APPENDIX A
(RECORDS BEING RELEASED IN THEIR ENTIRETY)

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	7/24/01	E-Mail from G O'Dwyer to J Dyer, Subject: ECCS Suctions DPV and Fwd: RE: BWR Mark I ECCS Concerns (1 page); 7/23/01 E-Mail from G O'Dwyer to J Grobe, Subject: BWR Mark I ECCS Concerns (3 pages)
2.	8/17/01	E-Mail from G O'Dwyer to J Dyer, Subject: My DPV Supplement (3 pages); 3/31/82 Memorandum for C Michelson from S Rubin, Subject: Engineering Evaluation - Potential for Air Binding or Degraded Performance of BWR RHR System Pumps During the Recirculation Phase of a LOCA (8 pages) **PDR ACCESSION NO. 8204270464**, NRR 6/19/01 Conference Call (1 page); NUREG/CR-2792, pages 74-76, **PDR ACCESSION NO. 8210050360**
3.	8/23/01	Memorandum to S Reynolds from J Dyer, Subject: Ad Hoc Review Panel for Differing Professional View Regarding BWR ECCS Suction Concerns (1 page); 8/17/01 E-Mail from G O'Dwyer to J Dyer, Subject: My DPV Supplement (3 pages); 3/3/1/82 Memorandum for C Michelson from S Rubin (8 pages) **ACCESSION NO. 8204270464**, NRR 6/19/01 Conference Call (1 page); NUREG/CR-2792, pages 74-76, **PDR ACCESSION NO. 8210050360**
4.	9/6/01	E-Mail from G O'Dwyer to S Reynolds, Subject: An Addition to my July 24 DPV (1 page)
5.	9/17/01	E-Mail from G O'Dwyer to K Riemer, Subject: Fwd: I would like to change my DPV recommendation (1 page); 9/13/01 E-Mail from J Dyer, Subject: I would like to change my DPV recommendation (1 page)
6.	9/20/01	E-Mail from D Schrum to K Riemer, Subject: Mr. O'Dwyer's Nuclear Safety Concerns (2 pages)
7.	11/30/01	Memorandum to G O'Dwyer from S Reynolds, Subject: Differing Professional View: ECCS Suction Concerns (2 pages)
8.	1/11/02	E-Mail from G O'Dwyer to S Reynolds, Subject: Response to your 11-30-01 questions (2 pages)
9.	1/29/02	E-Mail from D Schrum to K Riemer, Subject: Riemer Questions (2 pages)
10.	1/31/02	E-Mail from D Schrum to S Reynolds, Subject: Mr. O'Dwyers DPV Questions (3 pages)
11.	2/1/02	E-Mail from D Schrum to K Riemer, Subject: Mr. O'Dwyers DPV Questions (1 page)

12.	4/11/02	E-Mail from G O'Dwyer to D Schrum, Subject: DPV Panel Recommendation Memo (6 pages)
13.	4/8/02	Memorandum to J Dyer from S Reynolds, Subject: Ad Hoc Review Panel Recommendation Regarding a Differing Professional View on BWR ECCS Suction Concerns (6 pages)
14.	4/29/02	Memorandum to G O'Dwyer from J Dyer, Subject: Resolution of Differing Professional View on BWR ECCS Suction Concerns (1 page); 4/8/02 Memorandum to J Dyer from S Reynolds (6 pages)
15.	5/10/02	Memorandum to S Newberry from J Grobe, Subject: Potentially Generic Safety Issue - BWR ECCS Suction Concerns (2 pages)
16.	5/28/02	Memorandum to F Eltawila from S Newberry, Subject: Potentially Generic Safety Issue - BWR ECCS Suction Concerns (1 page) 5/10/02 Memorandum to S Newberry from J Grobe (2 pages)
17.	Undated	"Potential for Air Biding or Degraded Performance of BWR Residual Heat Removal System Pumps During the Recirculation Phase of a Loss-Of-Coolant Accident (1 page)
18.	--	NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, pages xi, xii, 11 and 12

From: Gerard O'Dwyer, R3
To: Jim Dyer, R3
Date: 7/24/01 9:44AM
Subject: ECCS suctions DPV and Fwd: Re: BWR Mark I ECCS Concerns

Since I was informed yesterday that NRR is not working on my non-loss of offsite power LOCA ECCS pumps' gas ingestion concerns and I was led to believe NRR experts were working comprehensively and continuously on my concerns since before June 19 and for other reasons, I am respectfully submitting this Differing Professional View that all operating reactors with Mark I and II containments should be shutdown until hardware repairs are made to prevent the now certain ingestion of gas during some LOCAs which would result in subsequent gasbinding of the LPCI pumps which would allow containment failure and the release of excessive radioactivity to the public. I also request that all the attached emails be made part of this DPV. I request that I be relieved of all routine duties and assigned exclusively to ensure that these problems are corrected. Yesterday I was informed that NRR acknowledged that my claims may be valid, NRR cannot find anything indicating that my concerns have already been proven groundless, NRR cannot even find anything indicating that my concerns have already been analyzed, and that it would take significant time to determine if my concerns had been addressed or not.

CC: Claudia Craig; Darrell Schrum; James Caldwell; John Grobe; John Jacobson; Ronald Langstaff; Roy Caniano; Satwant Bajwa; Tae Kim; Tony D'Angelo

All

From: Gerard O'Dwyer, R3
To: John Grobe, R3
Date: 7/23/01 11:06AM
Subject: Re: BWR Mark I ECCS Concerns

I have to respectfully disagree that the main technical issues of my concerns have been addressed. One of my main concerns has always been and still is the non-loss of AC LOCA. On the June 19 conference call Tony D'Angelo (to his eternal credit for honesty) admitted that he had not previously thought of the non-loss of AC scenario and said he would think about it and get back to me in a few days. Tony just informed me by phone that he told Kim and his managment that he could not dismiss my concerns out of hand, there may be problems, he cannot find anything indicating that my concerns have already been analyzed, and that it would take significant time for him to determine if they had or not. He also stated that NRR decided after the call that region III should send a TIA in order to justify that effort. No one has responded to me about motor starting times and I do not know who was tasked to respond to me about that. In my June 26 e-mail to Tony, I asked for a copy of AEOD report E218-1982. I respectfully disagree with the NRR staff as stated in attached statement that the report provided a good description of the phenomena in question. The report only addresses the ingestion of gas from dissolved gases and completely omits one of my main concerns of ingestion of the gases coming from the downcomer. I also contend that there will be significant amounts of gases coming from the downcomer even after blowdown and this has not been addressed. I also contend that those gases will be pulled into the LPCI suctions by vortexing and cause the LPCI pumps to gasbind. Ingestion by vortexing will be worst when the reactor pressure drops (during a small or medium break LOCA) sufficiently to allow higher flows through the pumps. Even worse is a Mark I containment during a medium break LOCA with an ECCS header where the velocity of the water in the connection from the torus will be increased substantially by the HPCI pump. Gas ingestion may have been the reason that the HPCI pump was inoperable during the June 5, 1970 small break LOCA at Dresden unit 2. I have not been able to confirm the reason of the HPCI inoperablity because the Region's microfiche records appear to be incomplete concerning the event, e. g., the July 27, 1970 response from ComEd to the NRC's 15 questions about the incident seemed to be missing. I have also not been able to follow up as I feel my concerns merit because of lack of time in the office. In your June 12 email you stated that my priorities were to complete training, prepare for, conduct and document my inspection assignments and I can work on my concerns as I have time during normal work hours therefore I have arranged my priorities as follows.

Priority: 1) Routine HX Inspections. Prep for Fermi HS inspection. Done - Request risk info for HX selection. Schedule entrance and exits. Flint said he'd tell me 7/20. Done - Associate Fermi HS inspection with resident report number. WORKING - Fermi Inspection plan. WORKING - Fermi travel planner. Prep for Prairie Island HS inspection because it requires so much lead time. WORKING - review previous HS inspection. Requested PRA info.

2) Dresden Dam Open item followup: Review Question 10 of the Request for Additional Information (RAI) on Dresden and QC Extended Power Uprate 10 which is about the Dresden Dam failure seismic event issues. This was suggested by Rossbach. Done - JMJ told me to request when Rossbach expects an answer to Dresden's RAI. I sent an email.

3) Byron TIA followup. WORKING - Call Kemker and check status of dewatering. WORKING - Writeup closeout for URI.

4) ECCS concern. Check when pressure builds in containment. DPV? JMJ told me to ask Deangelo when they are going to get back to us about the things they said they would: 1) sending the video tape, 2) NRR providing inspection guidance for inspectors to ensure new suction strainers do not violate the exclusion zone, 3) start time for LPCI pumps, 4) D'eangelo's thoughts on my scenario of LOCA but no LOOP. I left vmail for tony because nobody answered.

Schedule annual leave.

WORKING - Write DPV about RHR drawings.

Write email that my number one priority should be following up on my concerns and that the HX inspections should be delayed or someone else should do them. It is more important to followup on actual issues rather that just doing routinely scheduled inspections where we have no reason to believe there is a problem.

