories into actual practice presents some problems. What is "adequate protection" and what is needed to avoid "unreasonable risk" are spelled out to some extent in the Commission's regulations. Economic impacts can play no role in the application of these regulations in individual licensing cases or standardized design reviews since the regulations do not generally permit this. Thus, in order for economic impacts to play any role, there must be some change in the regulations. Such a change would also entail Commission repudiation of some broad language in several adjudicatory decisions that can be read as precluding consideration of economic impacts in making safety decisions.

Beyond this, the range of options open to the Commission in rule-making are not limitless. It is doubtful that the Commission could legally amend its regulations so as to vary the level of safety on a plant by plant basis, depending upon prevailing economic conditions and the financial resources of the applicant or licensee. It is a fair reading of the Act that Congress intended that uniform safety standards would prevail.

To clarify this matter the Commission could issue guidance to the staff concerning the circumstances under which economic impacts should be considered.

With respect to improving the quality of value-impact analysis of new requirements, the Regulatory Requirements Review Committee (RRRC) was established in early 1974 to review significant proposed changes to the Regulations, the Standard Review Plan (SRP) and Regulatory Guides. The RRRC focuses on proposed changes to requirements by reviewing staff evaluations of the proposed changes which include value-impact analyses. In accordance with the value-impact guidelines in SECY-77-388 approved by the Commission, NRR developed instructions for use by its staff for preparation of value-impact analyses in support of significant changes in regulatory requirements (NRR Office Letter No. 16 dated January 31, 1978).

Thus, each proposed change reviewed by RRRC is supported by a value-impact analysis. The quality of these analyses have varied. Some analyses have been very detailed (NUREG-0460 - ATWS); however, there has been and is a continuing need for improvement. Such a need was anticipated when Office Letter No. 16 was promulgated and NRR stated its intent as follows:

"After a period of one year the instructions [for preparing value-impact analyses] will be reviewed and changes made as necessary to reflect experience in their utilization."

In fact, some of the NRR initiatives underway which will provide for public comment on proposed changes (including the value-impact analyses) before RRRC consideration, may result in improved value-impact analyses of proposed changes to regulatory requirements.

Issue No. 3:

How should NRR decisions and the basis for new requirements best be documented and most expeditiously communicated to and implemented by those affected?

Issue No. 4

How can the NRR process (of developing and issuing new requirements) be opened to observation or participation by interested persons outside of NRC so as to improve the quality of new requirements and the timeliness of their implementation?

Response:

According to current NRR practices, the Director of NRR, makes a determination regarding the implementation and backfitting of changes in regulatory requirements following a review, evaluation and recommendation, by the Regulatory Requirements Review Committee (RRRC). Until recently, all documentation relating to the deliberations on proposed changes in regulatory requirements was limited to a distribution internal to the NRC. Although a written summary is generally issued within three weeks after a meeting of the RRRC, this summary usually consists of a brief statement by the RRRC that recommends action regarding a change in requirements. The basis for the recommendation is not normally discussed.

Because of resource limitations, the staff has been unable to implement the requirements at the time of Office Director approval of a change in requirements on a plant-specific basis. To assure that these new requirements are addressed in the licensing process, a decision was made that for CP and OL proceedings the Category II, and III matters should be addressed at the next licensing action. For example, for those applicants having a CP, these matters would be addressed at the OL stage. For standardized designs, these matters would be addressed at such milestones as PDA extensions, PDA amendments, or new PDAs and FDAs.

To complement this effort in the future, a generic letter will be sent to all applicants and licensees and approval-holders requesting that their plant designs be assessed every six months against all Category II and III matters that have been reviewed by the RRRC and approved by the Director NRR during the preceeding six-month period. For those plants which have received operating licenses, the implementation of the RRRC Cagegory II and III matters will be reviewed in connection with a recently initiated program to establish priorities for the backlog of Operating Reactor actions/amendments. This review will establish the general timetable for actions on Category II and III matters. For example, some will be acted on within the context of the Systematic Evaluation Program (SEP) conducted by the Division of Operating Reactors, NRR, some as part of the resolution of specific license amendments and some handled as a separate near-term action. (See Attachment 3 for staff response to Commission inquiry on matter related to SEP).

In addition, the staff is considering changes in the process of developing and issuing new requirements which would provide for earlier public and industry input to the NRR process.

Some specific changes that would provide for an increased opportunity for public participation in the process have been recommended very recently in a letter dated September 6, 1978, to Chairman Hendrie from Mr. B. B. Parker, Chairman of the Atomic Industrial Forum. We have reviewed the AIF recommendations and are planning to adopt the following changes in current RRRC practices:

1. Public comments will be requested on proposed substantive changes in Regulatory Guides or in the Standard Review Plan and the associated implementation schedules and draft value-impact analyses before the proposed change is considered by the RRRC.

2. All public comments received on such a proposed change and the associated material will be considered by the RRRC in developing its recommendations;

3. A summary of the results of the RRRC meeting, the Committee recommendations, and all of the associated documents and comments considered by the RRRC in reaching its recommendations will be made publicly available shortly after each Committee meeting; and

4. The NRC Office Director having responsibility for deciding whether to implement the recommendations of the RRRC will not take action on the Committee recommendations until after they have been made publicly available and a reasonable period for appeal of the Committee's recommendations has passed.

We believe that these changes in the current RRRC practices would enhance the opportunity for public involvement in the process of establishing and implementing new or modified regulatory requirements. The increased public involvement and visibility should help to improve decision and public perception of the quality of the decision-making process. However, each of these improvements will be gained probably at the cost of time, since each of the considered changes will add procedural steps to the process.

Issue No. 5

Might RRRC membership and structure be altered to more appropriately account for the extent of demands on the time of senior staff personnel and the possibility of conflicts with their other duties?

Response:

It is important to note that there is little current backlog of proposed changes or issues which are awaiting RRRC action or consideration. However there are a number of identified proposed changes being prepared or planned for RRRC consideration.

The RRRC is an important element of management control. The membership of the Committee brings balance, experience and diversity of perspective to the review of significant new requirements or relaxations of requirements.

The deliberative nature of the Committee's actions further decreases the probability that requirements might be imposed that do not reflect significant safety issues or that might actually decrease overall safety. The Committee's action can result not only in upgrading the overall safety of a plant, but also in making the licensing process more efficient, by eliminating requirements with little potential for improving the public health and safety. Thus, it is our view that the current membership and structure of the Committee is appropriate.

However, there are certain changes to current procedures and structure being examined in an attempt to improve the effectiveness and efficiency of the RRRC. One possible procedural change being considered is earlier involvement and input from RRRC. If a mechanism can be developed which would permit earlier consideration of proposed changes and therefore earlier instruction to the staff as to the initial position of RRRC, then the efficiency of the process and expenditure of staff resources may be improved. However, while such a change may have a beneficial effect, particularly with respect to the staff-perceived time-consuming nature of the process; there are several negative attributes associated with such a change. The RRRC will be required to make its decisions based upon incomplete or very preliminary staff evaluations. Negative decisions by the RRRC on this basis may be viewed by some as management insensitivity to the need for changes to improve safety. In addition, such a change may conflict with efforts for early non-NRC participation in the decision-making process.

Another change being contemplated would be the establishment of an Advisory Group made up of senior technical staff. Such an Advisory Group would function similarly to the Advisory Group to the Technical Activities Steering Committee described in NUREG-0410. This group, chaired by the Secretary of the RRRC, would review and comment on a proposal to be brought forward to the Committee and make specific recommendations to the Committee regarding their disposition.

As noted, these changes are being contemplated and deserve careful consideration by management to determine if they should be implemented. In view of other changes being considered to RRRC procedures, it may be prudent to implement these changes in a staged manner so that the efficacy of each change can be evaluated.

Issue No. 6

What changes in NRR procedure might be adopted which would take better account of the concern that the precedent established by imposing new requirements in individual cases in the interim prior to RRRC review and approval (so-called Category IV) makes RRRC approval and NRR adoption for generic use a foregone conclusion?

Response:

NRR Category IV items have been identified to be those proposed changes in regulatory requirements considered by the technical review staff to be of sufficient potential importance that they should be addressed immediately by applicants in ongoing reviews, prior to the normal review and decision-making process conducted through the RRRC.

A potential Category IV matter is proposed via a written submittal to the Director, NRR by the Division Director proposing the change in requirements. On approval by the Office Director, a Category IV matter can be addressed in ongoing reviews. Categorization of an item as NRR Category IV is intended as an interim measure. RRRC consideration of the matter is to be scheduled as soon as is practical and the matter, if recommended for approval, is then recategorized as an RRRC Category I, II, or III issue.

