April 24, 1979

MEMORANDUM FOR: Darrell G. Eisenhut, Deputy Director, Division of Operating Reactors

THRU: Steven A. Varga, Chief, Light Water Reactors Branch No. 4, Division of Project Management

FROM: F. J. Williams, Jr., Technical Coordinator, Division of Project Management

SUBJECT: PROBLEMS SURFACED IN REVIEWING RESPONSES TO BULLETIN 05A

The following problems/questions which have been developed to date during the review of licensee responses to Bulletin 05A are provided for your action. The sources of these items were the S. Varga Review Group, OI&E, or other staff members reviewing the responses independently.

Our primary evaluation of these items has been to determine whether they represent open items requiring coverage in the bulletin response safety evaluation being prepared for individual licensees. Where we think that safety evaluation coverage is required we will recommend a time frame for resolution. Beyond that, however, we are cataloging and addressing each item and providing our perception of its significance for staff consideration outside the scope of Bulletin response evaluations.

Potential Bulletin Problems

Although these items do not relate to the safety evaluation of bulletin responses they do relate to concerns expressed with certain Bulletin requirements and therefore should be evaluated on a high priority basis.

1. Bulletins require that licensees ensure that procedures specify continued operation of HPI if automatically initiated due to low pressure. Operation, in most cases, would continue for at least 20 minutes. Injection of this quantity of makeup could lift safety's (most licensees are now operating with PORV blocked) which, according to a recent staff survey, could suffer damage due to water passage. Such damage could result in a LOCA with break size less than that analyzed for ECCS. Michelsons (TVA) report suggests problems in dealing with that size break.

   If not considered initially, this Bulletin requirement should be evaluated in light of the above.

   R. McDermott has data collected to date on this subject.

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DATE ▶						

2. We understand that licensees (specifically, Westinghouse) have expressed concern with the requirement to operate with the pressurizer low level bistables tripped. The protection logic utilized in some plants would then require only a single pressure trip to result in initiation of safety injection. The concern is spurious signals resulting in safety injection.

Purge Valves

The safety evaluations, in the discussion relating to containment isolation and the concern with inadvertent transfer of radioactive gases outside of containment, point out the generic issue of purging through large lines as being an existing staff review item. We recommend that this issue be pursued on an expedited basis.

Cold Shutdown Capability - Main Feedwater Train

The ANO-1 response to the Bulletin requirement for plant shutdown on loss of auxiliary feedwater supply capability indicated that they could not get to cold shutdown conditions utilizing only the main feedwater train without exceeding other Technical Specification limitations. This licensee has two emergency feedwater trains and an auxiliary feedwater train in addition to the main feedwater train. He proposes to reduce power to 5% and to complete the plant shutdown when one of the three trains is returned to service.

We have no concern with the ANO-1 capabilities but it has been suggested that the capabilities of other licensees to achieve cold shutdown on the main feedwater train be investigated to determine whether significant problems exist.

Auxiliary Feedwater Capabilities

Concern has been expressed with the vulnerability of licensees auxiliary feedwater systems to a variety of failure modes. A significant quantity of material has been generated on this subject by R. McDermott. It is our understanding that it has been turned over to you under separate cover.

The remaining items are enclosed. They are listed as received from the Bulletin Response Review Group. We see none which represent open items for the bulletin response, although, for your information, several of the items have been addressed specifically in the safety evaluations and have resulted in requests for additional information or action on the part of the licensee. Such items are identified by asterisk. We have identified them as short term (A) and long term (B) priority items. The short term items generally relate to specific and usually unique

designs. The long term items are generally generic and considered to be part of the ongoing review of the incident. They also include items raised during this exercise but considered to be outside the scope of the TMI event.

The group formulated to consider problems/questions surfaced during these reviews presently consists of F. J. Williams, Jr., H. Rood, H. Silver, R. Scholl, F. Orr and R. McDermott. Time did not permit group consideration of the items discussed in and enclosed with this memo. These items were judged by Williams/Varga.