Practice starfire.

Do an IDP.

If these priorities are not correct please tell me how to reorder them. If they are correct, then I request that I be taken off my routine HX inspections so that I can have time to followup on my more important ECCS concerns.
>>> John Grobe 07/19/01 03:49PM >>>
Gerry,

NRR sent me the enclosed response documenting the results of the conference call we had on your questions regarding ECCS system function. There were three remaining actions from the call.

One was for NRR technical staff to provided us a video and possibly a training session regarding post-blowdown hydraulic response for our technical staff at a future inspector seminar . John has that as an action item and we will be coordinating that with DRP for the December seminar.

As they discussed during the call, it appears that the main technical issues have already been addressed. Please pursue those last two questions with NRR at your earliest opportunity. Let John know as soon as you get responses.

Thank You.

Jack

CC: Claudia Craig; Darrell Schrum; Dina Sotiropoulos; James Caldwell; Jim Dyer; John Jacobson; Ronald Langstaff; Roy Caniano; Satwant Bajwa; Tae Kim; Tony D'Angelo

On June 19, 2001, RIII and NRR held a phone call to discuss questions raised by a RIII inspector during an inspection at Fermi. The inspector questioned whether the non-condensable gases that are introduced into the suppression pool have been considered for ECCS pump operation. The concern was that the gases would airbind the pumps or result in catastrophic failure of the pumps due to cavitation. The inspector also raised a question regarding the method used at Fermi for calculating net positive suction head (NPSH) for the ECCS pumps. The concern was that the licensee may not have adequately compensated for the fact that the suppression pool water temperature is significantly higher than the water used during the tests of the pumps performed to determine required NPSH. Participating from the Region were Jack Grobe, John Jacobson, and Gerry O'Dwyer. Participating from NRR were S. Singh Bajwa, Tae Kim, Claudia Craig, Dave Terao, Tony D'Angelo, Kerri Kavanagh, and Gary Hammer.

In response to the inspector's questions, NRR informed the region that this issue had been addressed previously. For Mark I containments the bottom of the downcomer is 2-4 feet higher than the suction strainers. For a LBLOCA, the swell would be less than 1 diameter of the downcomer (the exclusion zone) and the air bubble should not get to the strainer. For Mark II containments or plants with quenchers, the non-condensable gas discharge is greater and this was looked at for the design basis accident (loss of AC). Regarding the NPSH calculation question, NRR responded that the inclusion of the vapor pressure term for the higher temperature condition in the available NPSH calculation correctly compensates for temperature.

As a result of installing bigger suction strainers to solve the problem of blocked strainers due to debris, some plants may have changed the design of the suction under 10 CFR 50.59 and the licensee may have intruded on the exclusion zone, but this may still not be a problem and needs to be looked at on an individual basis. The NRR staff performed 4 audits as a result of Bulletin 96-03 and identified no instances where the larger strainers would present a problem.

NRR took several actions as a result of the call: 1) send a copy of the video of a large scale model that demonstrates this phenomena to RIII for viewing, 2) provide some thoughts on the non-loss of AC scenario, and 3) find out how fast the pumps would start.

Following the phone call further questions were provided to the NRR staff. The NRR staff stated that an AEOD report provided a good description of the phenomena in question. This report was issued on March 31, 1982 and is available in NUDOCS.

From: Gerard O'Dwyer, R3
To: Jim Dyer, R3
Date: 8/17/01 1:16PM
Subject: My DPV supplement.

CC: Bruce Berson; Darrell Schrum; JMJ3; John Grobe

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Supplement to Differing Professional View (DPV) of July 24, 2001

Please supplement my DPV with the information provided below. This supplemental information is intended to specifically identify the technical issues of concern to me per the DPV Management Directive.

Summary of Prevailing Staff View: The prevailing staff view of the potential for gas binding or degraded performance of BWR RHR pumps during a LOCA is provided in AEOD/E218 report dated March 31, 1982 and the NRR summary of the June 19, 2001 conference call. That report concludes that the air bubble phenomena is not a significant concern with respect to its potential for adversely affecting RHR pump performance. (Attachment 1) The NRR summary of the June 19, 2001 conference call (Attachment 2) states that the NRR staff believes that the AEOD report contains a good description of the phenomena. NRR also informed the Region during the conference call that for Mark I containments, if there is a LBLOCA, the penetration of the containment atmosphere into the torus would be less than 1 diameter of the downcomer (the exclusion zone) and the air bubble should not get to the strainer.

Description of My Views and How they Differ from the Prevailing Staff View: I respectfully disagree with the AEOD report and the NRR staff position as documented in the teleconference summary that the AEOD report provided a good description of the phenomena. For LOCA's without the loss of offsite power. I believe that the LCPI pumps could fail to perform their safety function because of gas ingestion, vapor locking, cavitation, including pressure transients associated with LOCA blowdown, or any combination of these. I am also concerned that complete or partial failures of other components in the LPCI system, such as check valves, injection valves, etc could occur due to similar causes. I also contend that the containment atmosphere will penetrate much deeper into the torus than 1 downcomer diameter and the air bubble will get to the strainer and be ingested.

The mention of the AEOD reprt at the end of the NRR summary implies that NRR believes that the report addresses of the many phenomena of my concerns. The AEOD report only addresses one of my concerns which is the ingestion of gas from dissolved gases and completely omits all of my other concerns particularly the ingestion of the gases coming from the downcomer. I also contend that there will be significant amounts of gases coming from the downcomer even after blowdown, and this has not been addressed. In this regard, the Class III, May 1979, report from GE, NEDE-24537-P, on the development of downcomer lateral loads from full scale test facility (FSTF) data, supports my concern. The report notes that the drywell volume per downcomer for the FSTF was much lower than in actual Mark I plants. I believe this is because the test in question was run in order to minimize the amount of cushioning provided by air so that the hydrodynamic loading would be maximized. Therefore, this test might not be realistic regarding how long the blowdown lasts, the depth of penetration, and thus the impact on gas ingestion. I believe the air bubble could last considerably longer than 3 -5 seconds.

Further, I believe that those gases will be pulled into the LPCI suctions by vortexing and other mechanisms and cause the LPCI pumps to gas bind. Ingestion by vortexing will be worst when the reactor pressure drops during a LOCA allowing higher flows through the pumps. Even worse is a Mark I containment with an ECCS suction header where, during a LOCA the velocity of the water in the header will be increased substantially by the HPCI pump if the pump takes suction from torus. Even if gas binding does not prevent the pumps from performing their safety function, there is the potential that gas ingestion would cause cavitation that would cause the pumps to fail or not perform their safety function. See NUREG/CR-2792, Section 4.2. (Enclosure 3)

I also disagree with the following related staff views:

1. The AEOD report calculations assume that the volume of gas that is of concern is limited to that contained in 196 cubic feet of water, the assumed volume of the RHR suction piping. I disagree with this assumption because it ignores the volume of water in the torus which also contains dissolved gases.

2. The NRR summary of the teleconference states that the swell (exclusion zone) will be limited to less than 1 diameter of the downcomer.

3. The summary stated that for Mark II containments the gas discharge was looked at for the design basis accident (loss of AC); however I am concerned about other than the DBA (no loss of offsite power) which I believe NRR indicated has not been anayzed.

Therefore, I believe that all operating reactors with Mark I and Mark II containments should be shut down until hardware repairs are made to prevent gas binding or cavitation of the LPCI pumps.

<u>Assessment of consequences should my position not be adopted:</u> Gas binding or cavitation of the LPCI pumps would allow containment failure and the release of excessive radioactivity to the public.

Attachment 1

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

AEOD/E218

MAR 3 1 1982

This is an internal, pre-decisional document not necessarily representing a position of AEOD or NRC.

MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

FROM: Stuart Rubin, Lead Engineer
Office for Analysis and Evaluation
of Operational Data

SUBJECT: ENGINEERING EVALUATION - POTENTIAL FOR AIR BINDING OR DEGRADED PERFORMANCE OF BWR RHR SYSTEM PUMPS DURING THE RECIRCULATION PHASE OF A LOCA

References:

(1) Energy Suppression and Fission Product Transport in Pressure-Suppression Pools, ORNL-TM-3448, April 1972

(2) Chemical Engineer's Handbook, R. H. Perry, 1973

(3) Suppression Pool Temperature Limits for BWR Containments, NUREG-0783, September 1981

The potential for degraded performance of boiling water reactor (BWR) residual heat removal (RHR) system pumps during the recirculation (or pool cooling) phase of a loss-of-coolant-accident (LOCA), due to air bubble generation in the torus pool during the blowdown phase, has been studied. The concerns involved degraded capability of the RHR system pumps due to air bubble entrainment and attendant pumping of a water-air mixture through the RHR torus-to-pump suction piping. Air binding of the pump due to bubble rise coalescence potentially could be an associated concern and has also been assessed. These concerns were prompted by the results of preliminary experiments performed in connection with a test and analysis program documented in Reference 1.