Several improvements are under study regarding proposed Category IV items. The changes discussed below would reinforce the NRR commitment to initiate RRRC consideration of an existing Category IV matter as soon as practicable. This prompt consideration would also help to offset any perception, by observers of the regulatory process, that Category IV matters are automatically going to become Category II or III matters as a result of implementation before final RRRC evaluation. The changes under study are as follows:

1. Issuance of an NRR Office Letter directing that proposed Category IV matters must be justified on the basis that it is necessary from a safety standpoint or that to wait for an RRRC determination on the matter might foreclose the development and selection of design alternatives on specific, identified applications unless the applicant is advised immediately of the staff concern.

2. Classify a proposed requirements change as a Category IV matter only after it is approved by the Director, NRR, based on a written opinion from a Division Director, to the Director of NRR, which explains the staff position and provides the justification of the matter required by 1 above.

3. Make the decision of Director of NRR establishing a Category IV item publicly available. The Director's decision document and all staff material addressed to the Director in support of the proposed regulatory requirement will be made available for public comment. The new Category IV item will be forwarded to the RRRC for consideration within six months of the public announcment. Public comments and position statements from interested parties will be invited for RRRC consideration within 60 days after public announcement.

4. Require applicants with ongoing reviews affected by a Category IV requirement to respond promptly to the staff concerns. Applicants' responses will provide factual information concerning the capability of the proposed nuclear facility to satisfactorily ameliorate the staff concern expressed by the Category IV matter. Applicants' responses may contain value-impact analyses or any other information considered relevant to staff consideration of the matter as it relates to the specific application under review. The information submitted by the applicant, and the staff's technical evaluation of that information, should be sufficient to support a preliminary staff conclusion for the docket under review.

Issue No. 7:

How might NRR procedures be improved to prevent the further accumulation of generic issues and to introduce greater predictability with respect to requirements to be imposed?

Response:

Substantial steps have already been taken that have and should continue to aid in preventing the further accumulation of generic issues. NRR's Program for the Resolution of Generic Issues was initiated in 1977 to improve upper level management control and oversight of generic technical activities to assure their timely completion. Since the program's initiation four Category A and two Category B generic tasks have been completed.* On the other hand only two additional tasks (Category B) have been approved for inclusion in the program since the original assignments made by the Technical Activities Steering Committee in July of 1977. Accordingly, there has been no increase in the number of generic tasks since the generic issues program was initiated.

The staff is currently performing a re-evaluation of the priority assignments of all generic tasks in the NRR program. This re-evaluation utilizes in part a risk-based assessment of safety significance to aid in developing relative priorities. This risk-based assessment may form the basis for

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* See NUREG-0410, "NRC Program for Resolution of Generic Issues Related to Nuclear Power Plants," January 1978.

decisions not to proceed on certain generic tasks. At the same time consideration will be given to incorporating such risk-based assessments into a screening technique for proposed new tasks. Although all generic tasks cannot be judged by risk-based arguments, such an approach could be helpful in assuring that those tasks that have only minimal discernable value are not undertaken.

With regard to predictability of requirements to be imposed as a result of the resolution of generic issues, the Task Action Plan for each task provides an indication that new or revised requirements may be forthcoming. The Task Action Plans are publicly available and describe the technical problem being considered, the staff's approach to its resolution and the expected end product [illegible] the task. In many cases, the expected end products are revisions to the Standard Review Plan or draft Regulatory Guides. This information in the Task Action Plans puts the public and the industry on notice that changes in requirements are being considered.

In any event, any Regulatory Guide or SRP technical position that might result from a generic task must proceed through the normal management approval chain (including RRRC review and approval). As noted in response to an earlier question, changes in procedures are currently being considered to improve public and industry participation in this approval process.

Issue No. 8:

What might be done to better distinguish the basis for permitting a licensed reactor to continue operation pending implementation of a new requirement, whereas the operating license for a complete reactor may be withheld until the new requirement has been incorporated?

Response:

This issue recognizes that differences exist in implementing new requirements on new reactors (prior to OL issuance) and backfitting new requirements on operating reactors (post OL issuance). Some of these differences include man-rem exposure, cost, impact upon operation and difficulty of implementation. The RRRC considers these differences in implementing new requirements and evaluates the incremental improvement in safety which results prior to assigning a backfit category.

Several possibilities exist for improving the visibility of the bases for the RRRC position on backfitting.

1. Consolidate into a single document the NRR position on backfitting of new requirements to operating reactors. This should include: (1) the RRRC policy on backfitting from RRRC meeting Number 31, summary dated July 11, 1978; (2) the positions taken during budget development

with respect to delayed implementation for Category II and III requirements until operating plants are reviewed under SEP (Dircks memorandum for Case on Reprogramming dated June 19, 1978) and (3) the consideration of Category II and III matters in establishing priorities for DOR amendments/actions now underway.

2. The differences between CP, OL and Post-OL stage in licensing would be included in the Value-Impact Assessments required by NRR Office Letter No. 16.

3. Revise the RRRC procedures to include promulgation of a technical basis for the assignment of Category I, II or III to a new requirement. This technical basis should be proposed by the organization proposing the new requirement. A summary of the conclusions from the Value-Impact Assessment may be sufficient.

Issue No. 9:

How might NRR identify and eliminate elements of the Standard Review Plan which make an insignificant contribution to overall plant safety, so that staff and industry resources can be focused on matters of most significance to safety?

Response

The recent Lewis Committee Report (NUREG/CR-0400) stated that proper application of the risk assessment methodology contained in the Reactor Safety Study (RSS) can provide a tool for the NRC to make the licensing and regulatory process more rational, by more properly matching its resources to the risks provided. NRR agrees with this view and in cooperation with RES is attempting to reevaluate the assigned priorities for work on generic issues.

A similar application of this methodology to the SRP, as suggested in the Commission question, is to be initiated. RES is funding Sandia Laboratories in FY 1979 to conduct a preliminary review of the SRP to identify areas of most safety significance. NRR will follow the results of this effort closely. Such a study could identify areas of less importance.

ENCLOSURE 2

ATTACHMENT 1 - STAFF RESPONSE TO
MARCH 16, 1978 MEMORANDUM FROM SECRETARY TO EDO

STAFF RESPONSE TO MARCH 16, 1978 MEMORANDUM

FROM SECRETARY TO EDO

Discussion of "Substantial, Additional Protection"

Section 50.109(a) of 10 CFR Part 50 specifies that backfitting* is appropriate if the Commission "...finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security." (emphasis added).

While this regulation appears to be straightforward as to what backfitting is and when it may be required, determination of what constitutes substantial additional protection is subject to wide interpretation. The need to limit backfitting to changes that provide substantial, additional protection is clear, but the methods to be used to quantify 'substantial' protection are neither clear nor straightforward.

Possible approaches for a systematic method to quantify and evaluate incremental reductions in the risk to public health and safety have been examined by the staff and discussed with a number of knowledgeable organizations. This activity has resulted in a number of general conclusions as follows:

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* 'backfitting' which in the regulation means "...the addition, elimination or modification of structures, systems, or components of the facility after the construction permit has been issued."

(a) There is a need for a systematic and objective method of evaluating new safety requirements or issues.

(b) The application of probabilistic risk methodologies, have the potential to fill the need for a significant improved method in assessing new regulatory requirements and issues.

(c) Problem areas associated with probabilistic risk assessment include (1) data base adequacy; (2) lack of agreement on model and assumptions; (3) inability to verify results (4) lack of sufficient individuals knowledgeable of the methodology; (5) inability to quantify all contributors to risk, particularly concerning issues independent of design (e.g., emergency plans, QA, test programs); and (6) difficulty in comparing the reduction of risk in units equivalent to those terms used to evaluate impacts. While these difficulties indicate that probabilistic risk assessments are not a panacea, however, many effective and valuable risk assessment studies have been completed. Continued use and development of risk assessment methodologies will minimize these difficulties.

(d) Probabilistic risk assessments can be performed to provide at least a semi-quantitative basis to support 'backfit' decisions. These assessments would not only provide additional input to the decision, but would also provide insights into the rationale for the decisions made.

There are a number of ongoing and future efforts which are directed at developing analytical methods for assessing relative safety importance. These efforts when coupled with other improvements in the decision making process, should provide a consistent, defined and balanced basis for judgments regarding new regulatory requirements.

These activities involve some application of probabilistic risk assessment. The results of these efforts should extend and improve the present capability of the staff for performing such assessments, and for the development of criteria upon which to judge incremental improvements in safety.

The actions that the staff has identified in the responses to the September 1, 1978 memorandum from S. J. Chilk to L. V. Gossick will provide a more systematic and objective evaluation of new regulatory requirements. These actions will constitute a strong additional input to the current RRRC decision-making process.

