JUN 26

MEMORANDUM FOR: Frank J. Miraglia
Associate Director for Projects

FROM: Richard W. Starostecki, Associate Director
for Inspection and Technical Assistance

SUBJECT: TRANSFER OF CRGR APPROVED 10 CFR 50.54(f)
LETTER FOR ISSUANCE TO LICENSEES CONCERNING
DIABLO CANYON LOSS-OF-RHR EVENT

On Wednesday, June 10, 1987 the Committee to Review Generic Requirements (CRGR) reviewed the NRR proposal to send a 10 CFR 50.54(f) letter to licensees of PWRs. The letter requires those licensees to provide responses to NRR concerns relating to the Diablo Canyon loss of RHR with the reactor coolant loop partially filled. CRGR voted in favor of the NRR proposal and requested certain modifications to the letter. This final version of the 10 CFR 50.54(f) letter has been prepared for your issuance.

Original Signed by
R. W. Starostecki

Richard W. Starostecki, Associate Director
for Inspection and Technical Assistance

Enclosures: As stated

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ATTACHMENT 1

TO: All licensees of operating PWRs and holders of construction permits for PWRs

Gentlemen:

SUBJECT: LOSS OF RESIDUAL HEAT REMOVAL (RHR) WHILE THE REACTOR COOLANT SYSTEM (RCS) IS PARTIALLY FILLED

Pursuant to 10 CFR 50.54(f), the NRC is requesting information to assess safe operation of pressurized-water reactors (PWRs) when the reactor coolant system (RCS) water level is below the top of the reactor vessel (RV). The principal concerns are (1) whether the RHR system meets the licensing basis of the plant, such as General Design Criterion 34 (10 CFR Part 50, Appendix A) and Technical Specifications (TS), in this condition; (2) whether there is a resultant unanalyzed event that may have an impact upon safety; and (3) whether any threat to safety that warrants further NRC attention exists in this condition.

Our concerns regarding this issue have increased over the past several years, and lessons learned from the April 10, 1987 Diablo Canyon loss-of-RHR event require an assessment of operations and planned operations at all PWR facilities to ensure that these plants meet the licensing basis. Study of the Diablo Canyon event has led to identification of unanalyzed conditions that are of significance to safety. Although Diablo Canyon never came close to core damage, and could have withstood the loss-of-RHR condition for more than a day with no operator action, slightly different conditions could have led to an accident involving core damage within several hours. One unanalyzed condition involves boiling within the RCS in the presence of air, leading to RCS pressurization with the potential for ejecting RCS water via cold-leg openings, such as could exist during repair to a reactor coolant pump (RCP) or to a loop isolation valve. The lost water would no longer be available to cool the core, and if makeup water were unavailable, the core could be damaged in a significantly decreased time. The pressurization could also affect the capability to provide makeup water to the core. Other unanalyzed situations are also possible, and occurred at Diablo Canyon (e.g., boiling in the core). The seriousness of this situation is exacerbated by the practice of conducting operations with the equipment hatch removed, and by the lack of procedures that address prompt containment isolation should the need arise.

Loss of RHR and related topics are not a new concern to the NRC staff. This topic has been addressed in numerous communications with the licensee. Yet, these events continue to occur at a rate of several per year. This condition needs to be fully considered in order to ensure compliance with the licensing basis. Therefore, we request that you provide the NRC with a description of the operation of your plant during the approach to a partially filled RCS condition and during operation with a partially filled RCS to ensure that you meet the licensing basis. Your description is to include the following:

(1) A detailed description of the circumstances and conditions under which your plant would be entered into and brought through a draindown process and operated with the RCS partially filled, including any interlocks that could cause a disturbance to the system. Examples of the type of information required are the time between full-power operation and reaching a partially filled condition (used to determine decay heat loads); requirements for minimum steam generator (SG) levels; changes in the status of equipment for maintenance and testing and coordination of such operations while the RCS is partially filled; restrictions regarding testing, operations, and maintenance that could perturb the nuclear steam supply system (NSSS); ability of the RCS to withstand pressurization if the reactor vessel head and steam generator manway are in place; requirements pertaining to isolation of containment; the time required to replace the equipment hatch should replacement be necessary; and requirements pertinent to reestablishing the integrity of the RCS pressure boundary.