In the future we anticipate that most items will be provided for your action sometime during the day following receipt by the above group. We will attempt to identify the source of each item, provide clarification where necessary, and identify any individuals either working in that area or with pertinent information. We intend utilizing the transmittal form enclosed.

ORIGINAL SIGNED BY

F. J. Williams, Jr.
Technical Coordinator
Division of Project Management

Enclosures:
As stated

cc: D. Ross
S. Varga
R. Reid
F. Williams

Distribution
Central File
NRR Reading
F. Williams Reading

OFFICE ▶	TC:DPM	LWR#4:DPM				
SURNAME ▶	FJWilliams/bm	SAVarga				
DATE ▶	4/24/79	4/25/79				

SHORT TERM ACTION ITEMS

B&W I/E Bulletin-05A

(B) 1. In all cases, the licensees have used the small break LOCA analysis to bound the event of a loss of feedwater with auxiliary feedwater but the pressure operated relief valve sticking open. This bounds an AAO by an accident analysis. An analysis is needed which demonstrates that such an event will not result in more than a few fuel rod failures.

ok

(B) 2. The motive power for block valves, other essential valves and position indication in the auxiliary feedwater system has not been identified.

(A) 3. The capability of HPI pumps at Davis Besse to handle a TMI-2 event should be reviewed since there are lower head pumps (1600 psi shut-off head). There appears to be more reliance on makeup pumps which isolate on containment isolation.

(A) 4. Since all operators are now being trained on a simulator which has the TMI-2 event program, the adequacy and correctness of the simulation should be reviewed.

(B) 5. The adequacy of the display range of temperature and other process variable indications in the control room should be reviewed.

(B) 6. The level to which the steam generator level is controlled by the auxiliary feedwater system should be checked against that used in the small break LOCA analysis.

LONG TERM ACTION ITEMS

(B) I - I/E notes that feedwater system valve position indication is not redundant/diverse per IEEE 279 and Reg. Guide 1.49. Acceptability should be reviewed for all B&W facilities.

(B) II - I/E recommended that the possibility of damage to safety related equipment upon blowing the rupture disc in the quench tank should be evaluated at all B&W facilities except for Davis Besse which has performed such a review.

(B) III - The transients submitted by the licensees in response to Bulletin 79-06A should all be reviewed. It should be noted that the ICS has been involved in a significant number of those which involved unexpected results. ICS also may preclude independence under automatic control of auxiliary feedwater system.

(B) VI - The ESF's and safety related systems identified in response to Bulletin 79-05A varied from plant to plant. Action should be initiated to standardize the definitions for these terms if we are going to use them for issuance of instructions.

QUESTIONS FROM REVIEW OF BULLETIN 05A

Item 1

Concern 2

(B) 1) The effect of boric acid buildup on reliability of PORV to close should be reviewed. Occurred at Davis Besse.

Item 2

(B) 1. The transients submittal in response to Bulletin 79-05A should be reviewed.

Item 3

(B) 1. Events other than a feedwater event which could lead to voiding in the primary coolant system should be considered and training provided for operators.

(B) 2. A review should be made as to whether all events in Reg. Guide 1.33 are covered in plant procedures.

(B) 3. The effect on the life of the steam generator of a dry-out/refill event should be reviewed together with any precautions that should be considered for refilling a dry steam generator.

(A) 1. ANO-1 - They turn off RCP in the event of main steamline break. This would require reliance on natural circulation. Is this acceptable? It was not mentioned in the responses to other B&W reactors.

(A) 1. ANO-1 - No automatic MSIV closure on containment isolation. Is this acceptable? Not mentioned in other B&W plant responses.

Item 6

(A) 1. Ranche Seco - Leaves cooling water to control rod drives unisolated. Is this acceptable? Others do not mention.

(A) 2. Davis Besse isolates auxiliary feedwater on 24 psig containment pressure.

a) Is isolation at this pressure always acceptable?

b) Can this be a common mode failure for aux. feed.