Achieving a systematic, objective, uniform method of determining what changes constitute substantial additional protection required for public health and safety is a high priority activity within NRR. In the final analysis, however, we believe that the decision must be one of judgment. Yet, this decision must be guided by assessments of the likelihood and consequences of the safety concern, the impacts of implementing corrective action, and the need to assure that there is a balancing of potential sources of risk. The actions which have been identified are directed to accomplishing this objective.

ENCLOSURE 3

ATTACHMENT 2 - LEGAL MEMORANDUM (PREPARED BY OELD)
CONSIDERATION OF ECONOMIC IMPACTS

LEGAL MEMORANDUM
(Prepared by OELD)

CONSIDERATION OF ECONOMIC IMPACTS

Economic Impacts, Promotion of the General Welfare, and Broader Considerations

The issue to be discussed here is whether economic impacts, promotion of the "general welfare", and/or impacts on national programs and objectives, may properly influence Commission judgments regarding public health and safety or common defense and security matters.

There are three possible legal theories under which consideration of such matters could be authorized. These theories relate to interpretation of the standards set forth in the Atomic Energy Act of 1954, as amended ("Act"); interpretation of NEPA; and an interpretation of the Energy Reorganization Act of 1974 ("Reorganization Act"). Each of these possible legal theories will be discussed below in turn.

A. Standards Under the Atomic Energy Act of 1954, as Amended.

(1) Statutory Language

The Act starts out with a broad statement of National policy that, among other things, the "control of atomic energy" shall be directed "so as to make the maximum contribution to the general welfare, subject at all times to the paramount objective of making the maximum contribution to the common defense and security" 1/ and "so as to...improve the general welfare, increase the standard of living, and strengthen free competition in private enterprise." 2/ The Act continues that "it is the purpose of this Act to effectuate the policies set forth above by providing for...a program for Government control of the possession, use, and production of atomic energy and special nuclear material...so directed as to make the maximum contribution to the common defense and security and the National welfare..." 3/.

However, the statutory standards in the sections of the Act dealing with issuance of licenses and promulgation of regulations dealing with activities of licensees are, for the most part, numerous variations on a theme of protecting public health and safety and the common defense and security. For example, the Commission is prohibited from issuing any special nuclear material licenses if the Commission finds that this "would be inimical to the common defense and security or would constitute an unreasonable risk

1/ Section 1a.

2/ Section 1b.

3/ Section 3c.

to the health and safety of the public."[4/] A similar standard is set forth for exempting classes or quantities, or uses or users, of special nuclear material from licensing.[5/] The Commission is also authorized to establish by rule, regulation, or order, such standards regarding the possession and use of nuclear materials as the Commission "deems necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property."[6/]

The statutory standards for issuance of facility licenses are more complicated. The Commission may issue licenses for facilities for industrial or commercial purposes to persons "who agree to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish", and who agree to make technical data available to the Commission when the Commission determines that this is "necessary to promote the common defense and security and to protect the health and safety of the public."[7/] Applicants for facility licenses are also required to submit such information as the Commission may, by rule or regulation, "deem necessary in order to enable it to find that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public."[8/] Thus, it does not appear from the language of the Act that the broad concerns with such matters as promotion of the general welfare stated in sections 1 and 3 are to control the actual exercise of regulatory authority.

4/ Section 57c.(2)
5/ Section 57d.
6/ Section 161b.
7/ Section 103b.
8/ Section 182a.

More importantly, however, the Act does not by its terms require absolute protection or zero-risk. At one point "adequate" protection of public health and safety is required.[9/] In other places the words "unreasonable risk", "minimize danger", "protect", "inimical", "promote", "in accord with", and "guard against" are used. If something less than absolute protection is required, how does the Act contemplate that the Commission would determine whether a certain level of protection is "adequate" or "reasonable"? What additional costs should the country be willing to pay in order to reduce the risks to the public by an additional increment? Some measure of expert judgment is clearly called for, but the language of the Act provides no specific answer to these questions. Neither the Act nor its legislative history are clear regarding what factors are to be considered in making a judgment as to whether a particular quantum of risk is consistent with "adequate protection" and "no unreasonable risk." In this regard it would appear that the NRC has been delegated significant administrative discretion in making a judgment as to what consistutes "unreasonable risk."

(2) Legislative History

The legislative history of the Atomic Energy Act of 1954 confirms that licensing under the Act was not to be based upon a broad "public interest" or "public convenience and necessity" standard but, rather, was to be based essentially on consideration of matters of public health and safety and common defense and security. In the Joint Committee's Report on the measure, facility licensing under section 103 was described as "subject to

9/ Section 182a.

regulation by the Commission in the interest of the common defense and security and in order to protect the health and safety of the public", and the Commission was "required to issue licenses to all qualified applicants without other discretion on its part." 10/ In a separate statement that is part of the Joint Committee's Report, Representatives Holifield and Price criticized the Committee bill because the licensing standards were "barren of any recognition of the public interest in securing electric energy from this new resource at the lowest possible rates." 11/ They believed that the Federal Power Commission's advice during the JCAE hearings that "the grant of the (license) privilege should depend not solely on the negative consideration that national defense will not be harmed, but on the affirmative ground of benefit to the public interest in electric power" should have been followed. 12/ Views similar to those expressed by Representatives Holifield and Price were expressed by Senator Gore of Tennessee during Senate debates on the measure prior to passage. 13/ However, the bill was enacted with the "negative" licensing standards intact.

While the legislative history of the Atomic Energy Act of 1954 is clear that licensing standards were to be confined essentially to matters of public health and safety and common defense and security, the legislative history is barren of any clear indication of how the Commission was to

10/ S. Rep. No. 1699, 83rd Cong., 2d Sess. (1954) and H.R. Rep. No. 2181, 83rd Cong., 2d Sess. (1955) at 20, I Leg. Hist. 758, 1016.

11/ Id. at 121, I Leg. Hist. 369, 1117.

12/ Supra note 20 at 123, I Leg. Hist. 871, 1119.

13/ III Leg. Hist. 3454.

determine when "adequate" or "reasonable" protection is provided.

Subsequent amendments to the Act confirm that the licensing standards were to be confined essentially to matters of public health and safety and common defense and security. For example, section 274 of the Act, added in 1959, provides for the "discontinuance of the regulatory authority of the Commission" with respect to certain nuclear materials, where the Commission finds that the State regulatory program is "adequate to protect the public health and safety." 14/ However, the legislative history of subsequent amendments is similarly barren of any indication of how the Commission was to determine when "adequate" or "reasonable" protection is provided.

(3) Judicial Decisions

The limits of the Commission's regulatory authority were first addressed in New Hampshire v. AEC. 15/ In New Hampshire petitioners argued, among other things, that the terms "health and safety of the public" in the Act were broad enough to include alleged adverse effects attributed to thermal pollution. The Court held that "Congress has viewed the responsibility of the Commission as being confined to scrutiny of and protection against hazards from radiation" and rejected petitioners' argument. 16/ In Cities of Statesville v. AEC, 17/ the Court (on rehearing en banc) rejected

14/ See also S. Rep. No. 390 (to amend section 271), 89th Cong., 1st Sess. (1965) at 4.

15/ Supra note 1.

16/ Supra note 1 at 175.

17/ 441 F.2d 962 (D.C. Cir. 1969).

petitioner's claim that AEC erroneously refused to consider antitrust matters in a section 104b. facility construction permit proceeding. The Court stated that:

> While the regulatory agencies in most of the other fields concern themselves with establishing an efficient national allocation of resources in the area which they were administering, and base this goal on a "public interest" concept of free enterprise, the Atomic Energy Commission concerns itself with promoting technical innovation in a highly experimental field and implementing "public interest" concepts through protection of the health, safety and security of the nation." 18/.

Both the New Hampshire and Cities of Statesville decisions confirm the licensing standards under the Act are confined essentially to matters of public health and safety and common defense and security. However, neither case addresses the question how the Commission is to determine when "adequate" or "reasonable" protection is provided.

The Supreme Court's decision in Power Reactor Development Company v. Electrical Union[19/] ("PRDC") restates the public health and safety focus of Commission regulation[20/] and also begins to suggest answers to the questions of adequacy and reasonableness. That case involved a challenge to the AEC's grant of a provisional construction permit for the construction of the Fermi I reactor. The permit was granted without resolving several serious safety issues, including the issue whether the plant should be designed to withstand a meltdown of the reactor core;

18/ Id at 975.

19/ 367 U.S. 396 (1961).