Reference 1 provides the results of a program, sponsored by the former Atomic Energy Commission in the early 1970's, which was designed to investigate both the pressure suppression (i.e., energy absorption) characteristics and the fission product transport characteristics of a BWR drywell-wetwell containment system during a LOCA. The experimental apparatus used for the testing portion of the program included a large water-filled tank (with one side partly made of glass for purposes of observation) and a variable high pressure steam supply. These

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equipment were intended to simulate, respectively, the torus pool and the reactor-system steam blowdown from the drywell to the torus pool during the postulated accident. The steam supply delivered steam at saturated conditions for pressures up to 125 psig. Steam flow rates could be varied between 0 and 6000 lbs/hr. In the tests, steam was discharged into the tank underwater via a ¼ inch Schedule 40 pipe. The water temperature in the tank varied between 50^{o} and 100^{o}F.

The results of some preliminary experiments associated with the program are described in the second section of the referenced report. During these early (system behavior) tests, the investigators found that when saturated steam was discharged into a tank containing ordinary water which had not been previously degassed, numerous small gas (air) bubbles were generated. These bubbles were described as being less than 1 mm in diameter. According to the authors, small bubbles were generated in such vast numbers that the quenching behavior of the underwater steam-jet, which was to be studied in the latter experiments, was obscured from accurate visual or photographic observation. In order to avoid this undesirable effect, the experimenters replaced the original pool water with water which was deaerated. This procedure removed the source of air for air bubble generation and thereby eliminated their problem.

The small bubbles seen by the investigators in these early tests can be explained as having been caused by the liberation of air into its free gaseous form from its water-dissolved state. This occurred when the condensing steam-jet caused a local pool temperature rise in the vicinity of the jet. The temperature increase reduced the volume percent of air which could be kept dissolved in the water to a level which was below its initial near saturated value. Thus the air was forced out of solution as free air bubbles, much like the "fizzing" of a carbonated bottle of soda when the cap is removed.

The phenomena observed in the preliminary tests is of interest to the post-LOCA, emergency core cooling of a BWR because: (1) the steam blowdown-quenching arrangements and quenching characteristics of these early experiments is very similar, respectively, to the actual BWR LOCA pressure suppression arrangement and the quenching characteristics expected of a typical BWR drywell-wetwell containment design; (2) BWR suppression pools also do not utilize deaerated water; and (3) the air which would be released from solution represents a source of gas which might be drawn into the RHR system suction piping during the low pressure recirculation phase of emergency core cooling. During this latter phase, when the RHR pumps draw suction from the torus pool, degraded pump performance caused by air bubble entrainment in the suction stream could be a concern.

In order to assess the significance of this phenomena with respect to RHR pump performance, a representative BWR Mark I containment was considered. For the Mark I containment, such as at Browns Ferry 1, the torus pool normally contains about 126,000 ft^3 of water with a maximum temperature of 95°F as required by the plant technical specifications. The pool blanket gas inside containment is ordinary air since BWR containments do not have to be inerted. Air is normally about 80% nitrogen and about 20% oxygen by volume (not including other gases). For purposes of analysis, the containment air was assumed to be the one and only source of gases dissolved in the pool water.

A conservative calculation was performed using Henry's solubility law from Reference 2 to assess the extent to which the air dissolved in the torus pool would be liberated as free gas bubbles during a LOCA blowdown. This was then compared to the readily available information on the performance of RHR system pumps pumping water-air mixtures.

Attachment 1 provides a calculation of the air bubble content (on a volume percent basis of the air in the pool) which would be developed as a result of a postulated LOCA blowdown. The calculation assumed conservatively low initial and high final pool temperatures, an initially saturated solution of air in water, and no credit for the increase in containment pressure which would accompany a LOCA. The large air bubbles generated by the initial blowdown of drywell air into the torus pool was neglected in this assessment since this air source would be expected to rise quickly out of the pool water and become part of the pressurized air-vapor space trapped above the pool surface. As shown in the appendix, the gases liberated from solution would occupy less than 2% by volume of the resulting water-gas mixture in the pool. If the air bubbles are assumed to be: (1) generated and remain uniformly distributed in the pool; and (2) small enough so that they do not rise significantly during the entire injection phase of emergency core cooling, then the performance of the RHR pumps (which draw suction from locations near the bottom of the pool) may be conservatively assessed for a 2% by volume air bubble content.

In order to evaluate the significance of 2% gas content on RHR system pump performance, cognizant General Electric Company (GE) engineering personnel at San Jose, California, were contacted on March 11, 1982. Mr. Pio W. Ianni of GE stated that communications between GE and the Byron-Jackson Company (a manufacturer of RHR pumps) several years ago had led to the understanding that these pumps could tolerate "several percent" air content in the suction stream without a discernible loss in pump performance. Given the validity of this information, it may be concluded that unacceptable RHR pump performance during the recirculation phase of a postulated loss-of-coolant accident, due to air bubble formation in the pool during the blowdown phase, should not be expected.

The above conclusion that bubble formation in the pool during a LOCA should not be a concern tends to be further supported to a limited degree by actual plant operating experience. There have been a significant number of BWR events involving safety-relief valve discharges to the pool. There have also been a great many operations of the high pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system at BWRs over the years. These systems actuations, which involve the discharge of high energy steam to the pool, have occurred during both equipment testing and plant transients. In many cases, pool cooling via the RHR system has been initiated during or shortly after such steam discharges to the pool. Although for most of these events the maximum bulk average pool temperature did not rise to temperature levels closely approaching the maximum predicted pool temperatures during a LOCA, the temperature around the tailpipe quenchers or HPCI/RCIC exhausts likely rose sufficiently to generate considerable gas bubbles. The most severe domestic event to date (see Reference 3) involved a BWR which experienced a stuck open SRV from a reactor pressure of 980 psig. Bulk pool temperature rose from below 100°F to approximately 165°F. Torus pool cooling and pool sprays, using one RHR pump, successfully reduced torus pool temperature back to normal levels in a few hours. To date, neither this plant nor any other plant has reported unacceptable RHR pool cooling attendant to such steam blowdown events. At the same time during these events, small, unnoticed, but relatively insignificant degradations in pump performance cannot be ruled out.

Pump air binding caused by bubble rise and collection in the impeller region of the pump casing might also be a concern. However, the RHR suction piping generally slopes downward where it connects to the underside of the pool wall. This should preclude large amounts of air bubbles in the pool from rising into the pump impeller region. For analysis purposes, however, it was assumed that all of the bubbles which were drawn into and contained in the RHR suction piping were suddenly allowed to rise freely into the pump (as might be postulated if the RHR pumps were stopped within a short time of initiating recirculation or pool) cooling. As shown in Attachment 2, the effect of such a bubble rise would not be expected to cause air binding of the RHR pumps.

As a final note, the GE engineer contacted on March 11 stated that GE investigat the same concern several years ago and had concluded that a problem with RHR pum performance did not exist under such conditions. However, to his knowledge, no documentation of this work was ever submitted to the Commission for review. The work is on file at San Jose, however. Mr. Ianni indicated that a report could be provided upon formal request from the Division of Licensing, ONRR.

In view of the assessment provided herein, it is my conclusion that, at this time, the air bubble generation phenomena described in Reference 1 is not a significant concern with respect to its potential for adversely affecting RHR pump performance.

Stuart Rubin

Stuart Rubin, Lead Engineer
Office for Analysis and Evaluation
of Operational Data

Attachments:
As Stated

cc w/attachments:
J. Heltemes, AEOD
T. Wolf, AEOD
M. El-Zeftawy, AEOD

Attachment 1

Gas Release in Suppression Pool

Assume Water/Air at:

T_i = 68°F = 20°C   80% Nitrogen in Air

T_f = 195°F = 90°C   20% Oxygen in Air

Henry's Constants for Nitrogen:

@ 20°C H = 8.04×10^4 $\frac{\text{atm}}{\text{mole fraction}}$

@ 90°C H = 12.6×10^4

Partial Pressures at 20°C

Vapor pressure: 0.3391 psi $= \frac{0.3391}{14.7} = 0.023$ atm

N_2 pressure: $(1-0.023) \times 0.8 = 0.782$ atm.

O_2 pressure: $(1-0.023) \times 0.2 = 0.195$ atm.

Nitrogen: $X_{N_2} = \frac{0.782}{8.04 \times 10^4} = 9.726 \times 10^{-6}$ mole fraction.

$$\frac{9.726 \times 10^{-6}}{1-9.726 \times 10^{-6}} \times \frac{28}{18} \times 100 = 1.513 \times 10^{-3}\ gN_2/100g \text{ water.}$$

Oxygen: $X_{O_2} = \frac{0.195}{4.01 \times 10^4} = 4.863 \times 10^{-6}$ mole fraction.

$$\frac{4.863 \times 10^{-6}}{1-4.863 \times 10^{-6}} \times \frac{32}{18} \times 100 = 8.645 \times 10^{-4}\ gO_2/100g \text{ water.}$$

Partial Pressures at 90°C

Vapor pressure: 10.172 psi $= \frac{10.172}{14.7} = 0.692$ atm.