(2) A detailed description of the instrumentation and alarms provided to the operators for controlling thermal and hydraulic aspects of the NSSS during operation with the RCS partially filled. You should describe temporary connections, piping, and instrumentation used for this RCS condition and the quality control process to ensure proper functioning of such connections, piping, and instrumentation, including assurance that they do not contribute to loss of RCS inventory or otherwise lead to perturbation of the NSSS while the RCS is partially filled. You should also provide a description of your ability to monitor RCS pressure, temperature, and level after the RHR function may be lost.

(3) Identification of all pumps that can be used to control NSSS inventory. Include: (a) pumps you require be operable or capable of operation (include information about such pumps that may be temporarily removed from service for testing or maintenance); (b) other pumps not included in item a (above); and (c) an evaluation of items a and b (above) with respect to applicable TS requirements.

(4) A description of the containment closure condition you require for the conduct of operations while the RCS is partially filled. Examples of areas of consideration are the equipment hatch, personnel hatches, containment purge valves, SG secondary-side condition upstream of the isolation valves (including the valves), piping penetrations, and electrical penetrations.

(5) Reference to and a summary description of procedures in the control room of your plant which describe operation while the RCS is partially filled. Your response should include the analytic basis you used for procedures development. We are particularly interested in your treatment of draindown to the condition where the RCS is partially filled, treatment of minor variations from expected behavior such as caused by air entrainment and de-entrainment, treatment of boiling in the core with and without RCS pressure boundary integrity, calculations of approximate time

from loss of RHR to core damage, level differences in the RCS and the effect upon instrumentation indications, treatment of air in the RCS/RHR system, including the impact of air upon NSSS and instrumentation response, and treatment of vortexing at the connection of the RHR suction line(s) to the RCS.

Explain how your analytic basis supports the following as pertaining to your facility: (a) procedural guidance pertinent to timing of operations, required instrumentation, cautions, and critical parameters; (b) operations control and communications requirements regarding operations that may perturb the NSSS, including restrictions upon testing, maintenance, and coordination of operations that could upset the condition of the NSSS; and (c) response to loss of RHR, including regaining control of RCS heat removal, operations involving the NSSS if RHR cannot be restored, control of effluent from the containment if containment was not in an isolated condition at the time of loss of RHR, and operations to provide containment isolation if containment was not isolated at the time of loss of RHR (guidance pertinent to timing of operations, cautions and warnings, critical parameters, and notifications is to be clearly described).

(6) A brief description of training provided to operators and other affected personnel that is specific to the issue of operation while the RCS is partially filled. We are particularly interested in such areas as maintenance personnel training regarding avoidance of perturbing the NSSS and response to loss of decay heat removal while the RCS is partially filled.

(7) Identification of additional resources provided to the operators while the RCS is partially filled, such as assignment of additional personnel with specialized knowledge involving the phenomena and instrumentation.

(8) Comparison of the requirements implemented while the RCS is partially filled and requirements used in other Mode 5 operations. Some requirements and procedures followed while the RCS is partially filled may not appear in the other modes. An example of such differences is operation with a reduced RHR flow rate to minimize the likelihood of vortexing and air ingestion.

(9) As a result of your consideration of these issues, you may have made changes to your current program related to these issues. If such changes have strengthened your ability to operate safely during a partially filled situation, describe those changes and tell when they were made or are scheduled to be made.

Enclosure 1 contains insight which experience indicates should be well understood before commencing operation with a partially filled RCS. Your response to this 50.54(f) letter request should encompass the topics contained in Enclosure 1. Additional information is contained in the NRC Augmented Inspection Team report, NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon Unit 2, April 10, 1987." A copy of NUREG-1269 is enclosed.