20/ 367 U.S. 402, 928-29 "The Commission emphasized that 'public safety is the first, last, and a permanent consideration in any decision on the issuance of a construction permit or a license to operate a nuclear facility.'"

these matters were left for resolution at the operating license stage. The Court of Appeals for the D.C. Circuit found the AEC's grant of the permit improper, and the Government appealed. The Supreme Court, reversing, stated that there was "no doubt that construction permits, like all other licenses, can be issued [under the Act] only consistently with the health and safety of the public", 21/ and held that the standard set forth in the AEC's regulations for issuance of provisional construction permits "that [the AEC] has information sufficient to provide reasonable assurance that a facility of the general type proposed can be constructed and operated at the proposed location without undue risk to the health and safety of the public", and the standard set forth in the AEC's regulations for issuance of operating licenses "that the final design provides reasonable assurance that the health and safety of the public will not be endangered", both complied with the Act. In holding that the AEC could defer a definitive safety finding until the operating license stage, the Supreme Court indicated that the extent of investment already made (or, by extension, other like issues) was not to be an element of this "reasonable assurance":

> The respondents' argument is tantamount to an insistence that the Commission cannot be counted on, when the time comes to make a definitive safety finding, wholly to exclude the consideration that PRDC will have made an enormous investment. The petitioners conceded that the Commission is absolutely denied any authority to consider this investment when acting upon an application for a license for operation. PRDC has been on notice long since that it proceeds with construction at its own risk, and that all its funds may go for naught. With its eyes open, PRDC has willingly accepted that risk, however great. ... 22/

21/ Id. at 930.

22/ Supra note 32 at 414-415.

Despite the strength of the "absolutely denied the authority" language, we do not believe the Court's opinion should be read as forbidding the Commission ever to consider economic impacts when making safety judgments under the Atomic Energy Act. The opinion does not address the question to what extent economic or other impacts can be considered in promulgating safety standards, as distinct from deciding individual cases. It is implicit in the AEC's final opinion in the PRDC case that the policy of permitting some issue to remain unresolved at the construction permit stage turned upon a balancing or accommodation between the needs of a developing technology and the needs of sound regulation.23/ Such a balance or accommodation was responsible for the basic AEC decision to allow the project to proceed notwithstanding insufficient data regarding such matters as core melt. Also, PRDC dealt with treatment at the operating license state of safety issues that were unresolved at the construction permit stage. To say the applicant assumed the risk that later resolution of an unresolved item could be costly is different from saying that an applicant assumes all risks of changes regarding plant construction or operation in areas previously considered to be resolved. We do not believe PRDC excludes a cost v. safety gain balance in the latter case.

A later circuit court decision, Siegel v. AEC24/, fleshes out the distinction between case decision and policymaking by endorsing a regulation founded in part on cost considerations. In Siegel the Court upheld the

23/ 1 AEC 128 (1959). See also the discussion in the Government's brief before the Court at p. 45.

24/ 400 F.2d 778 (D.C. Cir. 1968).

AEC's regulations in 10 CFR §50.13 excluding the danger of foreign enemy attack from an inquiry into the merits of a facility construction permit application. In reaching its conclusions, the Court stated as follows:

> What the Commission has esentially decided is that to impose such a burden would be to stifle utterly the peaceful utilization of atomic energy in the United States. Such a decision hardly seems to us to conflict with the Congressional purposes underlying the Act, nor to exceed the scope of the authority given the Commission by Congress to realize those purposes. 25/

Here there is a clear indication that "stifling of the peaceful utilization of atomic energy" -- of which economic cost would be a logical ingredient -- is a factor that may be properly be taken into account in promulgating safeguards standards.

Northern States Power v. Minnesota 26/ carries this reasoning into the public health and safety area. In Northern States Power the Court held that the States were preempted by the Act from imposing limits on liquid radioactive discharges from nuclear power plants. In so holding, the Court stated that:

> through direction of the licensing scheme for nuclear reactors, Congress vested the AEC with the authority to resolve the proper balance between desired industrial progress and adequate health and safety standards. Only through the application and enforcement of uniform standards promulgated by a national agency will these dual objectives be assured. Were the states allowed to impose stricter standards on the level of radioactive waste releases discharged from nuclear power plants, they might conceivably be so overprotective in the area of health and safety as to unnecessarily stultify the industrial development and use of atomic energy for the production of electric power. (Emphasis added) 27/

25/ Id. at 783-784

26/ 447 F.2d 1143 (8th Cir. 1971), affirmed, 405 U.S. 1035 (1972).

27/ Id. at 1153-1154.

Both cases -- and the subsequent district court decision in Nader v. Ray[28]
-- can reasonably be construed as a judicial recognition that in establishing standards to regulate the peaceful uses of nuclear energy the Commission must make balancing judgments, taking cost factors into account. However, consideration of economic factors is neither specifically endorsed nor specifically rejected in the decisions.

This judicial endorsement for considering costs in standard-setting carries over into the Commission's adjudicatory function. The Commission's opinions in individual cases resolve disputed policy issues and in so doing announce general rules of broad application. In the formulation of such adjudicatory rules, the Commission seems equally entitled to make balancing judgments. And, similarly, balancing judgements may be made where the subject area is not covered by any substantive regulation or general policy. However, making the application of rules to the individual case turn on economic or similar factors, or granting temporary or permanent "exemptions" from the regulations on a case-by-case basis, appears not to be supported by these cases. Rather, that resembles the consideration of individual investment which the Court in PRDC found absolutely denied.

(4) Agency Practice

(a) Adjudicatory Decisions

The AEC's (and the Commission's) adjudicatory decisions contain no reference to any balancing process. Indeed, in its adjudicatory decisions the AEC

28/ 363 F. Supp. ..6, 954-55 (D.C.C. 1973) consistently used language implying a "pure safety" approach and rejected any suggestion that non-safety factors may enter into the decision. In the PRDC case the AEC stated that "there can be no doubt that public safety is the first, last, and the permanent consideration in any decision in the issuance of construction permit or a license to operate a nuclear facility." 29/ In the matter of Industrial Waste Disposal Corp., the AEC adopted a conservative safety approach to the licensing of sea disposal of byproduct materials. Commissioner Floberg, in a concurring opinion, characterized the majority opinion as involving "extreme...measures to satisfy even unreasonable popular doubts concerning health and safety," and suggested that part of the Commission's desire to assure the public that primary weight is given to public safety derives from the combination of regulatory and promotional functions in the Commission. 30/

In Public Service Company of Colorado (Fort St. Vrain Nuclear Generating Station), the AEC stated that the Act did not permit matters of economic feasibility of plant operation to interfere with the "controlling consideration of health and safety." 31/

In the more recent decision in Maine Yankee Atomic Power Company (Maine Yankee Atomic Power Station) 32/ the petitioners argued that, in addition to the NEPA cost-benefit analysis, the Atomic Energy Act required case-by-case weighing of safety risks against expected benefits. The Appeal Board

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29/ 1 AEC 128, 136 (1959). But see the discussion in the text supra.

30/ 1 AEC 399, 414 (1960).

31/ 4 AEC 214, 215 (1969)

32/ 6 AEC 1003 (1973)

found this argument inconsistent with the "normal import of the terms used by Congress and the Commission in their formulation of [safety] standards." As the Appeal Board saw it:

> The decision as to whether a threat to health and safety is posed by any particular activity obviously does entail an assessment of the nature and extent of the risks involved. But the quantum of protection to, or endangerment of, public health and safety is not dependent likewise upon how much benefit will be obtained from the activity. In the present context, a specific nuclear power facility is no safer because it is needed and, by the same token, is no more endangering to health and safety because it might be dispensable.
>
> "We might be prepared to lay the statutory terminology to one side if there were legislative history reflecting a congressional contemplation that the safety determinations mandated by the Act might, in some circumstances at least, involve risk-benefit balancing. Our attention has been directed to no such history and, insofar as we have been able to ascertain, there is none." 33/

While broadly stated, the Appeal Board's holding is fairly limited to licensing decisions in which a permanent departure from existing standards is sought on a single case basis, on grounds of economic or similar factors unique to the case. 34/ It is a fair inference from the Act and the case law that Congress intended the nuclear industry to be subject to a system of uniform safety and security standards.

33/ 6 AEC at 1006-1007.

34/ The Maine Yankee reasoning may not be logically reconcilable with the position developed in the previous section of this paper, that balancing is appropriate in cases of rulemaking and general substantive policy making. The same basic statutory standards apply. If the Maine Yankee reasoning was used, economic or similar considerations would not be appropriate even in rulemaking and general substantive policy making. Whether or not Maine Yankee can be distinguished as suggested in the text, economic or similar considerations are appropriate in rulemaking and general substantive policy making. While it is true that the "quantum of endangerment" of public health and safety is not dependent on the extent of the benefit, this is not necessarily to say that, tiven a particular "quantum of endangerment", the extent of the benefit should play no role in the judgment whether that "quantum of endangerment" is consistent with "adequate protection" and "reasonabel assurance."

