PDR-016

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

NRC FOIA REQUEST NUMBER(S): FOIA-87-714

RESPONSE TYPE: FINAL | X PARTIAL

DATE: FEB - 9 1988

DOCKET NUMBER(S) (If applicable)

REQUESTER: Ellyn R. Weiss, Esquire

PART I. – RECORDS RELEASED OR NOT LOCATED (See checked boxes)

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Agency records subject to the request that are identified in Appendix ______ are already available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC
X	Agency records subject to the request that are identified in Appendix D are being made available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and coying at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC
	Agency records subject to the request are enclosed. Any applicable charge for copies of the records provided and payment procedures are noted in the comments section.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______.

PART II.A – INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

X	Certain information in the requested records is being withheld from public disclosure pursuant to the FOIA exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.

Comments

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PDR FOIA
WEISS87-714 PDR

SIGNATURE, DIRECTOR, DIVISION OF RULES AND RECORDS

Donnie H. Grimsley

FREEDOM OF INFORMATION ACT RESPONSE — FOIA NUMBER(S): 87-714 — DATE: FEB - 9 1988

PART II.B — APPLICABLE FOIA EXEMPTIONS

Records subject to the request that are described in the enclosed Appendices E are being withheld in their entirety or in part under FOIA Exemptions and for the reasons set forth below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.5(a) of NRC Regulations.

- [ ] 1. The withheld information is properly classified pursuant to Executive Order 12356 (EXEMPTION 1)
- [ ] 2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (EXEMPTION 2)
- [ ] 3. The withheld information is specifically exempted from public disclosure by statute indicated. (EXEMPTION 3)
  - [ ] Section 141-145 of the Atomic Energy Act which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).
  - [ ] Section 147 of the Atomic Energy Act which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).
- [ ] 4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated. (EXEMPTION 4)
  - [ ] The information is considered to be confidential business (proprietary) information.
  - [ ] The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).
  - [ ] The information was submitted and received in confidence from a foreign source pursuant to 10 CFR 2.790(d)(2).
- [X] 5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency. (EXEMPTION 5)
- [ ] 6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (EXEMPTION 6)
- [ ] 7. The withheld information consists of investigatory records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (EXEMPTION 7)
  - [ ] Disclosure would interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow them to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (EXEMPTION 7(A))
  - [ ] Disclosure would constitute an unwarranted invasion of personal privacy (EXEMPTION 7(C))
  - [ ] The information consists of names of individuals and other information the disclosure of which would reveal identities of confidential sources. (EXEMPTION 7(D))

PART II.C — DENYING OFFICIALS

Pursuant to 10 CFR 9.9 and/or 9.15 of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Rules and Records, Office of Administration, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL: SECRETARY	APPELLATE OFFICIAL: EDO
James A. Fitzgerald	Assistant General Counsel for Adjudications and Opinions	App. E	X	

PART II.D — APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified in that section. Any such appeal must be in writing and must be made within 30 days of receipt of this response. Appeals must be addressed as appropriate to the Executive Director for Operations or to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-87-714

APPENDIX D

RECORDS MAINTAINED IN THE PDR UNDER THE ABOVE REQUEST NUMBER

1. Undated Purpose of Seminar. (27 pages)

2. Undated Enclosure C, Regulatory Analysis. (22 pages)

3. Undated Enclosure C, Backfit Analysis. (14 pages)

4. Undated Enclosure E, Draft Congressional Letter. (2 pages)

5. Undated Enclosure F, Draft Public Announcement. (3 pages)

6. 4/30/86 Memo for Speis from Bernero, subject: Request for Prioritization of Generic Safety Issue - NPSH For ECCS Pumps. (1 page)

7. 5/13/86 Memo for all NRR Employees from Harold Denton, subject: NRR Office letter No. 39, Revision 3 - NRR Procedures for Control and Review of Generic Requirements. (9 pages)

8. 5/18/86 Memo for Murley and Beckjord from Jordan, subject: Loss of Decay Heat Removal Function at Pressurized Water Reactors With Partially Drained Reactor Coolant Systems. (80 pages)

9. 5/21/86 Memo for Speis from Bernero, subject: Prioritization of Generic Issue - Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling. (24 pages)

10. 11/26/86 Region Office Policy Guide No. 0901, Revision 1. (22 pages)

11. 6/10/87 Memo for Beckjord from Murley, subject: Resolution of Generic Safety Issue 93 "Steam Binding of Auxiliary Feedwater Pumps". (3 pages)

12. 7/29/87 Memo for those on attached list from Jordan, subject: Long-Range CRGR Agenda. (15 pages)

13. 8/14/87 Memo for Beckjord from Murley, subject: Resolution of Generic Safety Issue 93, "Steam Binding of Auxiliary Feedwater Pumps". (2 pages)