Your response addressing items 1 through 9 (above) is to be signed under oath or affirmation, as specified in 10 CFR 50.54(f), and will be used to determine whether your license should be modified, suspended, or revoked. We request your response within 60 days of receipt of this letter. This information is required pursuant to 10 CFR 50.54(f) to assess conformance of PWRs with their licensing basis and to determine whether additional NRC action is necessary. Our review of information you submit is not subject to fees under the provision of 10 CFR 170. If you choose to provide a portion of your response in association with your owners group, such action is acceptable.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0011 which expires December 31, 1989. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington D.C. 20503.

Sincerely,

Frank J. Miraglia
Associate Director for Projects
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Enclosures:
As stated

ENCLOSURE 1 TO ATTACHMENT 1

INFORMATION PERTINENT TO LOSS OF RESIDUAL HEAT REMOVAL SYSTEMS

WHILE THE RCS IS PARTIALLY FILLED

Many maintenance and test activities conducted during an outage require lowering the water level in the reactor coolant system (RCS) to below the top of the reactor vessel (RV) or (as is done many times) to the centerline elevation of the RV nozzles. This operating regime is sometimes known as "mid-loop" operation. It places unusual demands on plant equipment and operators because of narrow control margins and limitations associated with equipment, instrumentation, procedures, training, and the ability to isolate containment. Difficulty in controlling the plant while in this condition often leads to loss of the residual heat removal (RHR) system (Table 1).

Although this issue has been the topic of many communications and investigations, such events continue to occur at a rate of several per year.

Recent knowledge has provided additional insight into these events. Although the full implications of this knowledge remain to be realized, our preliminary assessments have clearly established real and potential inadequacies associated with operation while the RCS is partially filled. These include: not understanding the nuclear steam supply system (NSSS) response to loss of RHR, inadequate instrumentation, lack of analyses addressing the issue, lack of applicable procedures and training, and failure to adequately address the safety impact of loss of decay heat removal capability.

The following items are applicable to these conclusions:

(1) Plants enter an unanalyzed condition if boiling occurs following loss of RHR. For example:

(a) Unexpected RCS pressurization can occur.

No pressurization would occur with a water/steam-filled RCS with water on the steam generator (SG) secondary side, because RCS steam would condense in the SG tubes and the condensate would return to the RV. Air in the RCS can block the flow of steam through passages, such as the entrance portion of SG tubes, so that steam cannot reach cool surfaces. Failure to condense the steam causes pressurization in the RCS until the air compresses enough for steam to reach cooled tube surfaces. This pressurization occurred during the April 10, 1987 event at Diablo Canyon since the RCS contained air. Pressure reached 7 to 10 psig, and would have continued to increase if RHR had not been restored. The operators began to terminate the event by allowing water to flow from the refueling water storage tank (RWST) into the RCS. Increasing pressure would have eliminated this option, and would have jeopardized options involving pumps with suction lines aligned (in part) to the RCS.

(b) Water that ordinarily would be available to cool the core might be forced out of the RV, thereby reducing the time between loss of RHR and initiation of core damage.

This is a potential concern whenever there is an opening in the cold leg, such as may exist for repair of reactor coolant pumps (RCPs) or loop isolation valves. Upper vessel/hot-leg pressurization could force the RV water level down with the displaced water lost through the cold-leg opening. A corresponding decrease in level would occur in the SG side of the crossover pipes between the SGs and the RCPs.

This occurrence could be particularly serious if the cold-leg opening were large or if makeup water flow to the RCS were small, as from a charging pump. Cold-leg injection with elevated pressure in the upper vessel may not provide water to the core.

(2) RCS water level instrumentation may provide inaccurate information. There are many facets to this issue. Instrumentation may be indicating

a level that differs from level at the RHR suction line, a temporary instrument may be in use that has no indication or alarms in the control room, and design and installation deficiencies may exist. We have observed the following:

(a) Connections to the RCS actually provide a water level indication up-stream of the RCP location. This water level is higher than the water level at the RHR suction connection because of flow from the injection to the suction locations and because of entering water momentum, which increases level on the RCP side of the cold-leg injection location.

Ingestion of air at the RHR suction connection will result in transporting air into the cold legs; this can potentially increase pressure in the air space in the cold legs relative to the hot legs. Level instrumentation may respond to such a pressure change as though RCS level were changing. In addition, such a pressurization would move cold-leg water into the hot legs and upper RV (or the reverse if a depressurization occurs).