A related case is one in which temporary waiver of safety-related rules is desired for economic reasons -- for example, to permit continued operation during a peak-power season, or, as in the case of the recent Con Ed request for Indian Point 1, to burn up the core. In this context, broad "economic" considerations are inappropriate. For example, Con Ed's chairman stated in support of their exemption request that unless the exemption was granted, 700,000 barrels of imported oil would have to be consumed, contrary to "the announced energy policy objectives of the President of the United States, the most important of which is to reduce the consumption of foreign oil." It would be wholly impractical for this Commission to consider such broadly parametered issues in such a narrow context, even had it the authority to do so.

Con Ed made no claim that the foreign oil is not available or that needed service would be disrupted by failure to grant the exemption. A claim that safety measures should be postponed for a brief period of peak demand, to preserve necessary baseload capacity, might present a different case. The rules need not be applied woodenly in disregard of the economic or other equities favoring a utility and its customers. Technical review may, or may not, show that no substantial danger would be posed by brief waiver of relaxation of the rules in particular circumstances. Alternative measures may permit limited operation to continue until the demand peak has passed. However, if the industry can seek temporary relaxation or waiver of rules on a case-by-case basis, urging economic or similar matters, a logical avenue will be open for nuclear critics to seek temporary upgrading of the general rules on a case-by-case basis. The better course of action is to limit consideration of economic or similar matters in applications for waiver of rules to cases where the applicant or licensee can show less burdensome alternative measures to achieve the same degree of risk reduction. The Commission's present rules (with one very limited exception) do not provide clearly for consideration of economic or similar matters in the application of rules or in the granting of exceptions or waivers.

(b) Rulemaking

With few exceptions, there appears to be no indication in any of the AEC's rulemaking proceedings that the statutory terms "adequate protection" and "unreasonable risk" permit or require some balancing judgment. AEC statements accompanying promulgation of safety standards said little about resulting industry costs, perhaps reflecting a self-consciousness about its dual "promotional" role. For example, the AEC's lengthy opinion accompanying promulgation of the ECCS rules says virtually nothing about what was in fact a main focus of concern, industry costs. 35/ Yet industry cost is obviously a valid consideration; safety standards necessarily involve trade-offs between cost and increased margins of safety. To state the extreme case, the Commission presumably could devise regulations providing for virtually risk-free reactors, but they would be so expensive

35/ See Acceptance Criteria for Emergency Core Cooling Systems, RAI-73-12-1085, et seq.

that no reactors would be built by private industry. More realistically, a proposed safety standard with high costs and a very marginal increment in safety could properly be rejected by the Commission on economic grounds. In our view, the Commission is fully authorized to tie new safety requirements (as it did explicitly in the ALAP proceedings) to a comparative assessment of cost.36/

There are a few other significant invocations of balancing techniques in a rule making context. Under section 274h. of the Act a Federal Radiation Council (FRC) was established "to advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards..." 37/ In FRC Staff Report No. 1, "Background Materials for the Development of Radiation Protection Standards", May 13, 1960, the FRC recommended that "radiation protection standards...be established by a process of balancing biological risk and the benefits derived from radiation use." This recommendation was adopted by the President, and the FRC guidance served as the basis for the amendments to 10 CFR Part 20 which became effective January 1, 1961. 38/ Since that time the AEC's radiation protection standards have always been based upon a balancing process. 39/

36/ See As Low As Practicable Opinion, NRCI-75/4 278 et seq.

37/ The functions of the FRC were vested in the Environmental Protection Agency under Reorganization Plan No. 3 of 1970, and the FRC was abolished.

38/ CCH Atomic Energy Law Reporter #4046.

39/ In Crowther v. Seaborg, 312 F. Supp. 1205 (D.Colo. 1970) this balancing process was recognized and endorsed.

Similar practice is evident in the criteria which AEC used for the approval of products intended for use by the general public containing byproduct material and source material. Such approvals are accomplished by the exemption, on a case-by-case basis, of the possession and use of the product from the licensing requirements of the Act. The AEC's policy statement in this regard states in part that:

> Approval of a proposed consumer product will depend upon both associated exposures of persons to radiation and the apparent usefulness of the product.
>
> * * *
>
> It is considered that as a general rule products proposed for distribution will be useful to some degree. Normally the Commission will not attempt an extensive evaluation of the public degree of benefit or usefulness of a product to the public. However, in cases where tangible benefits to the public are questionable and approval of such a product may result in widespread use of radioactive material, such as in common household items, the degree of usefulness and benefit that accrues to the public may be a deciding factor. In particular, the Commission considers that the use of radioactive material in toys, novelties, and adornments may be of marginal benefit. 40/

(c) Congressional Testimony and Reports to the President

AEC statements before Congressional Committees and reports to the President have varied on the quest[illegible] whether safety judgments require some balancing analysis. While of dubious legal authority when compared to the Commission's actual performance of its mandate, these appearances do provide a flavor of the Commission's thinking. A representative sampling of these statements is contained in the Appendix.

40/ 30 FR 3462 (March 16, 1965).

(5) Conclusions

Licensing and regulatory decisions under the Act are to be based primarily upon consideration of matters of public health and safety and common defense and security rather than a broad "public interest" or "public convenience and necessity" standard. Under the Act, the Commission has no general mandate to promote the public interest or welfare, or other national goals or policies. Yet, the Act does not require "zero-risk" or absolute safety or security. It speaks of "adequate protection" and "unreasonable risk." Neither the Act nor its legislative history are clear regarding what factors are to be considered in making the judgment whether a particular quantum of risk is consistent with "adequate protection and "no unreasonable risk." Substantial judicial support exists for the proposition that this judgment calls for a balancing of factors, including the need for industrial progress in atomic energy (which translates, roughly, into industrial cost), in reaching general policy conclusions.

Were the NRC writing on a clean slate, the Atomic Energy Act might be construed even more broadly, as authorizing the Commission to take into account such broad national needs and problems as emergent national power needs, dependency on foreign oil, and the like, in promulgating safety rules. As the courts have recognized, the abstract language of the Act grants the Commission very broad discretion in carryout its mandate.[41/]

41/ One court has characterized this Commission's regulatory scheme as one which is "virtually unique in the degree to which broad responsibility is reposed in the administering agency, free of close prescription in its character as to how it shall proceed in achieving the statutory objectives." Siegel v. AEC, 400 F.2d 778, 783 (C.A.D.C. 1968). Quoted with approval in Union of Concerned Scientists v. AEC, 499 F.2d 1069

Such an approach would present a sharp departure from past practice, carried out under the scrutiny and with the approval of the Joint Committee.[42] The AEC's posture was that safety is the overriding consideration. Conservatism in design and "defense in depth" were the watchwords. So far as we are aware, emergent national power needs and broad "economic" arguments (as distinguished from direct industry cost and other direct impacts) have never been publicly deemed justifications for relaxing safety standards in particular cases. If there is to be a departure from past practice along these lines, it would represent a major policy shift. It is likely courts would insist that this shift be made by Congress -- or, alternatively, in some mode (such as rulemaking) which would permit notice to the Congress and possible Congressional reaction before the change was put into effect.

(b) National Environmental Policy Act of 1969

In addition to the requirements of the Atomic Energy Act, NEPA requires systematic analyses of the costs and benefits of major Commission actions significantly affecting the quality of the human environment. Economic impacts may be the decisive factor in regulatory decisions under NEPA.[43] However, NEPA expressly provides that its policies and goals "are supplementary to those set forth in existing authorizations of Federal agencies."[44] NEPA does not, either expressly or by implication, repeal any existing law,[45] including the Atomic Energy Act of 1954, as amended. Rather, NEPA imposes certain additional responsibilities upon the Commission as

42/ The significance of close Joint Committee oversight and apparent approval has been stressed by the courts since the landmark PRDC case. Power Reactor Development Corp. v. Electrical Union, 367 U.S. 396, (1961). Siegal v. AEC, supra.

43/ Calvert Cliffs v. AEC, 449 F.2d 1109 (D.C. Cir. 1971)

44/ NEPA section 105

45/ U.S. v. SCRAP, 412 U.S. 669 (1973)

Judge Wright has stated, "perhaps the greatest importance of NEPA is to require ... agencies to consider environmental issues just as they consider other matters within their mandates."45/

NEPA impacts Commission decisions regarding public health and safety and common defense and security in matters where cost-benefit considerations indicate that requirements above and beyond those required by the Act should be adopted.46/

In the Calvert Cliffs decision, the court defined the trade-off or balance of environmental factors against economic and technical ones:

> The particular economic and technical benefits of planned actions must be assessed and then weighed against the environmental costs; alternatives must be considered which would affect the balance of values .... In some cases, the benefits will be great enough to justify a certain quantum of environmental costs; in other cases, they will not be so great and the proposed action may have to be abandoned or significantly altered 47/

It may be maintained that the relevant NEPA trade-off is the excess of benefits over costs which have been directly accounted for, balanced off against those environmental costs which result from the proposed action, but which would not be accounted for in the marketplace. As an example, the staff typically considers the monetary and environmental costs and benefits of alternative power plant fuels and sites in preparing environmental impact statements.