(A) 3. Arkansas - 1 - containment isolation is initiated by RCS pressure not HPI start. If HPI is mannually started before pressure trip is received, isolation would be by only the 4 psi containment pressure trip. Should containment isolation always be initiated by HPI initiation as Bulletin 05A states. This may cause a lot of spurious isolation.

(B) 4. A radiation signal to isolate containment should be considered.

(A) 5. I/E questions whether the installation of DC valve operators on one train of the auxiliary feedwater system should be delayed until the first refueling outage at Davis Besse as permitted by the license.

(B) 6. A detailed review should be considered of those features isolated and not isolated for CR-3 based on the response provided to Bulletin 05A.

(B) 7. Consideration should be given as to whether systems which are designed as capable of transferring radioactive fluids from containment should all have a radiation monitor.

Item 8

(B) 1. Ranche Seco defines 4.5% aux feed flow as 100% flow. For Crystal River 3 it is 3.5%. The acceptable value for auxiliary feedwater system flow should be reviewed.

(B) 2. Rancho Seco and Arkansas operate with the crossover valve between the two auxiliary feedwater lines normally open. Thus they are not independent with respect to a passive failure. The acceptability of this lineup as it relates to independence of the two trains should be reviewed.

(B) 3. The power supply and coolant supply to the auxiliary feedwater systems should be reviewed for adequacy, reliability and independence. This was not included in the scope at the review of responses to Bulletin 05A.

(B) 4. The signals used to open the auxiliary feedwater system block valves should be reviewed for CR-3 and also the other B&W plants.

* Item 8

(A) It is questioned whether Oconee meets the Bulletin requirement of two independent auxiliary feedwater trains.

Each unit has only one auxiliary feedwater pump. They claim the availability of auxiliary feedwater from the adjacent two units. This requires manual activation (at least 10 minutes required). In addition there is currently a failed block valve to steam generator 1B on unit 1 for which an alternate procedure has been established. It is not clear that water and steam supply system for auxiliary feedwater system is seismically qualified. Are additional requirements necessary for the Oconee station and/or for Oconee Unit 1.

Item 9

(A) 1. The I/E comments on this item for Davis Besse should be reviewed by someone who is familiar with this unique (for B&W plants) containment design.

OTHER

(B) 1. Tech Specs all 3 gpm leakage per pump for ~~decreasing~~ decay heat removal pumps. Is this acceptable? No 1.

(B) 2. A reactor scram upon turbine trip is not provided for B&W reactors. An ICS controlled runback to 15% power is initiated. The question has been raised as to whether a reactor trip should be provided to cause earlier shutdown for this event.

(B) 3. Data on failure rate at PORV's, ICS etc. should be requested.

(A) 4. The bulletins require manual or automatic isolation of containment upon automatic initiation of safety injection. Should there be a requirement for containment isolation upon manual initiation of safety injection.

I. D. _____ Date _____

MEMORANDUM FOR: Darrell G. Eisenhut, Deputy Director, Division of Operating Reactors

THRU: Steven A. Varga, Chief, Light Water Reactors Branch No. 4, Division of Project Management

FROM: F. J. Williams, Jr., Technical Coordinator, Division of Project Management

SUBJECT: PROBLEMS - TMI BULLETIN RESPONSES

Bulletin _____, Plant _____,

Source _____,

Resolution required for safety evaluation _____,

Problem -

Assessment by Problem Group -

Additional information available -

F. J. Williams, Jr.

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

APR 2 6 1979

MEMORANDUM FOR: R. L. Tedesco, Assistant Director for Reactor Safety, DSS

FROM: R. K. Frahm, Reactor Systems Branch, DSS

SUBJECT: OCONEE - SMALL BREAK ANALYSIS

I have performed a cursory review of the Duke Power submittal (W. Parker to H. Denton, dated April 21, 1979) on small break LOCA's for Oconee.