N_2 pressure: $(1-0.692) \times 0.8 = 0.246$ atm.

O_2 pressure: $(1-0.692) \times 0.2 = 0.062$ atm.

Attachment 2

Gas Volume for Potential Air Binding of the RHR Pumps

Assume that the RHR suction piping is:

L = 40ft. long
D = 30in. diameter

Then:

$$Vol = \frac{\pi D^2}{4} \times L$$

$$= \frac{\pi (30)^2}{4} \times (40 \times 12)$$

$$Vol = 339{,}291 \text{ in}^2$$

or

$$Vol = 196 \text{ ft}^3$$

from Attachment #1

Vol of gas = 196 x .02 = 3.92 ft^3

Taking credit for pressure rise in torus during blowdown:

$$P_f V_f = P_i V_i$$

$$V_f = \frac{P_i}{P_f} V_i$$

where V_i = 3.92 ft^3

P_i = 15 psi

P_f = 35 psi (i.e., ΔP_{Torus} = 20 psi)

$$V_f = \frac{15}{35} (3.92)$$

Thus V_f = $\underline{\underline{1.7 \text{ ft}^3}}$ (in pump impeller region)

Nitrogen: $X_{N_2} = \frac{0.246}{12.6 \times 10^{-4}} = 1.952 \times 10^{-6}$ mole fraction.

$$\frac{1.952 \times 10^{-6}}{1\text{-}1952 \times 10^{-6}} \quad \frac{28}{18} \times 100 = 3.036 \times 10^{-4} \quad gN_{2}/100g \text{ water.}$$

Oxygen: $X_{O_2} = \frac{0.062}{6.99 \times 10^{-4}} = 8.870 \times 10^{-7}$ mole fraction.

$$\frac{8.870 \times 10^{-7}}{1\text{-}8.870 \times 10^{-7}} = \frac{32}{18} \times 100 = 1.577 \times 10^{-4} \quad gN_{2}/100g \text{ water.}$$

Gas Release when the water is heated from 20^oC to 90^oC by weight.

Nitrogen: $1.513 \times 10^{-3} - 3.036 \times 10^{-4} = 1.2094 \times 10^{-3}$ gN_2/100g water.

Oxygen: $8.645 \times 10^{-4} - 1.577 \times 10^{-4} = 7.068 \times 1\text{-}^{-4}$ gN_2/100g water.

Or by volume at STP:

Nitrogen: $\frac{1.2094 \times 10^{-3}}{28} \times 2.24 \times 10^{4} = 0.968$ cc STP/100g water.

Oxygen: $\frac{7.068 \times 10^{-4}}{32} \times 2.24 \times 10^{4} = 0.495$ cc STP/100g water.

Adding: Total gas: = 1.363 cc STP/100g water. (Amount of gas released when the water is heated from 20^oC (68^oC) to 90^oC (194^oF)

Attachment 2

On June 19, 2001, RIII and NRR held a phone call to discuss questions raised by a RIII inspector during an inspection at Fermi. The inspector questioned whether the non-condensable gases that are introduced into the suppression pool have been considered for ECCS pump operation. The concern was that the gases would airbind the pumps or result in catastrophic failure of the pumps due to cavitation. The inspector also raised a question regarding the method used at Fermi for calculating net positive suction head (NPSH) for the ECCS pumps. The concern was that the licensee may not have adequately compensated for the fact that the suppression pool water temperature is significantly higher than the water used during the tests of the pumps performed to determine required NPSH. Participating from the Region were Jack Grobe, John Jacobson, and Gerry O'Dwyer. Participating from NRR were S. Singh Bajwa, Tae Kim, Claudia Craig, Dave Terao, Tony D'Angelo, Kerri Kavanagh, and Gary Hammer.

In response to the inspector's questions, NRR informed the region that this issue had been addressed previously. For Mark I containments the bottom of the downcomer is 2-4 feet higher than the suction strainers. For a LBLOCA, the swell would be less than 1 diameter of the downcomer (the exclusion zone) and the air bubble should not get to the strainer. For Mark II containments or plants with quenchers, the non-condensable gas discharge is greater and this was looked at for the design basis accident (loss of AC). Regarding the NPSH calculation question, NRR responded that the inclusion of the vapor pressure term for the higher temperature condition in the available NPSH calculation correctly compensates for temperature.

As a result of installing bigger suction strainers to solve the problem of blocked strainers due to debris, some plants may have changed the design of the suction under 10 CFR 50.59 and the licensee may have intruded on the exclusion zone, but this may still not be a problem and needs to be looked at on an individual basis. The NRR staff performed 4 audits as a result of Bulletin 96-03 and identified no instances where the larger strainers would present a problem.

NRR took several actions as a result of the call: 1) send a copy of the video of a large scale model that demonstrates this phenomena to RIII for viewing, 2) provide some thoughts on the non-loss of AC scenario, and 3) find out how fast the pumps would start.

Following the phone call further questions were provided to the NRR staff. The NRR staff stated that an AEOD report provided a good description of the phenomena in question. This report was issued on March 31, 1982 and is available in NUDOCS.

Attachment 3

NUREG/CR-27
CREARE TM-8

An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditio

Prepared by P. S. Kamath, T. J. Tantillo, W. L. Swift
1982
Creare, Inc.

Prepared for
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Commission

PDR Acc # 8210050360

OCT 15 1982

indicate that degradation becomes significant. (See Figure 3-7, where Florjancic's results show a 10% head reduction at 120% flow rate).

In addition to the considerations of flow limitations on the 2% allowed air ingestion rate, even small quantities of air affect the NPSH requirements for pumps. The results shown in Figure 4-1 apply to pumps operating with sufficient NPSH to avoid cavitation. The following section deals with the effects of air ingestion on NPSH and the combined effect of low NPSH and air ingestion on head degradation.

4.2 Cavitation and Air Ingestion

There are very few sources of data on the combined effects of cavitation and air ingestion on pump performance. Figure 4-2 shows results from [31] on a pump of specific speed 1074 operating near best efficiency point. The curves have been replotted for Figure 3-12 and head values have been normalized by the non-cavitating liquid head. The curves show cavitation 'breaks' at various levels of air ingestion. For each curve, the flow rate and speed are fixed and inlet pressure (NPSH) is varied. As NPSH decreases, the measured differential head decreases gradually and then abruptly, due to cavitation. The values of head are normalized by the non-cavitating value in liquid with no air.

Applying a commonly (albeit arbitrarily) used criterion of defining the NPSH required as the NPSH value at which head degrades by 3% from the non-cavitating value, one can construct a locus of the required NPSH as a function of the air ingestion level. Figure 4-3 shows four such points obtained by plotting the NPSH values for which head has degraded by 3% from the non-cavitating values. The plotted points are taken from the four curves shown in Figure 4-2 for air fractions of 0%, 3.3%, 6.6% and 9.9%, respectively. In order to establish a guideline for calculating the increased NPSHR in the presence of air, an arbitrary relationship is presented. This relationship is:

$$NPSHR_{air/water} = NPSHR_{water}\ (1+0.5\ AF)$$

where AF is the air volume fraction in percent.

The relationship is shown in Figure 4-3 as a straight line. It is evident from the figure that the equation for NPSH requirements in the presence of air provides a margin above the values obtained by Merry [31]. For example at 2% air volume fraction the NPSH requirement is equivalent to that obtained with 3.3% air volume fraction. The conservatism used in establishing the straight line is arbitrary. However, it is felt necessary because of the limited amount of data available upon which to base such a guideline. It should be noted that the guideline is only intended for use for air volume fractions less than 2%.