46/ 449 F.2d at 1112, 1 ELR at 20347

47/ 449 F.2d at 1121, 1 ELR at 20353

(c) Energy Reorganization Act of 1974

The Energy Reorganization Act of 1974 does not, by its terms, amend any of the substantive public health and safety and common defense and security standards set forth in the Atomic Energy Act of 1954, as amended. The House Committee Report specifically stated that "the Commission will continue to carry out those [regulatory] functions under pertinent provisions of the Atomic Energy Act of 1954, as amended ..."[48/] It thus could be maintained that (since the statutory standards remained unchanged) the factors for consideration in applying the standards should also remain unchanged.

A contrary argument draws on the major purpose of the Energy Reorganization Act of 1974, to separate the "developers" from the "regulators".[49/] This was emphasized in the Senate Report which, in describing the applicability of sections 1, 2, and 3 of the Act, states that "all references to encouraging, promoting, utilizing, developing and participating in atomic energy or the atomic energy industry shall not be applicable to the [Commission]."[50/] On this view consideration of such matters as progress in the utilization of nuclear energy, economic impacts, and "energy independence" would be tantamount to exercising some "promotional" function contrary to the intent of Congress.

On balance, we view the first of these two readings as the sounder: consideration of these factors is not prohibited by the Reorganization Act

48/ H.R. Rep. No. 93-707, 93rd Cong., 1st Sess. (1973) at 22, I Leg. Hist. 413. There is no indication of any contrary intent in the legislative history.

49/ Section 2(c) of the Energy Reorganization Act of 1974, as amended. See also, S. Rep. No. 93-980, 93rd Cong. 2d Sess. (1974) at 2, 19 27, II Leg. Hist. 965, 982, 990; H.R. Rep. No. 93-707, 93rd Cong., 1st Sess. (1973) at 4, I Leg. Hist. at 395.

50/ S. Rep. No. 93-980 at 97

itself. First, as the discussion above indicates, the balancing judgment called for by the Act is, strictly speaking, the exercise of a regulatory function -- not a "promotional" function. Second, the conclusion that balancing judgments are called for is based on the regulatory sections of the Act, not the preamble sections referred to in the Senate report. Third, if the statutory standards are unchanged, the factors to be considered in applying the standards should also be unchanged.

The legislative history of the Energy Reorganization Act of 1974 suggests that Congress sought to enhance the regulation of nuclear energy by establishing a separate agency with separate people to perform a purely regulatory mission not by imposing different statutory standards or specifying different factors for consideration in making public health and safety and common defense and security judgments. Thus, on balance, we conclude that the enactment of the Energy Reorganization Act does not affect the basic conclusions set forth in A. above.

It should be noted that this conclusion is contrary to the views of the Hosmer memorandum, which envisions an extremely broad mandate for NRC. The Hosmer position rests largely on an unbalanced and inaccurate analysis of the legislative history of the Reorganization Act. Because the old AEC was being divided along functional lines into ERDA and NRC and the Atomic Energy Act was not being rewritten, it was found helpful to specify which provisions of the Act would apply to NRC, which to ERDA, and which to both. This was done in an AEC General Counsel's memorandum which was inserted in the legislative record.[51] That memorandum stated that several provisions of the act, including Chapter 1 as a whole -- which includes the findings language about the "general welfare," "standard of living," etc. -- would apply to both agencies.

The Hosmer memorandum fixes on this point and in effect concludes, without further analysis, that the NRC mandate extends to consideration of anything arguably in the national interest. But there is a simpler and more reasonable explanation why the General Counsel's memorandum stated that the findings in Chapter 1 would apply to both agencies. Some of the findings and declarations clearly pertain to regulatory functions. [52] Others are applicable to ERDA and NRC, such as subsections 2(f) and (g). Still others have a strongly promotional flavor appropriate only to ERDA, such as subsections 1(a) and 3(a). Since these findings and declarations are largely cosmetic in any event, the draftsmen of the memorandum understandably did not go to the trouble of pointing out which individual findings should apply to ERDA, which to NRC, and which to both agencies.

In any event, for the purpose of determing the substantive scope of the agency's mandate, the Hosmer position attributes too much weight to the findings and declarations in Chapter 1 of the Act. Such statements may be given varying legal effects depending upon the context

51/ S. Rep. No. 93-980 on S. 2744, pp. 82-85; H. Rep. 93-707 on H.R. 11510, pp. 25-28.

52/ For example, section 2(c) states that --

> The processing and utilization of source, byproduct, and special nuclear material affect interstate and foreign commerce and must be regulated in the national interest.

in which they are relied upon. As in the case of the Atomic Energy Act, they may reveal the rationale of the legislation or point to its constitutional basis. 53/ Or they may help lead a court to uphold a statute against a claim of unconstitutionality on the ground that the legislature had no factual bases for its enactment. 54/ In resolving doubtful questions of interpretation of ambiguous substantive provisions, findings and declarations may influence the choice of meaning. But it is clear that legislative findings and declarations have no independent substantive effect, 55/ and that they are entitled to little, if any, more weight than committee reports and other legislative history. 56/ The pertinent legislative history, fairly read, is contrary to the Hosmer position.

Conclusions

Commission licensing of nuclear facilities and materials is not based upon a broad "public interest" or "public convenience and necessity standard". The Commission has no mandate to promote the public interest or other policies such as "energy independence". Rather, under the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended, the Commission's regulatory objectives are

53/ See United States v. Carolene Products Co., 304 U.S. 144, 153 (1939).

54/ See West Coast Hotel Co. v. Parrish, 300 U.S. 379 (1936).

55/ In Harper v. Virginia State Board of Elections, 383 U.S. 663 (1967), the Supreme Court invalidated State poll taxes on equal protection grounds and did not even cite a Congressional declaration of unconstitutionality.

56/ United States v. Carolene Products Co., supra; Katzenbach v. McClung, 379 U.S. 294, 299 (1964).

confined essentially to protecting the public health and safety and the common defense and security. Mr. Hosmer's arguments in a memorandum to the Commissioners to the effect that the Commission has a broad responsibility to promote the general welfare and other "National goals," such as energy independence, are without legal merit. However, the Act does not require "absolute" protection, but only that protection which is "adequate" and avoids "unreasonable risk". The definition of what constitutes "adequate protection" or "unreasonable risk" raises issues of general policy which may properly entail a balancing judgment by the Commission, including consideration of economic costs. 57/ Other factors which bear a direct and logical relation to the Commission's mandate and which are reasonably within the Commission's expertise -- such as need for power -- may also be considered. This conclusion does not apply, however, to application or waiver of substantive rules. The Commission generally could not consistently with past regulatory or judicial decisions permanently vary the level of safety to be required of individual facilities in accordance with prevailing economic or like conditions.

Other factors falling within the ambit of the "general welfare" and less directly related to the Commission regulatory decisions are largely

57/ While a balancing of factors is authorized, neither the Atomic Energy Act of 1954, as amended, nor the Energy Reorganization Act of 1974, as amended, requires a close balance to be drawn between reduction in risk and other factors such as economic impacts. Indeed, the better view would be that the protection of the public health and safety and the common defense and security deserves special weight, and is not simply one among several equal and competing factors. The balancing process is therefore unlike that conducted by the Commission under NEPA where no one factor is given special importance. However, the Commission could refuse to require adoption of a particular measure where the economic costs clearly outweigh, or are wholly out of proportion to, the degree of risk reduction that the measure would offer.

beyond the Commission's expertise -- for example, general improvement in standards of living. Administration policies such as "energy independence" would fall into this same category. The industry might argue, for example, that dependency on foreign oil should be reduced, invoking "Project Independance". That objective could be furthered, the argument would run, by a general relaxation of safety standards, thereby making it cheaper to build and operate reactors. In this vein, the memorandum sent to the Commissioners from Mr. Hosmer argues that:

> Therefore, it seems compelling that energy policies enunciated by the Government, of which NRC is a part, as a matter of course ought to be taken into account by NRC in arriving at its judgments. One of the most explicit of these policies is that the United States shall achieve energy independence at the earliest possible date.

As indicated, the Commission has no statutory mandate to achieve "energy independence" for the country or to promote the general welfare; nor does its expertise as an agency extend to many complex issues, including political issues, involved in achieving the "energy independence" goal. Should the Commission's authority be considered to extend to such matters, the balancing process would not be a balancing between alternative safety standards and the direct costs of achieving such standards, but one of balancing the overall costs of regulation against diffuse national problems, such as inflation, the balance of payments, interest rates, unemployment, etc. Even to state the proposition this way suggests the difficulties involved - even if the balancing process is confined to a rule making context. It would be virtually impossible to consider such diffuse national problems in individual licensing proceedings.