The analysis assumes a small break and failure of emergency feedwater system (EFWS). No ECCS failure is assumed concurrently with failure of the non-ECC EFWS. Operator action is assumed at 20 minutes to manually establish EFWS for those small LOCA's that will not depressurize the primary system to the high pressure injection set point. The limiting break was the largest size for the depressurization of the primary system to just above the HPI actuation set point before the system repressurized.

The analysis only used a blowdown code because the 50.46 criteria were considered to be met if no core uncovery occurred. Therefore, no heatup analyses were performed.

There are some basic questions that must be answered before the subject transmitted can be reviewed and accepted. It does not appear that the analysis took into account any inventory loss when the PORV set point is exceeded. This probably will cause inventory loss without significant depressurization toward the HPI actuation pressure. If a manual action is taken at 20 minutes to initiate the EFWS, what operator actuations are needed to terminate or control EFWS and HPI for the long-term cooling mode? These may occur in a very short time interval. It is not clear that Oconee has the proper instrumentation available to an operator so he can assess the need for a manual action in 20 minutes and the conditions for controlled long-term cooling. Since the pressures are close to HPI initiation and PORV actuation pressures, sensitivity studies on break area and ECCS single failures are needed. The actuation or initiation set points of reactor trip, ECCS, SRV's including initial conditions and other assumptions used in the analysis are needed; as well as criteria used and operator actions needed, for long-term cooling. A discussion of natural circulation is also needed.

Contact: Ronald Frahm, NRR
492-7341

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APR 6 1979

R. L. Tedesco -2-

I suggest we have a meeting with Oconee to get these answers.

Ronald K. Frahm
Reactor Systems Branch
Division of Systems Safety

cc: Z. Rosztoczy
T. Novak
S. Israel
G. Mazetis
B. Sheron
D. Ross

G. Mazetis

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

MAY 9 1979

MEMORANDUM FOR: Karl Kniel, Chief, Core Performance Branch, DSS

THRU: Daniel B. Fieno, Leader, Reactor Physics Section, Core Performance Branch, DSS

FROM: H. J. Richings, Reactor Physics Section, Core Performance Branch, DSS

SUBJECT: DECAY HEAT FOR TMI-2

The enclosed figure gives the most recent results of the calculation of decay heat for TMI-2 and should replace any values used in the past. It differs from previous calculations primarily in having an increased power operating history several months prior to the event (period A in the figure). This results in little change for the earlier shutdown times (several hours) but results in about a 5% increase at several days shutdown, 10% at about a month and up to 30% at a year plus.

The operating history model is based on the information (on boron follow) in the TMI-2 supplementary startup report (3-23-79), plus the assumption of constant 98% power for the last two weeks of operation based on verbal communication with TMI. The model is, of course, a simplification of the detailed history, but should represent integrated full power days with accuracy suitable for the calculation.

The calculations (which are by hand) use the new decay heat information as supplied in the most recent draft of the ANS Standard 5.1 (September 1978). It uses this directly rather than as a correction factor as was done in previous calculations. The calculation used the infinite operation data for operation period D for shutdown times up to 10^7 seconds, and the pulse data for greater shutdown times and for all of operation periods A and B.

The isotope power fractions and actinide production rate were estimated from general information. The isotope effect is small (a few percent or less) and uncertainty in the estimate should have little effect. The actinide power contribution (from U^{239} and Np^{239} decay) is significant, being as much as 24% at one day shutdown (about 15% at several hours and several days). But the uncertainty in its parameters should be small and thus the actinide contribution to uncertainty should be no more than 1 or 2 percent.

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The overall uncertainty is estimated to be 5 to 10% (assuming no explicit error in the hand calculations). The uncertainty in the new ANS data, considering the isotope mix, is about 5% (2σ). The remainder of the uncertainty is from inaccuracies in the operating history model and from estimates used in the hand calculations. These are probably biased in a positive direction. For example, the pulse time was taken as the center of the operating period, which should give a slightly too large result.