Re: FOIA- 87-714

APPENDIX E

RECORDS TOTALLY WITHHELD

NUMBER	DATE	DESCRIPTION & EXEMPTION	
1	12/01/86	Martin Malsch to Raymond Fraley re Licensee's Power to "Invoke" the Backfit Rule (8 pp.)	Ex. 5
2	08/06/86	Martin Malsch to Raymond Fraley re Application of the Backfit Rule to the Resolution of USI A-17 (Systems Interaction) (4 pp.)	Ex. 5
3	07/22/86	William Parler to Chairman Zech and Commissioners Roberts, Asselstine and Bernthal re Application of the Backfit Rule to Proposed Amendments to Part 55 (Operator Licensing) (5 pp.)	Ex. 5
4	06/23/86	Martin Malsch to Commission re Application of the Backfit Rule to Proposed Amendments to Part 50 Requirements on Communications Procedures (3 pp.)	Ex. 5
5	06/20/86	Martin Malsch to Commission re Application of Backfit Rule's Compliance exception to Commission's Response to Guard (4 pp.)	Ex. 5
6	06/05/86	Martin Malsch to Chairman Palladino re Application of the Backfit rule to Rules on Record-Keeping and Reporting (5 pp.)	Ex. 5
7	05/21/86	Martin Malsch to Commission re Proposed Backfit Analysis for Proposed Part 20 Revision (5 pp.)	Ex. 5
8	04/30/86	Martin Malsch to Commission re Application of Backfit Rule to Option 2 For Response to Guard (3 pp.)	Ex. 5
9	04/14/86	Martin Malsch to Chairman Palladino re Application of the Backfit Rule to the Proposed Insider Rules (8 pp.)	Ex. 5

Re: FOIA- 87-714

APPENDIX E

RECORDS TOTALLY WITHHELD

NUMBER	DATE	DESCRIPTION & EXEMPTION	
10	02/04/86	Herzel Plaine to Chairman Palladino re Backfit Analysis for the LEU/HEU Rule (SECY-86-17 and SECY-85-284) (7 pp.)	Ex.5
11	01/23/86	Martin Malsch to Commissioner Asselstine re Application of the Backfit Rule to Relaxation of Requirements (4 pp.)	Ex.5
12	12/11/85	Martin Malsch to Commission re Application of Backfit Rule to Part 20 Revision (SECY-85-147) (3 pp.)	Ex.5

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October 20, 1987

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-87-714
Rec'd 10-23-87

Director
Division of Rules and Records
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Freedom of Information Act Request

Dear Sir/Madam,

Pursuant to the federal Freedom of Information Act, I hereby request the following on behalf of the Union of Concerned Scientists:

1. All cost-benefit or value-impact analyses done since September, 1985 in connection with the consideration by NRC staff of generic or site-specific backfits.

2. Any and all lists, compilations or other identifications of potential generic or site-specific backfits under consideration by the NRC staff at any time since September, 1985.

3. Any and all memoranda or other documents since September 1985, from the Committee to Review Generic Requirements ("CRGR") containing requests or direction to the NRC staff to perform, modify or reconsider value-impact or cost-benefit analyses regarding any potential generic or site-specific backfit.

4. Any and all documents containing guidance, criteria or examples used by the NRC in deciding which generic or site-specific backfits are appropriate for cost-benefit analyses under the backfit rule and which are not so appropriate.

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Please call me if you have any questions regarding this request.

Very truly yours,

Ellyn R. Weiss

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General Counsel
Union of Concerned Scientists

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

Ja Cayxin
RCAD
(14)
AEOD - Ornstein

MAY 18 1987

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

FROM: Edward Jordan, Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: LOSS OF DECAY HEAT REMOVAL FUNCTION AT PRESSURIZED WATER REACTORS WITH PARTIALLY DRAINED REACTOR COOLANT SYSTEMS

Introduction

On April 10, 1987 with the reactor coolant system partially drained, the residual heat removal (RHR) pumps at Diablo Canyon 2 were tripped due to vortexing/cavitation. As a result, the plant lost its ability to remove decay heat for 85 minutes. During that 85 minute period, the reactor coolant system (RCS) heated up and bulk boiling was present in the RCS. The loss of the decay heat removal (DHR) function which occurred at Diablo Canyon 2 is one of 37 such events that have been reported to have occurred at U.S PWRs over the last 10 years. Table 1 presents a chronology of these 37 events. These events have the potential for leading to more serious events. Table 2 presents a chronology of NRC and industry actions in the area of DHR system losses.

This memorandum presents a composite set of NRC and industry recommended remedial actions (Enclosure 1) which are based upon the 1985 AEOD case study report C503 dealing with decay heat removal problems for PWR operation, analysis of events subsequent to the case study, and related industry recommendations including INPO SOER 85-4, INPO SER 79-84, and NSAC-52.

In its transmittal letter of the case study to the Director, ONRR, AEOD recommended that the recommendations contained within the report be considered in the resolution of Unresolved Safety Issue A-45. In a response the Director, ONRR believed that the AEOD recommendations were not directly applicable to the resolution of A-45, but instead planned to include them in the resolution of Generic Issue No. 99 "RCS/RHR Suction Line Interlocks." This issue was specifically concerned with loss of the RHR system during cold shutdown or refueling. GI-99 was subsequently modified to evaluate these issues.