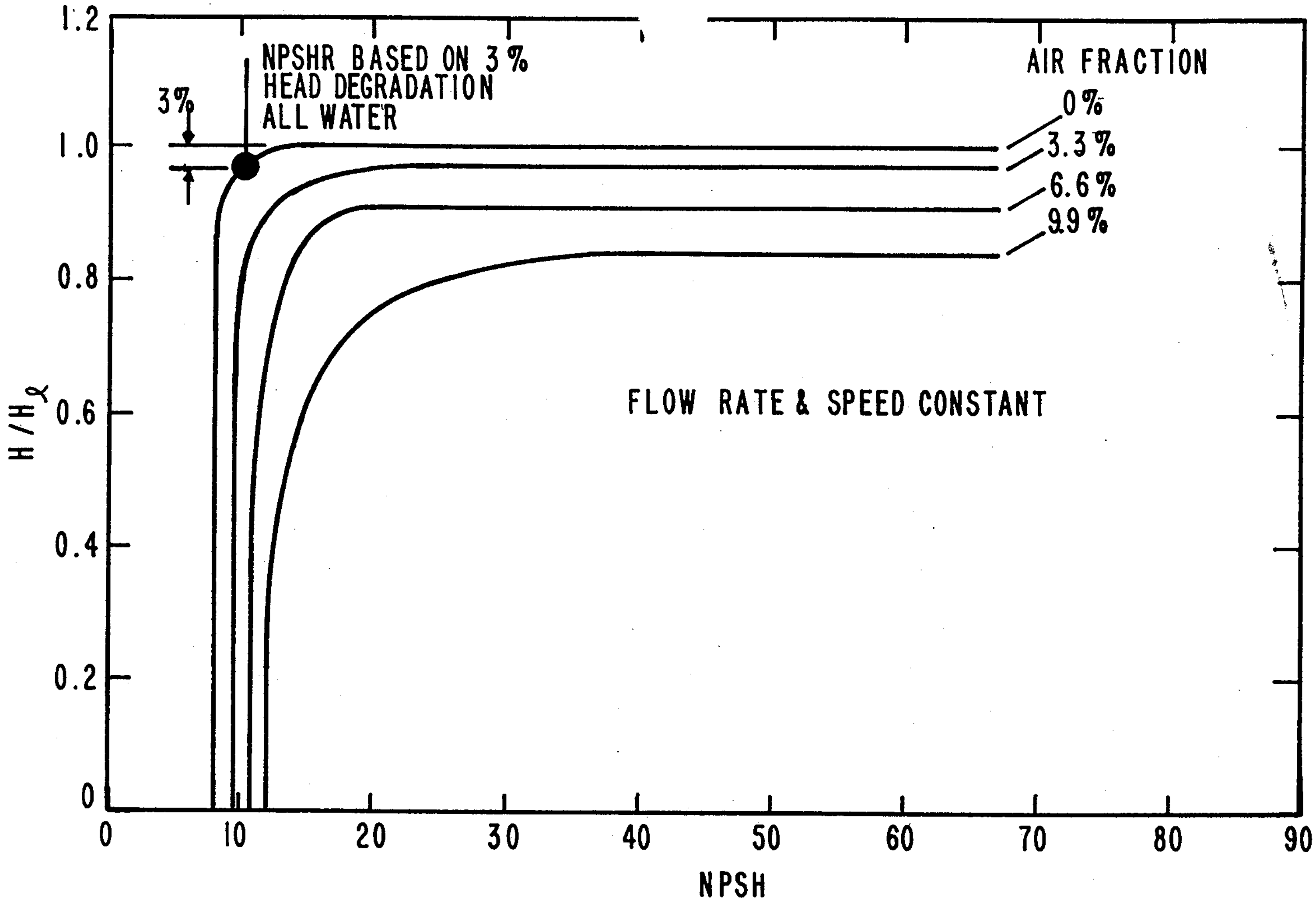

Figure 4-2. NORMALIZED HEAD VS. NPSH AT DIFFERENT VOID FRACTIONS FROM MERRY [31]

-hot water
NPSHR

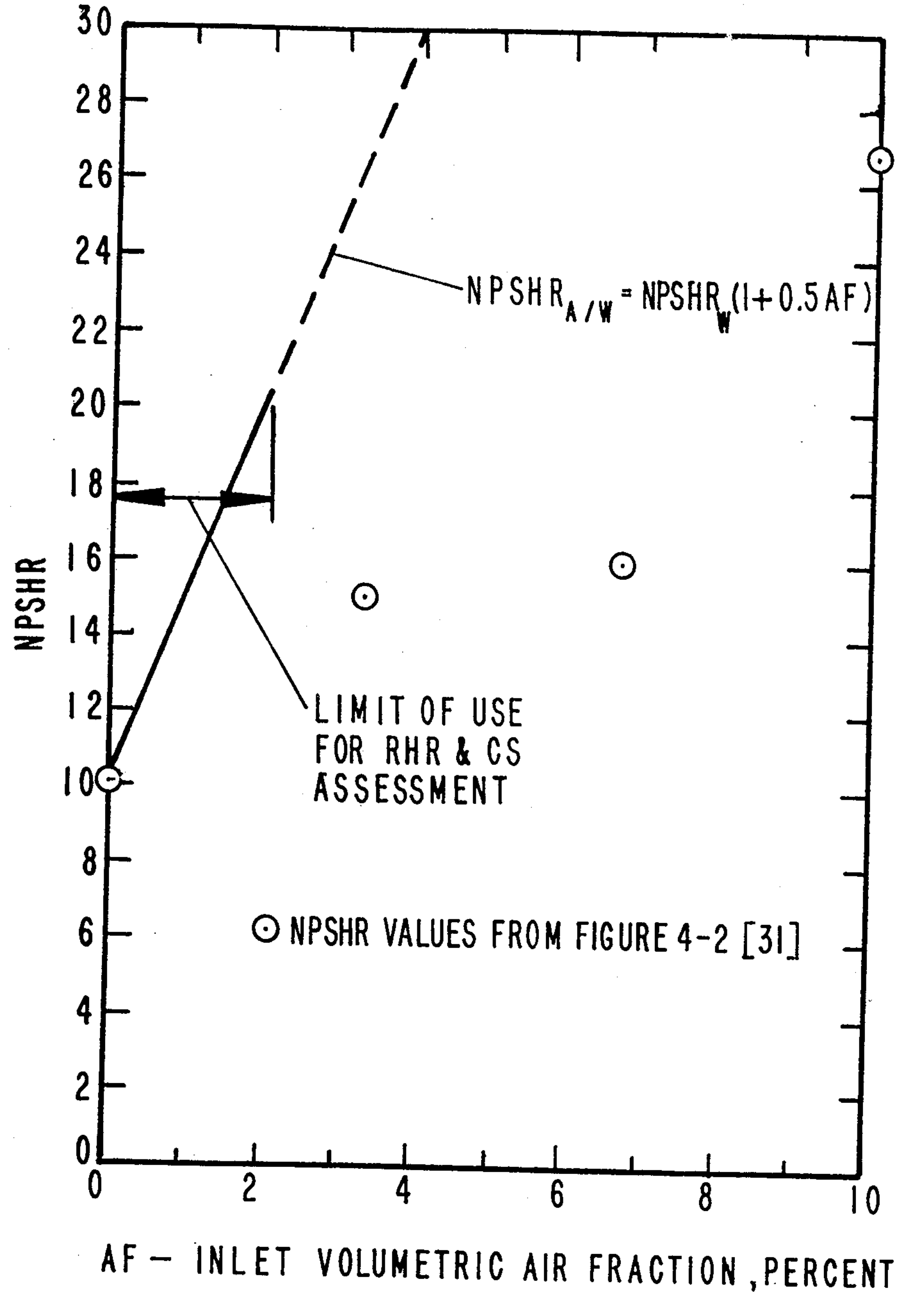

Figure 4-3. NPSH REQUIRED VS. AIR FRACTION BASED ON 3% HEAD DEGRADATION, MERRY [31]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

August 23, 2001

MEMORANDUM TO: Steven A. Reynolds, Deputy Director
Division of Reactor Projects

FROM: J. E. Dyer
Regional Administrator

James L. Caldwell

SUBJECT: AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW REGARDING BWR ECCS SUCTION CONCERNS

This memorandum is to confirm our conversation regarding the Differing Professional View (DPV) concerning BWR ECCS suction concerns (copy attached). In accordance with Management Directive 10.159, Differing Professional Views or Opinions, you have been appointed as the chairperson for the ad hoc review panel. Additionally, Kenneth Riemer, Division of Reactor Projects, Region III, has been appointed as a technically qualified member of the panel.

This memorandum also confirms that Darrell Schrum from the Division of Reactor Safety, Region III, has been selected as your other panel member. He was the only individual requested by the employee submitting the DPV.

You are to conduct the review of this DPV in accordance with Management Directive 10.159. You should complete your review and forward your recommendation to me by October 5, 2001.

Attachments: As stated

cc w/atts.: D. Schrum, DRS
K. Riemer, DRP

cc w/o atts.: J. McDermott, OD/HR
J. Grobe, RIII

A/3

From: Gerard O'Dwyer , R3
To: Jim Dyer , R3
Date: 8/17/01 1:16PM
Subject: My DPV supplement.

CC: Bruce Berson; Darrell Schrum; JMJ3; John Grobe

Supplement to Differing Professional View (DPV) of July 24, 2001

Please supplement my DPV with the information provided below. This supplemental information is intended to specifically identify the technical issues of concern to me per the DPV Management Directive.

Summary of Prevailing Staff View: The prevailing staff view of the potential for gas binding or degraded performance of BWR RHR pumps during a LOCA is provided in AEOD/E218 report dated March 31, 1982 and the NRR summary of the June 19, 2001 conference call. That report concludes that the air bubble phenomena is not a significant concern with respect to its potential for adversely affecting RHR pump performance. (Attachment 1) The NRR summary of the June 19, 2001 conference call (Attachment 2) states that the NRR staff believes that the AEOD report contains a good description of the phenomena. NRR also informed the Region during the conference call that for Mark I containments, if there is a LBLOCA, the penetration of the containment atmosphere into the torus would be less than 1 diameter of the downcomer (the exclusion zone) and the air bubble should not get to the strainer.