There has been no corrections for the possible loss of "volatile" fission products from whatever region of the fuel or reactor system is of interest to the user of the decay heat information. There are regions and shutdown times for which this loss, depending on the temperature, damage, and loss models, could be significant. As a brief reminder of the potential for decay heat change, the following table is presented, giving percentage contributions to decay heat (total, including actinides) from fission products with boiling points less than temperatures which may have existed in TMI-2 fuel in the 1 to 15 hour time frame. These values are estimates (taken from the work of England, Schenter and Scmittroth) to illustrate magnitudes involved with the more prominent "low" boiling point contributors.

% Contribution to Total Decay Heat

		Shutdown Time				
Isotopes	B.P. (°F)	∿1 hr	∿1 day	∿2 wk	∿4 mo	∿3 yr
Noble Gases	--	4	2	1	--	--
Halogens	--	22	14	5	--	--
(Te)	(1800)					
Cs (137)	1300	--	--	--	--	3
Sr (89,90)	2500	--	3	7	8	4
Ba (137,140)	3000	--	4	6	--	12
La (140)	--	--	18	35	1	--

Note that the use of such values requires considerable interpretation and modeling for the specific case. For example, these are members of decay chains and may have been some other element at the time of

high temperature. As an example La^{140} comes primarily from Ba^{140} The La^{140} has a 2 day half life, Ba^{140} a 13 day half life. At one day shutdown the La^{140} contribution comes mostly from atoms which were La^{140} during the high temperature regime and therefore not volatile. At two weeks shutdown, however, the La^{140} comes mostly from atoms which were Ba^{140} during high temperature, and which may in some cases have been "volatile". Similarly, some of the iodine contribution comes from atoms which were tellurium during high temperatures.

H. J. Richings
Reactor Physics Section
Core Performance Branch, DSS

cc: R. Mattson
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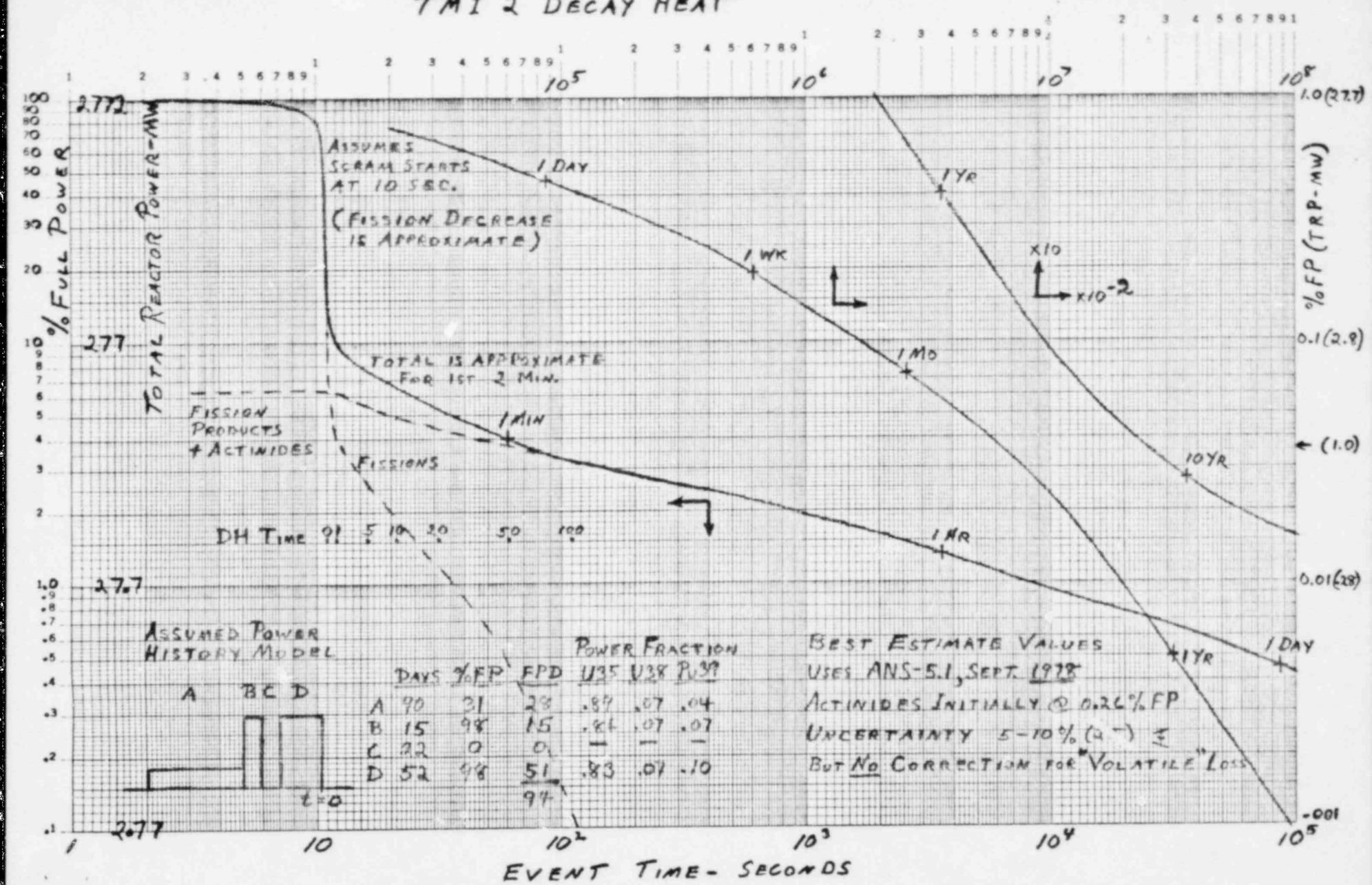
N. Lauben
B. Sheron
V. Stello
D. Davis
T. Telford
S. Weiss
C. Berlinger
R. Wood
D. Crutchfield