(b) Use of long lengths of small-diameter tubing which can lengthen instrument response time and cause perturbations such as RCS pressure changes to appear as level changes, installation with tubing elevation changes which can trap air bubbles or water droplets, and installation which makes it possible for tubing to be kinked or constricted.

(c) Some installations provide no indication in the control room, yet level is important to safety. Some provide one indication. Others provide diversity via different instrumentation, but do not provide independence because they share common connections.

(d) Tygon tube installations faintly marked at 1-foot intervals that have no provision for holding the tube in place.

(e) Instrumentation in which critical inspections were not performed after the installation.

(f) Instrumentation in which no provisions were made to ensure a single phase in connection tubing or that tubing was not plugged.

(g) Use of instrumentation without performing an evaluation of indicated RCS level behavior and instrument response.

(3) Vortexing and air ingestion from the RCS into the RHR suction line are not always understood, nor is NSSS response understood for this condition.

(a) On April 10, 1987, Diablo Canyon operators reduced indicated RCS level to plant elevation 106' 6" immediately after steam generator tubes drained, and indications of erratic RHR pump current were observed. Restoring the RCS level to 106' 10" was reported to have eliminated the problem. RHR operation was terminated a few hours later at an indicated level of 107' 4" because the operators observed erratic RHR pump current indications. The licensee later reported that vortexing initiated under those conditions at 107' 5-1/2", and was fully developed at 107' 3-1/2". Procedures in place at the time of the event indicated the minimum allowable level to be 107' 0" (the hot- and cold-leg centerline elevation) or 107' 3".

(b) Additional phenomena appear to occur under air ingestion conditions. These include:

- RHR pumps at Diablo Canyon were reported to handle several percent air with no discernible flow or pump current change from that of single-phase operation.

- A postulate is that air in the RHR/reactor coolant system can migrate or redistribute, and thus cause level changes which are at variance with those one would expect. This is a possible explanation for observed behavior in which lowering the RCS water level is followed by a level increase. Water in the RHR appears to be replaced by air. Similarly, an increase in RCS water level that is followed by a decreasing level may be due to voids in the RHR system being replaced by RCS water. Failure to understand such behavior leads operators to mistrust level instrumentation and to perform operational errors.

(c) Operators typically will start another RHR pump if the operating pump is lost. Experience and an understanding of the phenomena clearly show that loss of the second pump should be expected. The cause of loss of the first pump should be well understood and normally should be corrected before attempting to run another RHR pump.

(d) Typical operation while the RCS is partially filled provides a high RHR flow rate, which may be required by TS, but which may be unnecessary under the unique conditions associated with the partially filled RCS. Air ingestion problems are less at low flow rates.

(4) Only limited instrumentation may be available to the operator while the RCS is partially filled.

(a) Level indication is many times available only in containment via a Tygon tube. Some plants provide one or more level indications in the control room, and additionally provide level alarms.

(b) Typically, RHR system temperature indication is the only temperature provided to the operators. Loss of RHR leaves the operator with no RCS temperature indication. This can result in a TS violation, as occurred at Diablo Canyon on April 10 when the plant entered Mode 4, unknown to the operators, with the containment equipment hatch removed. It also resulted in failure to recognize the seriousness of the heatup rate, or that boiling had initiated.

(c) RHR pump motor current and flow rate may not be alarmed and scales may not be suitable for operation with a partially filled RCS.

(d) RHR suction and discharge pressures may not be alarmed and scales may not be suitable for operation with a partially filled RCS.

(5) Licensees typically conduct operations while the RCS is partially filled, the containment equipment hatch has been removed, and operations are in progress which impact the ability to isolate containment. Planning, procedures, and training do not address containment closure in response to loss of RHR or core damage events. This is inconsistent with the sensitivity associated with partially filled RCS operation and the history of loss of RHR under this operating condition.

(6) Licensees typically conduct test and maintenance operations that can perturb the RCS and RHR system while in a partially filled RCS condition. The sensitivity of the operation and the historical record indicate this is not prudent.