Description of My Views and How they Differ from the Prevailing Staff View: I respectfully disagree with the AEOD report and the NRR staff position as documented in the teleconference summary that the AEOD report provided a good description of the phenomena. For LOCA's without the loss of offsite power. I believe that the LCPI pumps could fail to perform their safety function because of gas ingestion, vapor locking, cavitation, including pressure transients associated with LOCA blowdown, or any combination of these. I am also concerned that complete or partial failures of other components in the LPCI system, such as check valves, injection valves, etc could occur due to similar causes. I also contend that the containment atmosphere will penetrate much deeper into the torus than 1 downcomer diameter and the air bubble will get to the strainer and be ingested.

The mention of the AEOD reprt at the end of the NRR summary implies that NRR believes that the report addresses of the many phenomena of my concerns. The AEOD report only addresses one of my concerns which is the ingestion of gas from dissolved gases and completely omits all of my other concerns particularly the ingestion of the gases coming from the downcomer. I also contend that there will be significant amounts of gases coming from the downcomer even after blowdown, and this has not been addressed. In this regard, the Class III, May 1979, report from GE, NEDE-24537-P, on the development of downcomer lateral loads from full scale test facility (FSTF) data, supports my concern. The report notes that the drywell volume per downcomer for the FSTF was much lower than in actual Mark I plants. I believe this is because the test in question was run in order to minimize the amount of cushioning provided by air so that the hydrodynamic loading would be maximized. Therefore, this test might not be realistic regarding how long the blowdown lasts, the depth of penetration, and thus the impact on gas ingestion. I believe the air bubble could last considerably longer than 3 -5 seconds.

Further, I believe that those gases will be pulled into the LPCI suctions by vortexing and other mechanisms and cause the LPCI pumps to gas bind. Ingestion by vortexing will be worst when the reactor pressure drops during a LOCA allowing higher flows through the pumps. Even worse is a Mark I containment with an ECCS suction header where, during a LOCA the velocity of the water in the header will be increased substantially by the HPCI pump if the pump takes suction from torus. Even if gas binding does not prevent the pumps from performing their safety function, there is the potential that gas ingestion would cause cavitation that would cause the pumps to fail or not perform their safety function. See NUREG/CR-2792, Section 4.2. (Enclosure 3)

I also disagree with the following related staff views:

1. The AEOD report calculations assume that the volume of gas that is of concern is limited to that contained in 196 cubic feet of water, the assumed volume of the RHR suction piping. I disagree with this assumption because it ignores the volume of water in the torus which also contains dissolved gases.

2. The NRR summary of the teleconference states that the swell (exclusion zone) will be limited to less than 1 diameter of the downcomer.

3. The summary stated that for Mark II containments the gas discharge was looked at for the design basis accident (loss of AC); however I am concerned about other than the DBA (no loss of offsite power) which I believe NRR indicated has not been anayzed.

Therefore, I believe that all operating reactors with Mark I and Mark II containments should be shut down until hardware repairs are made to prevent gas binding or cavitation of the LPCI pumps.

<u>Assessment of consequences should my position not be adopted:</u> Gas binding or cavitation of the LPCI pumps would allow containment failure and the release of excessive radioactivity to the public.

WASHINGTON, D. C. 20555

MAR 3 1 1982

This is an internal, pre-decisional document not necessarily representing a position of AEOD or NRC.

MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

FROM: Stuart Rubin, Lead Engineer
Office for Analysis and Evaluation
of Operational Data

SUBJECT: ENGINEERING EVALUATION - POTENTIAL FOR AIR BINDING OR DEGRADED PERFORMANCE OF BWR RHR SYSTEM PUMPS DURING THE RECIRCULATION PHASE OF A LOCA

References:

(1) Energy Suppression and Fission Product Transport in Pressure-Suppression Pools, ORNL-TM-3448, April 1972

(2) Chemical Engineer's Handbook, R. H. Perry, 1973

(3) Suppression Pool Temperature Limits for BWR Containments, NUREG-0783, September 1981

The potential for degraded performance of boiling water reactor (BWR) residual heat removal (RHR) system pumps during the recirculation (or pool cooling) phase of a loss-of-coolant-accident (LOCA), due to air bubble generation in the torus pool during the blowdown phase, has been studied. The concerns involved degraded capability of the RHR system pumps due to air bubble entrainment and attendant pumping of a water-air mixture through the RHR torus-to-pump suction piping. Air binding of the pump due to bubble rise coalescence potentially could be an associated concern and has also been assessed. These concerns were prompted by the results of preliminary experiments performed in connection with a test and analysis program documented in Reference 1.

Reference 1 provides the results of a program, sponsored by the former Atomic Energy Commission in the early 1970's, which was designed to investigate both the pressure suppression (i.e., energy absorption) characteristics and the fission product transport characteristics of a BWR drywell-wetwell containment system during a LOCA. The experimental apparatus used for the testing portion of the program included a large water-filled tank (with one side partly made of glass for purposes of observation) and a variable high pressure steam supply. These

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equipment were intended to simulate, respectively, the torus pool and the reactor system steam blowdown from the drywell to the torus pool during the postulated accident. The steam supply delivered steam at saturated conditions for pressures up to 125 psig. Steam flow rates could be varied between 0 and 6000 lbs/hr. In the tests, steam was discharged into the tank underwater via a ¼ inch Schedule 40 pipe. The water temperature in the tank varied between 50° and 100°F.

The results of some preliminary experiments associated with the program are described in the second section of the referenced report. During these early (system behavior) tests, the investigators found that when saturated steam was discharged into a tank containing ordinary water which had not been previously degassed, numerous small gas (air) bubbles were generated. These bubbles were described as being less than 1 mm in diameter. According to the authors, small bubbles were generated in such vast numbers that the quenching behavior of the underwater steam-jet, which was to be studied in the latter experiments, was obscured from accurate visual or photographic observation. In order to avoid this undesirable effect, the experimenters replaced the original pool water with water which was deaerated. This procedure removed the source of air for air bubble generation and thereby eliminated their problem.

The small bubbles seen by the investigators in these early tests can be explained as having been caused by the liberation of air into its free gaseous form from its water-dissolved state. This occurred when the condensing steam-jet caused a local pool temperature rise in the vicinity of the jet. The temperature increas reduced the volume percent of air which could be kept dissolved in the water to a level which was below its initial near saturated value. Thus the air was forced out of solution as free air bubbles, much like the "fizzing" of a carbonated bottle of soda when the cap is removed.

The phenomena observed in the preliminary tests is of interest to the post-LOCA, emergency core cooling of a BWR because: (1) the steam blowdown-quenching arrangements and quenching characteristics of these early experiments is very similar, respectively, to the actual BWR LOCA pressure suppression arrangement and the quenching characteristics expected of a typical BWR drywell-wetwell containment design; (2) BWR suppression pools also do not utilize deaerated water; and (3) the air which would be released from solution represents a source of gas which might be drawn into the RHR system suction piping during the low pressure recirculation phase of emergency core cooling. During this latter phase, when the RHR pumps draw suction from the torus pool, degraded pump performance caused by air bubble entrainment in the suction stream could be a concern.

In order to assess the significance of this phenomena with respect to RHR pump performance, a representative BWR Mark I containment was considered. For the Mark I containment, such as at Browns Ferry 1, the torus pool normally contains about 126,000 ft^3 of water with a maximum temperature of 95°F as required by the plant technical specifications. The pool blanket gas inside containment is ordinary air since BWR containments do not have to be inerted. Air is normally about 80% nitrogen and about 20% oxygen by volume (not including other gases). For purposes of analysis, the containment air was assumed to be the one and only source of gases dissolved in the pool water.

A conservative calculation was performed using Henry's solubility law from Reference 2 to assess the extent to which the air dissolved in the torus pool would be liberated as free gas bubbles during a LOCA blowdown. This was then compared to the readily available information on the performance of RHR system pumps pumping water-air mixtures.

Attachment 1 provides a calculation of the air bubble content (on a volume percent basis of the air in the pool) which would be developed as a result of a postulated LOCA blowdown. The calculation assumed conservatively low initial and high final pool temperatures, an initially saturated solution of air in water, and no credit for the increase in containment pressure which would accompany a LOCA. The large air bubbles generated by the initial blowdown of drywell air into the torus pool was neglected in this assessment since this air source would be expected to rise quickly out of the pool water and become part of the pressurized air-vapor space trapped above the pool surface. As shown in the appendix, the gases liberated from solution would occupy less than 2% by volume of the resulting water-gas mixture in the pool. If the air bubbles are assumed to be: (1) generated and remain uniformly distributed in the pool; and (2) small enough so that they do not rise significantly during the entire injection phase of emergency core cooling, then the performance of the RHR pumps (which draw suction from locations near the bottom of the pool) may be conservatively assessed for a 2% by volume air bubble content.