Loss of DHR during shutdown is clearly not a new issue. However, the continued occurrence of loss of DHR events, the apparent lack of effectiveness of licensee corrective action in response to past NRC and industry actions, re-assessment of the estimated risk of such events, and the dependence of the risk estimates on human performance, all indicate that prompt regulatory action is now needed to minimize the loss of DHR during periods with a partially drained-primary system and to help assure its rapid recovery should it be lost.

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Discussion

U.S. PWR experience has shown that loss of DHR events have been occurring at a rate of approximately one every 3 to 4 reactor years, and in particular there have been 7 loss of DHR events in the last 2 years when the RCS was in a drained-down condition. Human errors were the root causes of most of those events.

Plants may be subjected to relatively high risks when they undergo partially drained (mid-loop) operations. It is standard procedure for PWRs to drain-down the RCS during shutdown to allow for steam generator maintenance, inspection, and tube plugging, and/or reactor coolant pump seal inspection/maintenance. Factors which contribute to the accident risk during such operations are:

1. The containment is likely not to be isolated (the equipment hatch is often open).

2. Plant design may dictate a very narrow band of allowable RCS levels during drained-down operations (e.g., at Diablo Canyon 2 the range of acceptable RCS levels was only a few inches - the constraints being the elevation of the steam generator nozzle and the suction head required by the RHR pump to prevent air binding.)

3. RCS level measurement during drained-down operations frequently depends upon jury-rigged equipment which is unanalyzed and prone to errors which may exceed the required control band (e.g., at Diablo Canyon the level measurement error was on the same order as the range of acceptable operation - possibility 3 to 12 inches).

4. Generally, procedures for operation during modes 3, 4, and 5 are of an ad hoc nature, scant or even nonexistent. Similarly, procedures for recovery from a loss of DHR are not necessarily well thought out. In addition, operators may not be trained in recovery from a loss of DHR. During shutdown operations, operators may not be fully aware of what equipment is out of service vs. what alternative equipment is available for recovery from a loss of DHR. Operators are not necessarily aware of time available for recovery from loss of DHR events. For example, at Diablo Canyon 2 operators thought that if the DHR function was lost the RCS heatup rate would be 1°F/minute. However the RCS heat up rate was 2.7°F/minute. Therefore, the operators were not expecting bulk boiling to begin as soon as it did.

5. Plants may not have adequate instrumentation available to determine RCS temperature in the reactor during a loss of DHR event. For example, Diablo Canyon 2 had disconnected the core thermocouples prior to the loss of DHR event in anticipation of head removal.

In January 1983, the Electric Power Research Institute's (ERPI) Nuclear Safety Analysis Center (NSAC) published a report on RHR experience at U.S. PWRs (NSAC-52). NSAC-52 provided data on loss of DHR events, as well as recommendations to industry to improve the situation. Similarly, numerous industry reports (e.g., INPO SERs 17-86, 79-84, INPO SOER 85-4) have been written providing information on loss of DHR events, including recommendations for improving the situation. Nonetheless, in recent years, we have been unable to detect a significant industry-wide improvement in DHR loss experiences.

In July 1984 EPRI's Nuclear Safety Analysis Center published NSAC-84, a PRA which addressed operation at Zion 1 and 2 during shutdown. That PRA utilized maintenance and operation records and control room logbook information to estimate equipment availabilities and recovery times. To our knowledge, it was the first comprehensive PRA to address operations at U.S. PWRs during modes 4, 5, and 6. That study shows that the likelihood of a core damage event in non-power modes is comparable to that during power operation.

NSAC-84 notes that, 10 days after shutdown, if the plant is in a drained-down (mid-loop) condition, fuel damage* can occur 4 hours after losing the DHR function. Assuming the same decay heat curve, we conclude that if a loss of DHR were to occur during drained-down operations at Zion 4 days after shutdown, fuel damage could occur within about 80 minutes. (NSAC-84 data indicates that, for some maintenance outages, drain-down of the RCS to mid-loop operation was reached within 4 days from time of shutdown).

Recent experience at other U.S. PWRs has shown that there have been many loss of DHR events during drained-down conditions which were caused by level measurement errors. Many of these events lasted more than 80 minutes. There have also been many similar shorter duration events which resulted in the initiation of bulk boiling (see Table 3).

Review of plant operations during modes 4, 5, and 6 have shown that the key to prevention, mitigation and recovery from loss of the DHR function depend strongly on operators and their ability to perform certain tasks. Because of the strong dependency upon human performance, and the large error bands inherent in quantifying human reliability, the results of risk assessments for operations (estimated to be in the range of 1 to 5 x 10^{-5}/RY) in modes 4, 5, and 6 are subject to large uncertainties. This is noted in both C503 and NSAC-84.