Table 1

37 LOSS-OF-DHR* EVENTS ATTRIBUTED TO INADEQUATE RCS LEVEL

Docket	Plant	Date	Duration	Heatup
344	Trojan	05/21/77	55 min.	Unknown
		03/25/78	10 min.	Unknown
			10 min.	Unknown
		04/17/78	Unknown	Unknown
334	Beaver Valley 1	09/04/78	60 min.	145-175°F
366	Millstone 2	03/04/79	Unknown	150-208°F
272	Salem 1	06/30/79	34 min.	Unknown
334	Beaver Valley 1	01/17/80	Unknown	Unknown
		04/08/80	35 min.	None
		04/11/80	70 min.	101-108°F
		03/05/81	54 min.	102-168°F
344	Trojan	06/26/81	75 min.	140-150°F
369	McGuire 1	03/02/82	50 min.	105-130°F
339	North Anna 2	07/30/82	46 min.	Unknown
338	North Anna 1	10/19/82	36 min.	Unknown
		10/20/82	33 min.	Unknown
369	McGuire 1	04/05/83	Unknown	Unknown
339	North Anna 2	05/03/83	Unknown	Unknown
		05/20/82	8 min.	Unknown
			26 min.	Unknown
			60 min.	Unknown
280	Surry 1	05/17/83	Unknown	Unknown
328	Sequoyah 2	08/06/83	77 min.	103-195°F
370	McGuire 2	12/31/83	43 min.	Unknown
		01/09/84	62 min.	Unknown
344	Trojan	05/04/84	40 min.	105-201°F
316	DC Cook 2	05/21/84	25 min.	Unknown
368	ANO-2	08/29/84	35 min.	140-205°F
295	Zion 1	09/14/84	45 min.	110-147°F
339	North Anna 2	10/16/84	120 min.	Unknown
413	Catawba 1	04/22/85	81 min.	140-175°F
327	Sequoyah 1	10/09/85	43 min.	<1°F
296	Zion 2	12/14/85	75 min.	~15°
361	San Onofre 2	03/26/86	49 min.	114-210°F
382	Waterford 3	07/14/86	221 min.	138-175°F
327	Sequoyah 1	01/28/87	90 min.	95-115°F
323	Diablo Canyon 2	04/10/87	85 min.	100-220°F

* Decay heat removal

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 27, 1987

NRC INFORMATION NOTICE NO. 87-23: LOSS OF DECAY HEAT REMOVAL DURING LOW REACTOR COOLANT LEVEL OPERATION

Addressees:

All holders of an operating license or a construction permit for pressurized-water reactor facilities.

Purpose:

This notice provides information regarding the loss of decay heat removal capability at pressurized water reactors resulting from the loss of RHR pump suction during plant operations with low reactor coolant levels. It is expected that recipients will review this information for applicability to their reactor facilities and consider actions, if appropriate, to prevent similar problems. Suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On April 10, 1987 the Diablo Canyon Unit 2 reactor experienced a loss of decay heat removal capability in both trains. The reactor coolant system had been drained down to the mid-height of the hot-leg piping in preparation for the removal of the steam generator manways. During the 85 minute period that the heat-removal capability was lost, the reactor coolant heated from 87° F to boiling, steam was vented from an opening in the head, water was spilled from the partially unsealed manways, and the airborne radioactivity levels in the containment rose above the maximum permissible concentration of noble gases allowed by 10 CFR 20. The reactor, which was undergoing its first refueling, had been shut down for seven days at the time and the containment equipment hatch had been opened.

Erroneous level instrumentation, inadequate knowledge of pump suction head/flow requirements, incomplete assessment of the behavior of the air/water mixture in the system and poor coordination between control room operations and containment activities all contributed to the event. Under the conditions that existed, the system that indicated the level of coolant in the reactor vessel read "high" and responded poorly to changes in the coolant level. In addition, the intended coolant level, established for this operation, was later determined to be below the level at which air entrainment due to vortexing was predicted to commence. At the time of the event, the plant staff believed that the coolant level was six inches or more above the level that would allow vortexing.