In order to evaluate the significance of 2% gas content on RHR system pump performance, cognizant General Electric Company (GE) engineering personnel at San Jose, California, were contacted on March 11, 1982. Mr. Pio W. Ianni of GE stated that communications between GE and the Byron-Jackson Company (a manufacturer of RHR pumps) several years ago had led to the understanding that these pumps could tolerate "several percent" air content in the suction stream without a discernible loss in pump performance. Given the validity of this information, it may be concluded that unacceptable RHR pump performance during the recirculation phase of a postulated loss-of-coolant accident, due to air bubble formation in the pool during the blowdown phase, should not be expected.

The above conclusion that bubble formation in the pool during a LOCA should not be a concern tends to be further supported to a limited degree by actual plant operating experience. There have been a significant number of BWR events involving safety-relief valve discharges to the pool. There have also been a great many operations of the high pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system at BWRs over the years. These systems actuations, which involve the discharge of high energy steam to the pool, have occurred during both equipment testing and plant transients. In many cases, pool cooling via the RHR system has been initiated during or shortly after such steam discharges to the pool. Although for most of these events the maximum bulk average pool temperature did not rise to temperature levels closely approaching the maximum predicted pool temperatures during a LOCA, the temperature around the tailpipe quenchers or HPCI/RCIC exhausts likely rose sufficiently to generate considerable gas bubbles. The most severe domestic event to date (see Reference 3) involved a BWR which experienced a stuck open SRV from a reactor pressure of 980 psig. Bulk pool temperature rose from below 100°F to approximately 165°F. Torus pool cooling and pool sprays, using one RHR pump, successfully reduced torus pool temperature back to normal levels in a few hours. To date, neither this plant nor any other plant has reported unacceptable RHR pool cooling attendant to such steam blowdown events. At the same time during these events, small, unnoticed, but relatively insignificant degradations in pump performance cannot be ruled out.

Pump air binding caused by bubble rise and collection in the impeller region of the pump casing might also be a concern. However, the RHR suction piping generally slopes downward where it connects to the underside of the pool wall. This should preclude large amounts of air bubbles in the pool from rising into the pump impeller region. For analysis purposes, however, it was assumed that all of the bubbles which were drawn into and contained in the RHR suction piping were suddenly allowed to rise freely into the pump (as might be postulated if the RHR pumps were stopped within a short time of initiating recirculation or pool) cooling. As shown in Attachment 2, the effect of such a bubble rise would not be expected to cause air binding of the RHR pumps.

As a final note, the GE engineer contacted on March 11 stated that GE investigat the same concern several years ago and had concluded that a problem with RHR pum performance did not exist under such conditions. However, to his knowledge, no documentation of this work was ever submitted to the Commission for review. The work is on file at San Jose, however. Mr. Ianni indicated that a report could be provided upon formal request from the Division of Licensing, ONRR.

In view of the assessment provided herein, it is my conclusion that, at this time, the air bubble generation phenomena described in Reference 1 is not a significant concern with respect to its potential for adversely affecting RHR pump performance.

Stuart Rubin

Stuart Rubin, Lead Engineer
Office for Analysis and Evaluation
of Operational Data

Attachments:
As Stated

cc w/attachments:
J. Heltemes, AEOD
T. Wolf, AEOD
M. El-Zeftawy, AEOD

Attachment 1

Gas Release in Suppression Pool

Assume Water/Air at:

$T_i = 68^oF = 20^oC$ 80% Nitrogen in Air

$T_f = 195^oF = 90^oC$ 20% Oxygen in Air

Henry's Constants for Nitrogen:

@ 20^oC $H = 8.04 \times 10^4$ $\frac{\text{atm}}{\text{mole fraction}}$

@ 90^oC $H = 12.6 \times 10^4$

Partial Pressures at 20^oC

Vapor pressure: 0.3391 psi $= \frac{0.3391}{14.7} = 0.023$ atm

N_2 pressure: $(1-0.023) \times 0.8 = 0.782$ atm.

O_2 pressure: $(1-0.023) \times 0.2 = 0.195$ atm.

Nitrogen: $X_{N_2} = \frac{0.782}{8.04 \times 10^4} = 9.726 \times 10^{-6}$ mole fraction.

$$\frac{9.726 \times 10^{-6}}{1-9.726 \times 10^{-6}} \times \frac{28}{18} \times 100 = 1.513 \times 10^{-3}\ gN_2/100g \text{ water.}$$

Oxygen: $X_{O_2} = \frac{0.195}{4.01 \times 10^4} = 4.863 \times 10^{-6}$ mole fraction.

$$\frac{4.863 \times 10^{-6}}{1-4.863 \times 10^{-6}} \times \frac{32}{18} \times 100 = 8.645 \times 10^{-4}\ gO_2/100g \text{ water.}$$

Partial Pressures at 90^oC

Vapor pressure: 10.172 psi $= \frac{10.172}{14.7} = 0.692$ atm.

N_2 pressure: $(1-0.692) \times 0.8 = 0.246$ atm.

O_2 pressure: $(1-0.692) \times 0.2 = 0.062$ atm.

Attachment 2

Gas Volume for Potential Air Binding of the RHR Pumps

Assume that the RHR suction piping is:

$$L = 40\text{ft. long}$$
$$D = 30\text{in. diameter}$$

Then:

$$Vol = \frac{\pi D^2}{4} \times L$$

$$= \frac{\pi (30)^2}{4} \times (40 \times 12)$$

$$Vol = 339{,}291 \text{ in}^2$$

or

$$Vol = 196 \text{ ft}^3$$

from Attachment #1

$$\text{Vol of gas} = 196 \times .02 = 3.92 \text{ ft}^3$$

Taking credit for pressure rise in torus during blowdown:

$$P_f V_f = P_i V_i$$

$$V_f = \frac{P_i}{P_f} V_i$$

where $V_i = 3.92 \text{ ft}^3$

$P_i = 15$ psi

$P_f = 35$ psi (i.e., $\Delta P_{Torus} = 20$ psi)

$$V_f = \frac{15}{35} (3.92)$$

Thus $V_f = \underline{\underline{1.7 \text{ ft}^3}}$ (in pump impeller region)

Nitrogen: $X_{N_2} = \frac{0.246}{12.6 \times 10^{-4}} = 1.952 \times 10^{-6}$ mole fraction.

$$\frac{1.952 \times 10^{-6}}{1-1952 \times 10^{-6}} \quad \frac{28}{18} \times 100 = 3.036 \times 10^{-4} \text{ gN}_2\text{/100g water.}$$

Oxygen: $X_{O_2} = \frac{0.062}{6.99 \times 10^{-4}} = 8.870 \times 10^{-7}$ mole fraction.

$$\frac{8.870 \times 10^{-7}}{1-8.870 \times 10^{-7}} = \frac{32}{18} \times 100 = 1.577 \times 10^{-4} \text{ gN}_2\text{/100g water.}$$

Gas Release when the water is heated from 20°C to 90°C by weight.

Nitrogen: $1.513 \times 10^{-3} - 3.036 \times 10^{-4} = 1.2094 \times 10^{-3}$ gN_2/100g water.

Oxygen: $8.645 \times 10^{-4} - 1.577 \times 10^{-4} = 7.068 \times 1\text{-}^{-4}$ gN_2/100g water.

Or by volume at STP:

Nitrogen: $\frac{1.2094 \times 10^{-3}}{28} \times 2.24 \times 10^{4} = 0.968$ cc STP/100g water.

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In response to the inspector's questions, NRR informed the region that this issue had been addressed previously. For Mark I containments the bottom of the downcomer is 2-4 feet higher than the suction strainers. For a LBLOCA, the swell would be less than 1 diameter of the downcomer (the exclusion zone) and the air bubble should not get to the strainer. For Mark II containments or plants with quenchers, the non-condensable gas discharge is greater and this was looked at for the design basis accident (loss of AC). Regarding the NPSH calculation question, NRR responded that the inclusion of the vapor pressure term for the higher temperature condition in the available NPSH calculation correctly compensates for temperature.

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In addition to the considerations of flow limitations on the 2% allowed air ingestion rate, even small quantities of air affect the NPSH requirements for pumps. The results shown in Figure 4-1 apply to pumps operating with sufficient NPSH to avoid cavitation. The following section deals with the effects of air ingestion on NPSH and the combined effect of low NPSH and air ingestion on head degradation.

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Applying a commonly (albeit arbitrarily) used criterion of defining the NPSH required as the NPSH value at which head degrades by 3% from the non-cavitating value, one can construct a locus of the required NPSH as a function of the air ingestion level. Figure 4-3 shows four such points obtained by plotting the NPSH values for which head has degraded by 3% from the non-cavitating values. The plotted points are taken from the four curves shown in Figure 4-2 for air fractions of 0%, 3.3%, 6.6% and 9.9%, respectively. In order to establish a guideline for calculating the increased NPSHR in the presence of air, an arbitrary relationship is presented. This relationship is:

$$NPSHR_{air/water} = NPSHR_{water} \; (1+0.5 \; AF)$$

where AF is the air volume fraction in percent.