While there may have been over a hundred loss of DHR function events that have been successfully mitigated in the past 10 years at U.S. PWRs, the potential for a serious event is apparent particularly during drained-down conditions. The frequency of such events continues to be several per year even after extensive NRC and industry communications; the estimated probability is in the range of 10^{-5} core damage/RY and there is no assurance that containment would be available; and often the operator, being the key element in loss of DHR function events, is not provided with adequate information (instrumentation), or well thought out procedures, and training.

The cost-benefit analysis for the implementation of remedial actions shows that improvements can be made at modest cost and that the cost/benefit ratio justifies action (Enclosure 2). The total cost range from $13 million to a savings of $321 million. The benefits from averted doses range from 59,000 person-rem to 177,000 person-rem.

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*NSAC-84 assumes that fuel damage occurs when the RCS boils off to the mid-plane of the core.

Conclusion

Adequate justification exists for an appropriate generic communication requiring prompt corrective action to minimize the loss of RHR during periods when the RCS is partially drained. We trust that the composite list of recommended remedial action and the cost-benefit analysis will assist you in preparation of the generic communication.

AEOD is ready to assist your offices in the preparation and implementation of the generic communication.

Original Signed By
E D Jordan

Edward Jordan, Director
Office for Analysis and Evaluation
of Operational Data

Enclosure: As stated

Distribution:
VStello
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*SEE PREVIOUS CONCURRENCES

ROAB:DSP:AEOD	ROAB:DSP:AEOD	*ROAB:DSP:AEOD	*D:DSP:AEOD	*DD:AEOD	*D:AEOD
HOrnstein:md	PLam	JRosenthal	TNovak	CHeltemes	EJordan
5/14/87	5/14/87	5/14/87	5/14/87	5/14/87	5/15/87

Table 1

Chronology of 37 loss of DHR Events Attributed to Inadequate RCS Level

Docket	Plant	Date	Duration	Heatup
344	Trojan	5/21/77	55 min.	Unknown
		3/25/78	10 min.	Unknown
		3/25/78	10 min.	Unknown
		4/17/78	Unknown	Unknown
334	Beaver Valley 1	9/4/78	60 min.	145 - 175°F
366	Millstone 2	3/4/79	Unknown	150 - 208°F
272	Salem 1	6/30/79	34 min.	Unknown
334	Beaver Valley 1	1/17/80	Unknown	Unknown
		4/8/80	35 min.	0
		4/11/80	70 min.	101 - 108°F
		3/5/81	54 min.	102 - 168°F
344	Trojan	6/26/81	75 min.	140 - 150°F
369	McGuire 1	3/2/82	50 min.	105 - 130°F
339	North Anna 2	5/20/82	8 min.	Unknown
		5/20/82	26 min.	Unknown
		5/20/82	60 min.	Unknown
		7/30/82	46 min.	Unknown
338	North Anna 1	10/19/82	36 min.	Unknown
		10/20/82	33 min.	Unknown
369	McGuire 1	4/5/83	Unknown	Unknown
339	North Anna 2	5/3/83	Unknown	Unknown
280	Surry 1	5/17/83	Unknown	Unknown
328	Sequoyah 2	8/6/83	77 min.	103 - 195°F
370	McGuire 2	12/31/83	43 min.	Unknown
		1/9/84	62 min.	Unknown
344	Trojan	5/4/84	40 min.	105 - 201°F
316	DC Cook 2	5/21/84	25 min.	Unknown
368	ANO-2	8/29/84	35 min.	140 - 205°F
295	Zion 1	9/14/84	45 min.	110 - 147°F
339	North Anna 2	10/16/84	120 min.	Unknown
413	Catawba 1	4/22/85	81 min.	140 - 175°F
327	Sequoyah 1	10/9/85	43 min.	<1°F
296	Zion 2	12/14/85	75 min.	~15°
361	San Onofre 2	3/26/86	49 min.	114 - 210°F
382	Waterford 3	7/14/86	221 min.	138 - 175°F
327	Sequoyah 1	1/28/87	90 min.	95 - 115°F
323	Diablo Canyon 2	4/10/87	85 min.	100 - 220°F

Table 2

Chronology of NRC and Industry Actions

A - Chronology of NRC Actions

- USI A-45 (circa - 1980) originally focused on all phases of shutdown for PWRs and BWRs - redirected in 1986, no longer concerned with modes 4, 5, and 6.

- IEB 80-12/IE IN 80-20 requested licensees to review Davis-Besse 2½ hour loss of DHR (4/19/80), and to analyze their own plant's procedures, focusing on redundancy, administrative controls, and technical specifications.

- Generic Letter 6/11/80 - Requested licensees to review St. Lucie's upper head voiding event, amend technical specifications regarding DHR capability.

- IE IN 81-09 discussed Beaver Valley's loss of RHR (drain-down - Tygon).

- NUREG/CR 4005 (Parameter, Inc., 6/85) closeout of IE Bulletin 80-12 - Stated that the issue of DHR operability was closed out at 75% of affected facilities (did not address operation during drained-down conditions, Tygon etc.).