(c) Congressional Testimony and Reports to the President

AEC statements before Congressional Committees and reports to the President have been far from consistent on the question whether safety judgments require some balancing analysis. During the 1960 hearings before the JCAE on "Indemnity and Reactor Safety", Mr. Price, the AEC Director of Regulation, testified as follows:

> We have been asked to comment on a suggestion that the Atomic Energy Commission require applicants to offer alternate sites and to justify the particular site proposed in comparison to other sites.
>
> Site selection by an applicant covers many complex factors in addition to the purely safety considerations. With respect to power reactors, the additional factors include the location of existing power transmission and generation facilities, potential load growth and markets to be served by the new reactor, transmission costs, availability of sites, availability of water for coolant purposes, proximity to railroad lines, land costs, and many others.
>
> The Commission's reactor licensing regulations are based on the philosophy that site selection is the applicant's responsibility, site approval is the Commission's responsibility. Whereas site selection involves economic as well as safety considerations, site approval by the Commission should involve only safety considerations. If a site does not meet the safety requirements it must be rejected by the Commission. If it does meet safety requirements it should be approved.
>
> A mandatory requirement that alternate sites be considered by applicants would produce one of three situations.
>
> First, alternate acceptable sites. To require a formal showing of having considered alternate sites and then to require that the applicant go to an alternate which is considered more safe than his preferred site, which also meets safety requirements, would put the Commission in the position of selecting the

> applicant's site. This could not be done without judging economic considerations, and the Commission has been reluctant to do this in the regulatory program where health and safety and national defense and security are the statutory criteria. (Emphasis added). 1/

Of similar effect is the following statement by Representative Holifield during the same hearings:

> In concluding this particular part of the hearings on the part of your testimony, I think I speak the sentiments of the full committee that we want to commend you folks for the fine job that you are doing and the careful job you are doing and the fact that you are so concerned with the safety in this field. We realize the economic pressures and other things are always in existence in a situation like this, where we are dealing with this type of very dangerous material. We hope that you will maintain your same fine record of looking at this from the safety standpoint, and let other people worry about the economic standpoint. In other words, don't let the economic standpoint cause you to adulterate your own convictions in regard to safety.
>
> As I said, if we have one large-scale incident in this country, we might just as well wash out the whole atomic power program. You know that this committee amended the law to put your committee in a statutory position. (Emphasis added). 2/

And finally, even more directly in point, is the following testimony by Mr. Muntzing, the AEC Director of Regulation, before the Subcommittee on Reorganization, Research and International Organizations of the Senate Committee on Goverment Operations during hearings on the bill that was to become the Energy Reorganization Act of 1974:

1/ "Indemnity and Reactor Safety", Hearings Before the Subcommittee on Research and Development and the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, 86th Cong., 2d Sess., April 26 and 27, 1960, Part 2, at 275.

2/ Id. at 257.

Senator Ribicoff: Does the AEC ever allow cost to stand in the way of installing the newest safety safeguard devices?

Mr. Muntzing: I would answer that, Senator, by saying that under the Atomic Energy Act, the AEC is charged with assuring the reasonable safety of the facilities it licenses and for that reason, the first decision is made with regard to safety. If it costs additional money, it costs additional money. The cost-benefit relationship is evaluated, however, as part of the environmental impact statements that are prepared.

But the first decision must be what does safety require? And once that decision is made, if it requires backfitting, then it will be done.

Now from time to time we will see that some safety issues are more a matter of probability, and for that reason we will make judgments applicable to future reactors coming along, as opposed to the present ones operating, and as opposed to an early vintage. We will distinguish between them because I think we must be careful that some reactors designed for one mode of operation not be forced into a mode that is technically incompatible.

So we have to be alert to that and I think your question had the implication that possibly that should be watched. And, in fact, we do watch that.

Essentially, however, it is very important that we put safety first. We know that this brings economic penalties, but those are things that must be borne.

Senator Ribicoff: I mean, do the different utilities which build these nuclear reactors know that from time to time there will be additional costs that they may not have anticipated when they first built the plant?

Mr. Muntzing: Yes, they certainly do. And we hear a good deal of complaint about this from time to time. The favorite phrase for this in the business in "ratcheting" and we do have complaints about that. They certainly realize that it does occur. (Emphasis added). 3/

3/ V Leg. Hist. 327.

An early indication of a different "balancing" approach can be found in the section 202 JCAE hearings in 1956. The Commission submitted a report on "Health and Safety" which, while stating in one place that "the overriding objective, however, must be to assure public health and safety," 4/ also contained the following:

> Absolute shielding of radioactivity would be possible only at infinite cost. For atomic energy to be economically feasible, there must be a balance between the cost of protection and the exposure that the populace may be asked to accept with due regard for public health and safety. 5/

And again during the 1960 section 202 JCAE hearings:

> Representative Holifield. This illustrates again, Mr. Chairman, the contradictory position which the Commission finds itself in. They are anxious, and the committee is anxious to have reactors built.
>
> You enter into this problem: If you require 50 miles, you lose a certain amount of electricity in transmission, and therefore it affects the economics. On the other hand, if you put it 10 miles, you may be putting a reactor where it is not safe, at the 10-mile spot, where it would be perfectly safe at 50.
>
> You have this conflict continuously, and you are on both sides of the table. You want to build the reactor, you want to issue the license, we want you to do it, and the plant people want it to be economic, and we want it to be economic, but then comes the conflict. <u>It is a safety conflict</u>. That is why I asked

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4/ "Development, Growth, and State of the Atomic Energy Industry", Hearings before the Joint Committee on Atomic Energy, 84th Cong., 2d Sess., Feb. 7, 8, 15, 16, and 23, 1956, at 173.

5/ Id. at 180.

the question about putting down definite, quantative figures in these regulations.

If you surrender completely on that point, what you are doing, of course, is trading safety for dollars. I am not saying you are doing it, you understand.

* * *

AEC Commissioner Graham. I think you are right, Mr. Ramey. And it is desirable if there can be alternate sites.

I believe sometimes a utility may not have that much leeway. I think Mr. Holifield has summed up the dilemma that we are faced with. Where we do have to look, in the final analysis, at the public health and safety. In all of these there is somewhat of a judgment factor, particularly when you get into the Northern States Power case where you could have gone along pretty well up to one point, but when you add on something new that has not been designed, that changes the whole thing.

I think we all recognize that there are no easy answers. We are struggling with them, and, with your help, we hope to solve them a little better than we are doing. (Emphasis added) 6/

In 1961 a JCAE Staff Study of the regulatory process was submitted to the JCAE. This study contained the following:

The nature of the questions presented.--In the ordinary case of initial licensing by other Federal agencies, the agency's primary responsibility is to adjudicate between competing private applicants, and determine the public interest in granting licenses for a commercial enterprise, as, for example, the operation of an airline, or the construction of a natural gas line or a television station. The primary concern of the AEC, by way of contrast, in considering license applications (or requests for construction of AEC-owned reactors in "parallel proceedings") is the health and safety of the public.

The safety of a nuclear facility will be judged on the basis of considerations of a scientific and technical character. These considerations may not be conclusive, and absolute safety cannot

6/ "Development, Growth, and State of the Atomic Energy Industry", Hearings before the Joint Committee on Atomic Energy, 86th Cong., 2d Sess., February 16-19, 23-25, 1960 at 111-112.

be attained; the only way to avoid all nuclear risks is to build no nuclear facilities. Therefore, if the development of the great natural resource embodied in fissionable materials is to proceed, some risk must be tolerated. Admittedly, the judgment of how much risk is too much is not simply a scientific and technical question, but the scientific and technical factors that determine the nature and extent of the risk must be ascertained and fully understood by both the applicant and the governmental agency. A policy judgment based on incorrect or incomplete assumptions concerning the physical risk of a particular case may be erroneous and even dangerous. (Emphasis added) 7/

During the 1961 JCAE Hearings on "Radiation Safety and Regulation", AEC Commissioner Olson stated:

> Mr. Olson. It may, but only if it should. I think the public must be considered. The affected public is definitely a party in interest here. As Dr. McCullough has told you, there is no such thing as absolute safety. In every case where you build a reactor it is a relative proposition of how much risk is imposed upon the public as opposed to the cost in additional safety features.
>
> Increased safety costs money. On the other hand, we have to strike a balance between those needs. I think the public should be an informed participant here. There may be more interventions, Mr. Chairman, but I am not fearful of that at all because these people do have an interest. (Emphasis added) 8/

Later, during these same hearings, Dr. Thompson of the ACRS agreed with Commissioner Olson:

> Dr. Thompson. That is right. I think there can be no question of the wisdom of these policy decisions to build reactors. Thus, the Commission has, as a policy matter, to be able to encourage the growth of this reactor technology and reactor industry and at the

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7/ "Improving the AEC Regulatory Process", Joint Committee on Atomic Energy Print, March 1961, Vol. I, at 45.