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The event began at about 8:43 pm, when a test engineer in preparation for a planned containment penetration local leak rate test, began draining a section of the reactor coolant pump leakoff return line, which he believed to be isolated. However, because of a leaking boundary valve, this action caused the volume control tank fluid to be drained through the intended test section to the reactor coolant drain tank. The control room operators, who were not aware that the engineer had begun conducting the test procedure, increased flow to stop the fluid reduction from the volume control tank. A few minutes later the operators were informed that the reactor coolant drain tank level was increasing but they could not determine the source of the leakage. Although the actual level of coolant in the reactor vessel was apparently dropping below the minimum intended level, the indication of level in the vessel remained within the desired control band. At 9:25 p.m. the electrical current of the active RHR pump (No. 2-2) was observed to be fluctuating. The 2-1 pump was started and the 2-2 pump was shut down. However, the current on the 2-1 pump also fluctuated, so it was immediately shut down as well.

The operators did not immediately raise the water level in the reactor because they still did not know either the source of the leakage, the true vessel level, or the status of the work on the steam generator manways. Operators were sent to vent the RHR pumps. One pump was reported to be vented at 10:03 p.m. At 10:21 p.m. an attempt was made to start this RHR pump, but the current fluctuated and it was shut down again. During this period the operators did not know the temperature of the coolant in the reactor vessel because the core exit thermocouples had been disconnected in preparation for the planned refueling. By 10:30 p.m. airborne activity levels in the containment were increasing and personnel began to evacuate from the containment building.

At 10:38 p.m. when the operators learned that the steam generator manways had not been removed, action was initiated to raise the reactor vessel water level by adding water from the refueling water storage tank. About 10 minutes later the test engineer identified the source of the leakage and stopped it. By 10:51 p.m., the vessel level had been raised sufficiently to restart one of the RHR pumps. The indicated RHR pump discharge temperature immediately rose to 220° F. At this time the reactor vessel was slightly above atmospheric pressure and steam was venting from an opening in the reactor vessel head.

Discussion:

The NRC has documented numerous instances in the past where decay heat removal systems have been disabled because pump suction was lost while the plant was being operated at low reactor coolant water levels. IE Information Notice 86-101 describes four such events that occurred in 1985 and 1986. NRC Case Study Report AEOD/C503 describes six such events that occurred in 1984, five that occurred in 1983, and seven that occurred in 1982. IE Information Notice 81-09 described an event at Beaver Valley in March 1981. The case study report further indicates that a total of 32 such events occurred from 1976 through 1984. The documentation includes descriptions of a total of 23 events that have occurred since 1981 involving loss of decay heat removal capability resulting from a loss of pump suction while operating at reduced water levels.

For all but four of these 23 events the primary cause of the loss of pump suction and loss of decay heat removal capability was attributed to incorrect, inaccurate, or inadequate level indication. Two events were attributed to loss of pump suction because of vortexing brought on by the simultaneous operation of both pumps. In many of these events procedural errors were also a contributing factor. In at least nine of the cases, the redundant pump was lost because air was entrained when the operators, not understanding the cause of the problem, switched to the second pump. There are repeated references to difficulties in getting the pumps vented quickly after air binding had occurred and to the operators' inability to take immediate action to raise reactor vessel levels until the safety of personnel working on the primary systems could be assured. The length of time that decay heat removal was completely lost varied from eight minutes to two hours and averaged almost an hour. In at least three previous cases, boiling is known to have occurred.

A number of actions have been recommended previously to prevent the loss of RHR pump suction during low vessel level operations. These include:

Providing accurate level instrumentation designed for reduced vessel water level operations.

Providing alarms in the control room for low decay heat removal flow and low water level.

Including in the procedures specific requirements for frequent monitoring and strict limits on level.

Considering in the procedures the possibility of vortex formation and air entrainment, including a precaution against starting a second RHR pump until the cause of the loss of the first pump is determined and corrective actions have been taken.

Training the operators on the correlation between water level and pump speed at the onset of vortexing and air entrainment.

Careful planning, coordination, and communication with control room personnel regarding all ongoing activities which could affect the primary system inventory.