- AEOD Case Study C503 (12/85) - Addressed loss of DHR, included 32 events during drained-down conditions (1976-1984). Indicated that the situation is not improving. Five major recommendations were made, including: reliable level measurement, operator aids, improved procedures for DHR operations, improved procedures/training for recovery from loss of DHR events, improved technical specifications.

- In response to C503, NRR noted it would include the recommendations of C503 in GI-99 (interlocks). To resolve this issue, Brookhaven National Lab is to extrapolate the Zion DHR PRA (NSAC-84) to other PWRs and assess the effect of implementing C503's recommendations. A preliminary report is due in June, 1987. Preliminary results indicate that core melt frequency due to shutdown may be as high as 5.4×10^{-5}/Ry (which is three times higher than NSAC-84's result). Brookhaven's preliminary results indicate that implementing C503's recommendations may reduce the core melt frequency to about half that value.

- IE IN 86-101 12/86 "Loss of DHR due to Loss of Fluid Levels in RCS" discussed events at SONGS 2 (3/86), Zion 2 (12/85), Sequoyah 1 (10/85), and Catawba 1 (4/85). Referenced AEOD Case Study C503, IE IN 81-09, NSAC-52.

- AEOD is presently contacting a foreign country for information on improved level measurement equipment. IRS report #659 (8/86) indicates that a foreign country is testing improved level gauges based on "different physical principles."

Table 2 (Contined)

B - Chronology of Industry Actions

NSAC-52 "Residual Heat Removal Experience and Safety Analysis, Pressurized Water Reactors," January 1, 1983. NSAC-52 reported on 96 loss of RHR events that occurred at US PWRs from 1977-1981. It concluded that procedures are the key to RHR system performance. The report provided many suggestions for improving RHR operations. The suggestions addressed procedures and administrative controls relating to: maintenance and evolution planning; monitoring of reactor vessel level during partially drained operations; control over plant status, maintenance decisions, and outage coordination. In addition NSAC-52 suggested improvements in human engineering and hardware, including: control room indication; audible alarms for low RHR flow; redundant independent RCS level indicating systems; improved instrumentation; and improved data collection for shutdown operations.

INPO SER 79-84 "Loss of Shutdown Cooling Due To Inaccurate Level Indication" - November 1984. The SER discussed numerous events in which the DHR function was lost due to inaccurate RCS level indication and air-binding of the RHR pumps. The SER noted the need for accurate RCS level indication and discussed methods for improving RCS level control. The SER provided comments on the problems associated with using tygon tubing. It also discussed air entrainment and vortexing, and it noted that methods for recovery from loss of DHR cooling should be included in operator training and procedures.

NSAC-84 "Zion Nuclear Plant Residual Heat Removal PRA," July 1985. The report presented a PRA for Zion during modes 4, 5, and 6. It indicated that there were large uncertainties in the estimates of risk for shutdown operations. It concluded that modes 4, 5, 6 may present significant risk relative to operating modes 1, 2, and 3. Core melt frequency for shutdown operations was estimated at 1.8×10^{-5}/Ry.

INPO SOER 85-4 "Loss of Degradation of Residual Heat Removal Capability in PWRs," August, 1985. The SOER noted that probabilistic risk studies had identified loss of RHR as a significant contributor to the potential for core damage. Other areas addressed in the SOER were automatic suction valve closures and loss of RHR pumps. The report stated that analyses had shown that under adverse conditions with a partially drained reactor it is possible to uncover the core within 15 to 30 minutes after loss of DHR due to boiling off the RCS. The SOER noted that controlling RCS level in the "required narrow range is a difficult evolution." It referred to INPO SERs 60-83, and 79-84 which point out the need for reliable RCS level information. The SOER stated that the use of certain procedures, operational controls, training and hardware could have prevented many of the referenced loss of RHR events. Specific recommendations addressed training, operating procedures and emergency procedures relating to drained-down operations.

INPO SER 17-86 "Loss of Shutdown Cooling Flow," May 1986. The SER discussed errors inherent in the tygon tube manometer system that was used for RCS level measurement: gas bubbles in the tubing, lack of procedural controls regarding the routing of the tygon tubing, and the lack of operator awareness of the potential for vortexing. The SER also presented potential corrective actions.

Table 3

Recent Loss of DHR Events Which Occurred During Drained-down Operations Attributed to Inadequate RCS Level Measurement

	Date	Duration	Boiloff initiated
	7/14/86	221 minutes	Yes
	10/16/84	120 minutes	No
	1/28/87	90 minutes	No
2	4/10/87	85 minutes	Yes
	4/22/85	81 minutes	No
	3/26/86	49 minutes	Yes
	8/29/84	35 minutes	Yes

ENCLOSURE 1

Recommended Remedial Action for Reducing Risk from DHR Operations
(Based Upon NRC and Industry Sources)

(1) Licensees should maintain containment integrity to the maximum extent practicable during periods of highest DHR risk (i.e., early stages of shutdown and drain-down operations).