8/ "Radiation Safety and Regulation", Hearings before the Joint Committee on Atomic Energy, 87th Cong., 1st Sess., June 12-15, 1961, at 311-312.

same time they must insure that the industry is safe in all respects. This puts a dual responsibility on them to balance the achievement of their aim of safe nuclear power and at the same time they must be sure not to stifle this budding industry and its goal of economic nuclear power with overregulation. This is indeed a difficult problem. The problem of overregulation is one which is very important in many of our minds.

For a moment, let us consider the fundamental requirements for a reasonable regulations. I believe the Commission is doing this. No regulation, it seems to me, should be promulgated without a real and demonstrable need for such a regulation. No regulation should be written unless it is based upon careful technical studies, reviews of good practice, and careful projections as to the effects of such regulation--not only on safety--but also on the growth and flowering of the industry. Any regulation written should be written in the broadest possible terms to carry out its aims, and in the most nonrestrictive manner possible. Regulations probably should not be written piecemeal but should be written by area (for example, containment, controls, operations, etc.) in such a way that they form an integrated whole. Regulations should at this time avoid numerical limits wherever possible, since such numerical limits would be subject to change as the field progresses.

* * *

Mr. Thompson. I simply wanted to say that I agree with Professor Davis and Commissioner Olson, that the ACRS as such has considered and does consider and has discussed quite freely what we call the problem of "the gain versus the risk."

There is no question that even a body which is completely involved and solely interested in safety must consider this dual role problem to some extent. There is no way of completely divorcing promotion from safety since, if you want absolute safety, you must not build any reactors whatsoever.

If you make the policy decision to build reactors, then you have incorporated a certain amount of risk and you have incorporated certain gains. This policy decision is up to the AEC.

We advise them on the fine structure of that basic decision in regard to the safety of various facilities on the basis again of trying to set up a system which is at least as safe as normal industrial practice. (Emphasis added). 9/

9/ Id. at 348-349, 387.

ENCLOSURE 4

ATTACHMENT 3 - STAFF RESPONSE TO NOVEMBER 15, 1977
MEMORANDUM FROM SECRETARY TO EDO

"adequate protection" is met, 1/ and there is no legal requirement to even address, let alone satisfy, standard review plans, regulatory guides, etc.2/ However, there is a legal requirement that each party to a proceeding be prepared to discuss in some detail the basis for its conclusions that NRC safety regulations are or are not satisfied.

The Staff recognizes that standard review plans and similar Staff documents are not legal requirements, but apparently believes that a sound regulatory program for nuclear power plants must allow for improvements in safety as the technology develops and new information arises from safety research programs and other sources. To date, these

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1/ Maine Yankee Atomic Power Company (Maine Yankee Atomic Power Station), ALAB-159, 6 AEC 1003 (1973). Of course, exemptions from the regulations may be granted pursuant to 10 CFR§50.12. In such cases the NRC review is directed toward compliance with the statutory standard of "adequate protection."

2/ "As their title suggests, regulatory guides are issued for the basic purpose of providing guidance to applicants with respect to, inter alia, acceptable modes of conforming to specific regulatory requirements. But they are not regulations per se and are not entitled to be treated as such, they need not be followed by applicants; and they do not purport to represent that they set forth the only satisfactory method of meeting a specific regulatory requirement." Gulf States Utilities Company (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).

Attachment 3

STAFF RESPONSE TO NOVEMBER 15, 1977

MEMORANDUM FROM SECRETARY TO EDO

Background

In a memorandum from the Secretary to the Executive Director for Operations dated November 15, 1977, (regarding SECY-77-561), the staff was requested to examine the need for modifying Section 50.109(a) of 10 CFR Part 50 on "backfitting" in light of new information or considerations encountered during Phase II of the Systematic Evaluation Program (SEP).

Discussion

On the basis of the staff's experience in the SEP and other actions with operating reactor licensees, we do not see a need to modify Section 50.109(a) of 10 CFR Part 50 at this time. We have found that licensees, when informed of safety deficiencies, have taken appropriate action to resolve the issue including plant shutdown, if necessary. Seldom has the staff had to exercise its authority set forth in the existing regulations to elicit licensee cooperation. On the other hand, the staff does exercise its regulatory authority, e.g., Section 50.54(f) of 10 CFR Part 50 and Orders under Section 2.200 of 10 CFR Part 2, to confirm and make legally binding licensee commitments to resolve safety issues.

The need to modify Section 50.109(a) will continue to be assessed, however, as additional experience is obtained in conducting the SEP program and as a result of implementing improved procedures for determining changes which should properly be classified as providing "substantial Additional Protection" required for the protection of public health and safety.

ENCLOSURE 5

ATTACHMENT 4 - OELD COMMENTS

OELD COMMENTS

The paper accurately describes present Staff review practices. In the discussion of Category II and Category III requirements on pages 8-12, the need for satisfying current staff positions concerning acceptable ways of implementing the regulations is among a number of factors which are described as material to the question whether backfitting should be imposed. Also, the paper proceeds upon a premise that there will or should be NRC safety "requirements" imposed on new plants above and beyond those previously thought to be required for other plants to meet the NRC's safety regulations in 10 CFR Parts 50 and 100. For example, the paper (page 9) appears to suggest that the Staff may decide to impose some new "requirement" (thought to be necessary to comply with existing safety regulations) based upon the "result of value-impact assessments which indicate that the margin of public protection can be measurable increased without facility modification and at little overall cost." Further, the paper as a whole may be read to indicate that the Commission's detailed safety requirements are not set forth in the regulations, but in the standard review plans, regulatory guides, and branch technical positions.

From a legal standpoint and as stated in the paper, standard review plans, regulatory guides, and branch technical positions are not requirements; they reflect only the staff's view of one (and not necessarily the only) way of staisfying the regulations. If a plant complies with the regulations, then (absent some showing of special circumstances) the statutory standard in the Atomic Energy Act of

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safety improvements have been reflected in evolving standard review plans and other similar Staff documents. But such safety improvements do not become mandatory for any class of plants merely by incorporating them in these and other similar Staff documents, and efforts to make the safety improvements mandatory by "reinterpreting" the regulations so as to include them gives rise to serious complications. How does one explain that the regulation means one thing for "old" plants, and something else (probably more stringet) for "new plants," particularly when the regulations on their face do not contemplate any considerations of value-impact, administrative convenience, or other "non-safety" factors.

The dilemma confronting the Staff may be illustrated by the following hypothetical example. A regulatory guide is issued which implements one of the general design criteria (GDC) that has been in effect since 1971. The Guide states as the Staff position that one acceptable way of meeting the GDC is by installation of a new safety feature. The Guide goes on to state that it will not be applied by the Staff to plants which have already received a CP.3/ In the OL review for Unit 1 of hypothetical nuclear plant X, the Staff must take the position that the new feature is not required to meet the GDC if the Staff is to support the issuance of the OL.4/ However, having taken that position

3/ Many Guides contain similar "grandfather clauses."

4/ An exemption from the GDC may be granted under 10 CFR 50.12, but these exemptions are not favored by the Staff.

in the Unit 1 review, how can the Staff contend that the new feature must be installed before the identical Unit 2 on the same site may receive a CP? On the other hand, if the Staff takes the position in the Unit 2 CP review that the CP should not be issued unless the plant design includes this new feature, how can the Staff support issuance of the OL for Unit 1 without the new feature? It seems inescapable that either both units comply with the GDC or both fail to comply. The problem is particularly troublesome when the Unit 2 with the new feature poses less risk to public safety then Unit 1 with it. The so-called "backfit" rule does not come close to solving the problem for operating plants. Moreover, the rule (10 CFR §50.109) does not even apply to construction permit reviews.

The problem, in a nutshell, is that the regulations in Parts 50 and 100, and Commission adjudicatory decisions construing them, generally reflect a "black or white" view of protection of the public health and safety. There is little room for shades of grey, which may move toward "black" or "white" depending on value-impact. In our view, to take the hypothetical example, if the Staff position is that the GDC are met for Unit 1 without the new feature, one clear option that is available which would enable the Staff to insist that the new feature be installed in Unit 2 would be to issue a rule that would impose this new requirement on new plants, but would "grandfather" old plants. As indicated above, most of the new "requirements" are not currently set forth in rules, but are set forth in regulatory guides, standard review plans, etc.