TMI 2 Decay Heat



METROPOLITAN EDISON COMPANY A SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

December 2, 1976
GQL 1642

Director of Nuclear Reactor Regulations
Attn: Mr. Robert W. Reid
Chief Operating Reactor Branch #4
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Docket No. 50-289
Operating License DPR 50

As stated in our letter of October 8, 1976 (GQL 1420) the repairs to the dikes at Three Mile Island Nuclear Station (TMINS) were completed on November 24, 1976 prior to the deadline date of December 1, 1976.

Sincerely,

R. C. Arnold
Vice President

RCA:DGM:daf

cc: Mr. D. B. Vassallo
Light Water Reactor Branch No. 4
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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ROUTING AND TRANSMITTAL SLIP		ACTION
1 TO (Name, office symbol or location) L. G. Hulman, Chief, Hydrology-Meteorology Branch, DSE	INITIALS DATE	CIRCULATE COORDINATION
2 B. K. Grimes, Chief, Environmental Evaluation Branch, DOR	INITIALS DATE	FILE INFORMATION
3	INITIALS DATE	NOTE AND RETURN PER CONVERSATION
4 cc: W. Bivins E. Adensam ORPM- J. Zwetzig T. Johnson	INITIALS DATE	SEE ME SIGNATURE

REMARKS

PLANT NAME: THREE MILE ISLAND 1
DOCKET NUMBER: 50-289
DATE OF DOCUMENT: 12/2/76
MAIL ROOM CONTROL NUMBER: 12268
DOCUMENT TITLE: LTR. ADVISING DIKE REPAIRS

REQUESTED DATE FOR REVIEW COMPLETION: —
EEB CONTACT: ADENSAM PHONE NUMBER: 28066

For your information.

E. Adensam - we plan follow-up inspection by HES staff experienced in riprap design and placement. Site visit will be in early 1977 and arranged through ORPM.

Bivins

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE
E. Adensam for B. K. Grimes, Chief, Environmental Evaluation Branch, DOR	12/20/76 PHONE 28066

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 101-11.206

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