It is recognized that the containment equipment hatch must be open to allow major inspections or repairs during maintenance and refueling outages. Never-the-less licensees should take actions to minimize the risk to the public by: delaying the time of opening the equipment hatch following shutdown, and improving the procedures and training to minimize the time required to re-establish containment integrity during a loss of DHR event. For example, task analyses to integrate equipment hatch opening with the maintenance and refueling operations should be performed. Measures to permit reclosing of the equipment hatch during outages should be developed based on the task analysis.

It should be recognized that operability of the containment purge valves is relied upon during shutdown operations. We also note that during an accident inoperable containment purge valves could compromise containment integrity. Therefore the task analyses should address the containment purge valves and any other valve whose operation is needed to re-establish containment integrity during periods of highest DHR risk.

This item reflects the staff risk analyses based on NSAC-84 and BNL's on going work in support of GI-99. The risk analyses contained in Enclosure 2 focused staff attention on the importance and benefit of containment integrity during shutdown operations.

(2) Licensees should improve planning, coordination, procedures, and personnel training during shutdown to ensure the availability of DHR.

NRC C503, INPO SOER 85-4, NSAC-52, INPO SER 79-84 all recognized the importance of this issue and contained recommendations, suggestions and observations to this effect.

We believe that significant improvements in DHR system availability and reliability can be achieved by focusing on human factors aspects of plant shutdown. Emphasis should be placed on detailed planning of test, surveillance and maintenance activities, and the equipment or system interactions which have frequently caused loss-of-DHR events.

In addition, plant practices regarding the procedures and training of personnel for performance of normal (non-emergency) operations during shutdown should be evaluated. For example: all operations and maintenance staff (licensed and non-licensed) should receive training to assure that they become sensitized to the risks associated with plant shutdown. Emphasis should be placed upon understanding the risks and high vulnerability associated with times of high decay heat rate, drain and fill operations, disabling redundant safety equipment, etc.

(3) Licensees should have a reliable method of measuring and monitoring reactor vessel level during shutdown modes of operation and corresponding technical specification requirements for operability.

NRC C503, NSAC-52, INPO SER 79-84, INPO SOER 85-4 and INPO SER 17-86 all addressed the importance of reliable level instrumentation.

Common industry practice using unanalyzed makeshift devices such as tygon tube sight gages to monitor RCS level during plant shutdown should be modified or discontinued. Reliable, redundant level indication should be required during modes 4, 5, and 6 to ensure availability of trending data, and to warn operators in advance of unacceptably low RCS level. In addition, plant procedures should be modified to assure that the frequency of RCS level monitoring is commensurate with plant status (e.g., as noted in section 4.1 of C503, one plant could have monitored vessel level as infrequently as once every 16 hours, whereas fuel uncovery could occur only a few hours after a loss of DHR). As a minimum, each plant's safety review committee should review the instrumentation and procedures used for RCS level measurement during modes 4, 5, and 6 to ensure that a high level of reliability is achieved.

(4) Licensees should perform a task analysis of DHR operation.

NSAC-52 recognized the need for improvements in human engineering. Performance of a task analysis per se is a specific AEOD recommendation.

We recognize that all DHR losses cannot be totally eliminated by good planning, good procedures, well-trained personnel, etc. We believe that if all licensees would perform human factors analyses of their plant's DHR operations, (including normal and abnormal conditions) and modify their plant practices and man/machine interfaces accordingly, the risks from DHR losses would be significantly reduced. A model to use for such human factors analyses is one used by NRR (Ref. 1). Reference 1 requires licensees to perform specific task analyses, and to integrate instrumentation, alarms and annunciators into normal and emergency procedures for transients and accidents occurring during power operation. Licensees should be required to perform similar reviews for shutdown operations, with emphasis on detection and mitigation of loss-of-DHR events.

The operators should be provided with information outlining the time margins available for recovery from postulated loss-of-DHR events as a function of time from reactor trip for a representative set of DHR loss transients (such as Figure 4 of C502, parametric curves of uncovery time vs. shutdown time). Examples of such transients are: primary system filled at maximum DHR system temperature primary system drained to minimum level and open to the atmosphere; RCS at refueling temperature, etc. Information on time margins available would assist operators in recognizing the potential seriousness of the event, and assist them in choosing appropriate methods for restoring the DHR function.

[1] U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," II.F.2 Instrumentation for Detection of Inadequate Core Cooling, (NUREG 0737), November 1980.

(5) Autoclosure interlocks should minimize loss-of-DHR events.

NSAC-52 and NRC C503 both address interlocks.

In order to prevent inadvertent DHR suction/isolation valve closures (during DHR system operation) it is recommended that NRR consider either requiring the removal of the autoclosure interlocks to the DHR suction/isolation valves, or requiring removal of power to the DHR suction/isolation valves when valve motion is not required. Prior to implementing this recommendation, it is necessary to ensure that there is adequate relief capacity to prevent over-pressurization of the DHR system.