The NRC review of the Diablo Canyon event indicated that vortexing and air entrainment may occur at higher water levels than anticipated. In addition, operation at mid-hot-leg levels can lead to unanticipated conditions which may not have been adequately considered in instrumentation design and procedure preparation.

The NRC staff's initial assessment of this event has identified the potential for a significant loss of decay heat removal capability both from a total loss of the RHR system and from a loss of the steam generator heat sink due to air blanketing of the steam generator tubes. Correct operator actions then become critical for plant recovery.

NRC communications in the past have expressed serious concern with failures to maintain adequate decay heat removal capability. IE Information Notice 81-09 pointed out that loss of shutdown cooling capability had been found to be a potentially significant contributor to the total risk. AEOD/C503 and other sources indicate that the time available to restore shutdown cooling before core uncovery can occur is not necessarily large. At four days after shutdown from long-term power operation, with the vessel drained down to the RHR suction loss level, the vessel water can heat to the boiling point in about 1/2 hour. Under such conditions boiloff to the core uncovery level can occur in less than two hours.

Following the loss of decay heat removal capability on April 10, 1987 at Diablo Canyon, PG&E took a number of actions to prevent loss of RHR suction during low level operation and to improve recovery should such a loss occur. These actions included the following:

Evaluation of the reactor vessel level indicating system to determine the level at which vortexing would occur and the effect of vortexing on the level measurement.

Enhancements of the instrumentation to include accurate level measurement, alarm capability and core exit temperature measurement during low level operation.

Enhancement of procedures to include requirements for verifying proper RHR pump suction before starting the second RHR pump. Also included are precautions specifying minimum vessel levels as a function of RHR flow.

Improvements in work planning, control and communication to include a restriction of the work scope to items that do not have the potential to reduce RCS inventory.

Improvement of operator training including a discussion of the potential causes of RHR flow loss, as well as recovery procedures.

The NRC is currently considering additional generic action on this issue.

This information notice requires no specific action or written response. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Donald C Kirkpatrick, NRR
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Attachment:
1. List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
INFORMATION NOTICES 1987

Information Notice No.	Subject	Date of Issuance	Issued to
87-22	Operator Licensing Requalification Examinations at Nonpower Reactors	5/22/87	All research and nonpower reactor facilities.
87-21	Shutdown Order Issued Because Licensed Operators Asleep While on Duty	5/11/87	All nuclear power facilities holding an OL or CP and all licensed operators.
87-20	Hydrogen Leak in Auxiliary Building	4/20/87	All nuclear power facilities holding an OL or CP
86-108 Sup. 1	Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion	4/20/87	All PWR facilities holding an OL or CP.
86-64 Sup. 1	Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures.	4/20/87	All nuclear power facilities holding a CP or OL.
85-61 Sup. 1	Misadministrations to Patients Undergoing Thyroid Scans	4/15/87	All licensees authorized to use byproduct material
87-19	Perforation and Cracking of Rod Cluster Control Assemblies	4/9/87	All Westinghouse power PWR facilities holding an OL or CP
87-18	Unauthorized Service on Teletherapy Units by Non-licensed Maintenance Personnel	4/8/87	All NRC licensees authorized to use radioactive material in teletherapy units
87-17	Response Time of Scram Instrument Volume Level Detectors	4/7/87	All GE BWR facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

ROUTING AND TRANSMITTAL SLIP Date 7/13/87

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
1. Jean Rathje		
2. Public Document		
3. Room		
4.		
5.		

	Action		File		Note and Return
	Approval		For Clearance		Per Conversation
	As Requested		For Correction		Prepare Reply
	Circulate		For Your Information		See Me
	Comment		Investigate		Signature
	Coordination		Justify		

REMARKS

Please put the attached document on public display.

cc: I. Bailey
DCS-042

[accession # 8707060033]

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
Hazel Smith	P-433
	Phone No. 28972

5041-102 ★U.S.GPO:1985-0-481-274/20024 **OPTIONAL FORM 41 (Rev. 7-76)** Prescribed by GSA FPMR (41 CFR) 101-11.206

Please change DCS to PDR.

HS 7/13