(6) Plant technical specifications should be modified to ensure that the DHR system is available during mode 4 and the early stages of mode 5.

While INPO SER 17-86 acknowledged shortcomings in plant technical specifications, modification of the technical specifications was recommended in AEOD C503. Even though NRR's generic letter of 1980 on DHR addressed DHR system redundancy, plant technical specifications do not require DHR redundancy throughout periods when it is most needed (mode 4 and the early stages of mode 5). Since test, maintenance, and other shutdown activities can be initiated during these periods it is apparent that as a result, a DHR loss could occur at a time when the risk is highest.

We recommend that NRR address the DHR system operating requirements and that plant technical specifications be modified to:

- Ensure all plants have proper shutdown mode definitions (as discussed in sections 4.3 and 5.3 of C503); and

- Ensure that both trains of the DHR system are operable during periods of high decay heat load, i.e.; mode 4 and the early stages of mode 5. (The 1980 generic letter permits one train to be inoperable during this time.)

Since the loss-of-DHR experience has not greatly improved following the issuance of NSAC-52 and NRR's generic letter, we believe that technical specification modifications are necessary to ensure adequate redundancy.

(7) Licensees should analyze the hydraulics associated with drained-down operations.

Level measurement errors observed at Diablo Canyon 2 (April 1987), preliminary information from the AIT assigned to Diablo Canyon, INPO SER 79-84, and SER 17-86 which reported on problems resulting from gas entrainment in tygon measurement equipment etc, all indicate that this issue should be addressed.

Large errors in RCS level measurements have been observed during drained-down operations because of air or gas entrainment which resulted from draining or venting operations, RHR pump vortexing, etc. At many plants the elevations of the steam generator nozzles, pressurizer surge line, reactor hot legs, and reactor coolant pump discharge are such that the there is little margin for measurement error prior to gas entrainment/vortexing. The Diablo Canyon licensee ran tests which indicated gas entrainment caused erratic level measurements. We recommend that licensees perform a detailed hydraulic analyses of their plants' drain-down configuration to assure that the RCS level measuring equipment remains accurate, and operators are aware of the range allowable RCS levels which will assure reliable operation of the RHR pumps.

ENCLOSURE 2

Cost-Benefit Analysis for Proposed NRC Generic Communication Loss of Decay Heat Removal Function in PWRs

I. Introduction

This analysis provides an estimate of the costs and benefits associated with implementing plant and procedural modifications intended to reduce the likelihood of loss of the DHR function in modes 4, 5, and 6 at U.S. PWRs. The analysis was performed based on the NRC's value impact methodology and it employed data which was extrapolated from the most comprehensive probabilistic risk assessment presently available for pressurized water reactors during shutdown (NSAC-84 July 1985). NSAC-84 presented the results of work that was performed by Pickard, Lowe and Garrick to quantify core melt frequency for the Zion nuclear plants during modes 4, 5, and 6. It reviewed operating experience at Zion 1 and 2 during shutdown. It utilized detailed plant and maintenance logbook records to estimate availability and performance of systems and subsystems during modes 4, 5, and 6.

Preliminary results from an NRC contractor working on this issue (Brookhaven National Laboratory), and AEOD's review of recent operating experience indicate that the core melt frequencies appearing in NSAC-84 may be overly optimistic and the value of DHR system improvements recommended by AEOD may be significantly greater than the values listed in this cost-benefit analysis.

II. Analysis

Benefit - averted dose:
Based upon NSAC-84:
Core melt frequency due to operations during shutdown:

$$1.8 \times 10^{-5}/RY$$

Installing a "perfect alarm system" to guarantee the operators are aware of loss of cooling would halve the core damage frequency to $.9 \times 10^{-5}$

The benefit of such a system is quantified as follows:

The equipment hatch is assumed open 1/2 of the time while the plant is shut down. The release is either a category 2 or 3 release.

$$\text{or}\quad \left.\begin{array}{l} 4.8 \times 10^{6} \text{ person rem/accident} \\ 5.4 \times 10^{6} \text{ person rem/accident} \end{array}\right\} \text{avg.} = 5.1 \times 10^{6}$$

$$\text{Averted Dose} = (.9 \times 10^{-5}) \times (.5) \times 5.1 \times 10^{6} = 23 \ \frac{\text{person-rem}}{\text{RY}}$$

Alternatively, per Generic Issue 99's prioritization in NUREG-0933, the core melt from a loss of RHR system would proceed as T_1 MLU of the Oconee RSSMAP analysis. The releases would be as follows:

Category	Probability	Dose (person-rem)
3	.5	5.4×10^6
5	.0073	1.0×10^6
7	.5	2.3×10^3

Averted Dose = $.9 \times 10^{-5} \times .5 \times 5.4 \times 10^6 = 24.3 \frac{\text{person rem}}{\text{RY}}$

PWR population (present plus future plants)

W	55 reactors	1785 RY
B&W	10 reactors	298 RY
CE	15 reactors	485 RY
	80 reactors	2568 RY

Total Averted Dose = $23 \frac{\text{person rem}}{\text{RY}} \times 2568$ RY = 59,000 person-rem averted

Cost:

NRC Labor: from NUREG-0933 Resolution of Generic Issue-99 (Interlocks)

For W only: 8 man-wks = $38,000

Assume CE & B&W require similar efforts 2 x $38,000 = $76,000
Total cost for interlocks = $114,000

Assume a similar effort is needed for level measurement $114,000 but that issue is more complex, and plant specific inspections will be necessary. Each plant will need to be inspected, procedures reviewed, etc.

Assume 300 hrs/plant x 80 plants x $50/hr = $1.2 M

Total NRC labor cost = $1.4 M

Cost:

Industry labor and hardware: from NUREG-0933 Resolution of Generic Issue-99 (Interlocks)

NUREG 0933 estimated resolution of interlocks
At W plants the cost would be $47,200/plant (including hardware, licensing, review, technical specifications, etc).
Assume this cost would exist at all PWRs
80 plants x 47,200/plant = $3.8 M

Assume other hardware would also be used - "perfect alarm," level instrumentation, improved planning, procedures etc. - assume these items cost 2 x as much as the interlocks
(add $7.6 M)

Total industry cost and hardware labor = $11.4M

Benefit: Onsite property damage cost avoidance

$$- \$2 \times 10^{9}/\text{core-melt} \times .9 \times 10^{-5} \frac{\text{core melt}}{\text{RY}} \times 2{,}568 \text{ RY}$$

= - $46 M ; however the present worth assuming 15 yrs avg and 5% discount rate is - $23 M

Benefit: Cost reduction from having shorter outages due to better planning and avoidance of non core-melt loss of DHR events

Shorten outages due to better planning - estimate 3 hours/RY

Avoidance of non core-melt loss of DHR events - frequency of non core-melt losses of DHR is one every 4 RY - assume such losses cause on average a 4-hour delay (extension of outage for a more severe event which includes investigation of the causes of inoperability and the effort required to assure that adequate corrective action is taken) - the net delay = 1/4 x 4 = 1 hr/RY.

$$\text{Total} = 4 \frac{\text{hrs}}{\text{RY}} \times \frac{-\$500{,}000 \text{ replacement power cost}}{\text{per 24-hour day}} \times 2568 \text{ RY}$$

= -$213 M
present worth (15 yr. avg & 5% discount rate)
= -$107 M

Onsite dose and Onsite dose avoidance are neglected.

Uncertainties

1 - BNL has reviewed NSAC-84 and has added one or more accident scenarios and has reexamined the models used for NSAC 84.

BNL has found that the core melt frequency presented in NSAC 84 is low by a factor a 3. If BNL is correct then the benefit from averted dose should be 3 times that listed in this analysis.

$$\frac{69 \text{ person-rem}}{RY} \quad ; \quad 177{,}000 \text{ person-rem total}$$

2 - Time available for successful operator actions to recover from loss of DHR.

NSAC-84 data indicates drain-down during maintenance outages were completed in 4 days or less from time of rod insertion. The decay heat after 4 days is such that the drained-down system could heatup and boiloff to the fuel mid-plane (criteria used for core damage in NSAC-84) in under 80 minutes! However, the loss of cooling event trees assume operator recovery in 1-8 hours with mean error rates of $1x10^{-5}$ to $2x10^{-3}$. These rates appear to be overly optimistic for actions which allow as little as 80 minutes for recovery from a high stress situation especially if the operators have no procedures, no training and inadequate information regarding the status of equipment availability... Recent experience has shown that there have been many severe loss of DHR events during drained-down operation which lasted more than 80 minutes and there have been many shorter duration events which resulted in the initiation of boiloff. For example:

Plant	Date	Duration
*Waterford 3	7/14/86	221 min
North Anna 2	10/16/84	120 min
Sequoyah 1	1/28/87	90 min
*Diablo Canyon 2	4/10/87	85 min
Catawba 1	4/22/85	81 min
*San Onofre 2	3/26/86	49 min
*ANO-2	8/29/84	35 min

3 - NSAC-84 assumes that operator recovery improves with shift change, i.e., if there is a shift change, discovery/recovery from the casualty is assured. This assumption does not agree with recent DHR loss event experience; e.g., on 3/26/86 SONGS 2 had a loss of DHR event which was exacerbated by the shift change.

*Denotes initiation of boiloff.

III. Summary

Cost

NRC labor: $1.4 M
Industry labor + equipment: $11.4 M
Sum: $12.8 M

Property Damage: - $23 M (could be as high as - $69 M)

Replacement Cost: - $107 M (could be as high as - $321 M)

Total Cost

- $321 to $13 million

Benefit

Offsite doses: 59,000* person-rem averted

(could be as high as 177,000 person-rem averted)

Total Benefit

$59 - 177 million