The relationship is shown in Figure 4-3 as a straight line. It is evident from the figure that the equation for NPSH requirements in the presence of air provides a margin above the values obtained by Merry [31]. For example at 2% air volume fraction the NPSH requirement is equivalent to that obtained with 3.3% air volume fraction. The conservatism used in establishing the straight line is arbitrary. However, it is felt necessary because of the limited amount of data available upon which to base such a guideline. It should be noted that the guideline is only intended for use for air volume fractions less than 2%.

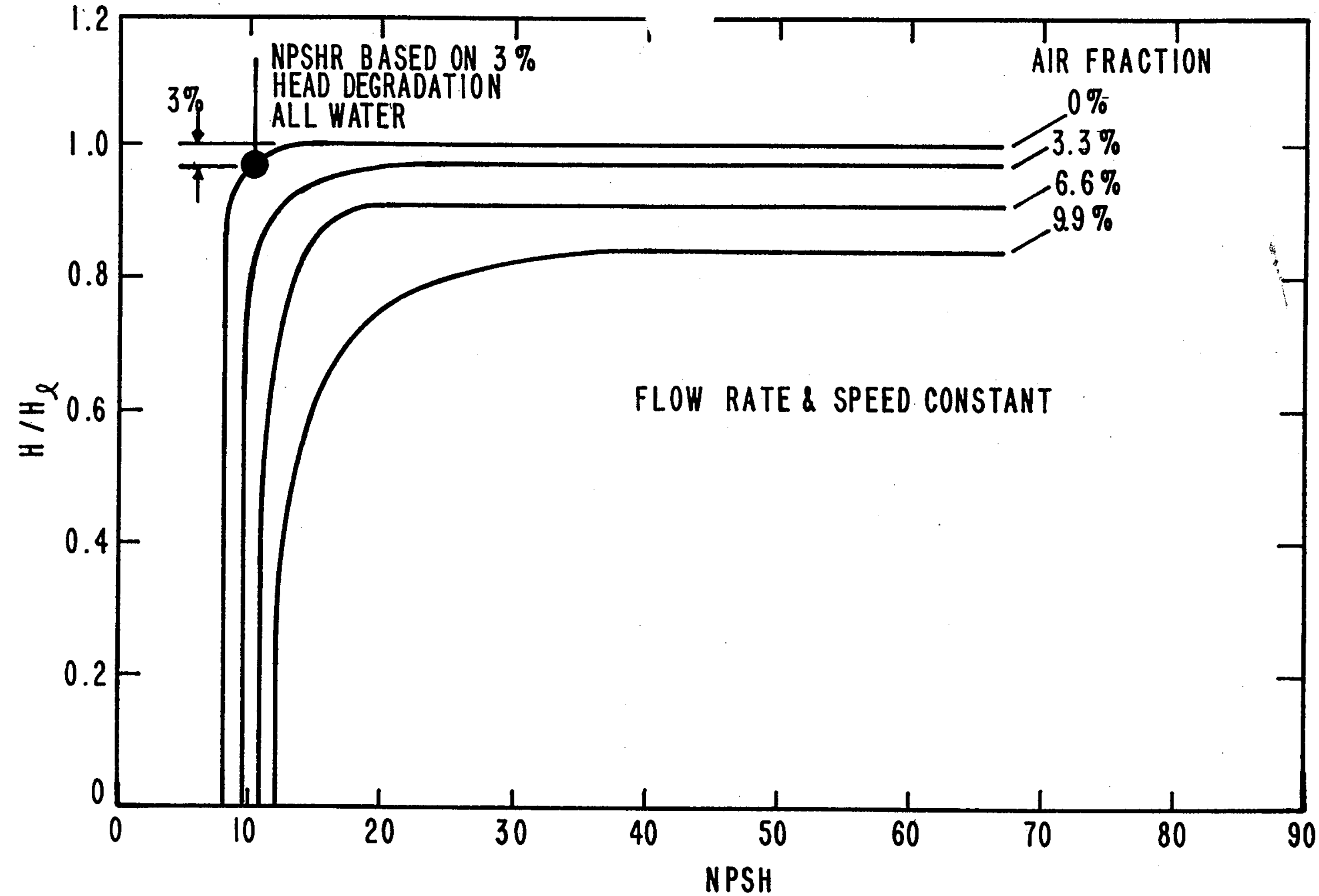

Figure 4-2. NORMALIZED HEAD VS. NPSH AT DIFFERENT VOID FRACTIONS FROM MERRY [31]

hot water
NPSHR

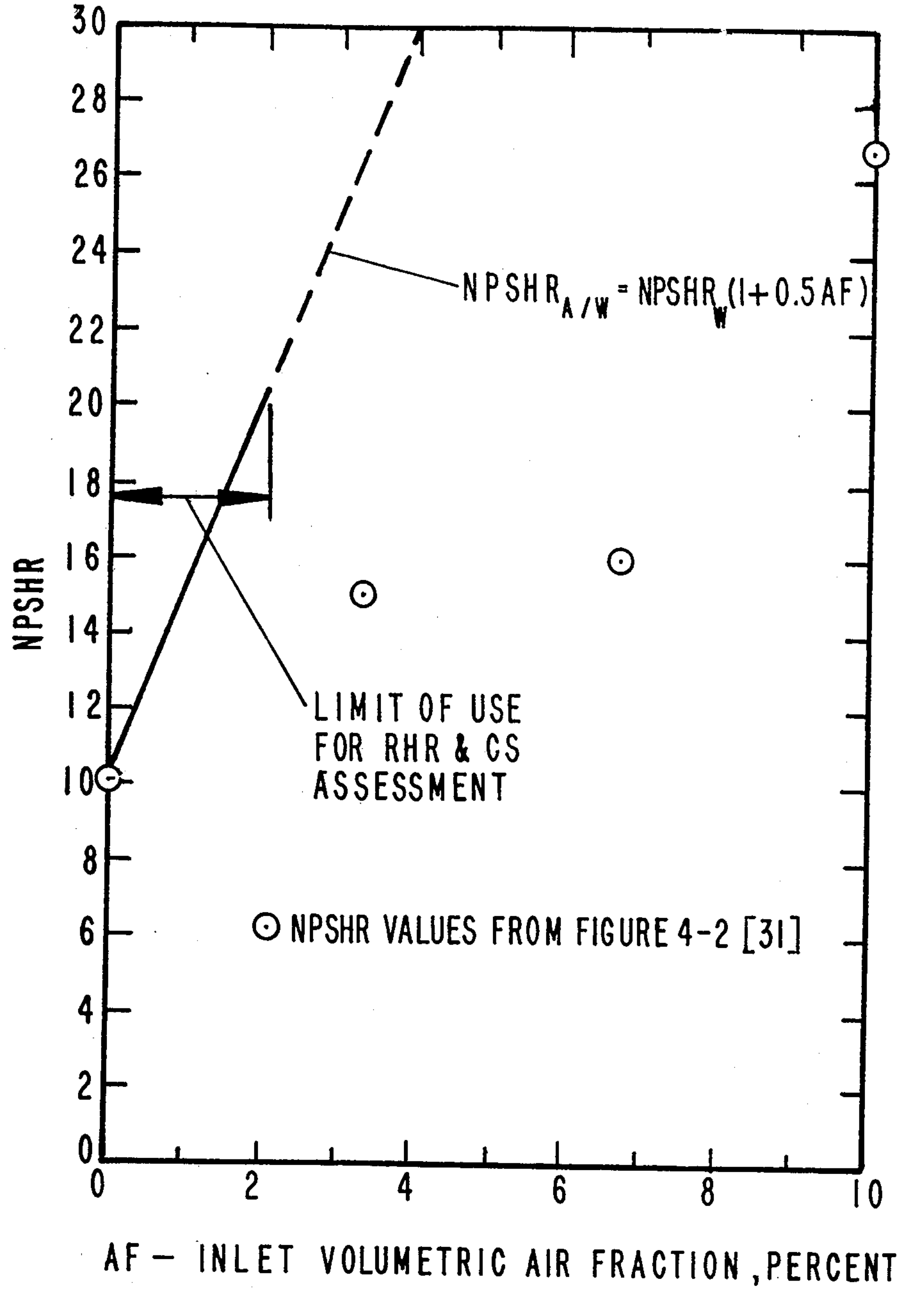

Figure 4-3. NPSH REQUIRED VS. AIR FRACTION BASED ON 3% HEAD DEGRADATION, MERRY [31]

From: Gerard O'Dwyer , E3
To: Steven Reynolds , R3
Date: 9/6/01 1:42PM
Subject: An addition to my July 24 DPV.

On September 5, 2001, you, Mr. Reynolds told me that the DPV panel would not be able to review my DPV for the next three weeks and if that would be all right with me. I do not believe it is my responsibility to decide how rapidly to review my DPV. I have already indicated the urgency of my beliefs by calling for the shut down of all operating reactors with Mark I and II containments. I am surprised that you want to delay for three weeks because I was criticized harshly by my divisional management for not promptly following up these issues with NRR (even though NRR did not followup appropriately and hence I wrote the DPV). I now respectfully request to add this email to my DPV. I did not add it earlier because I felt under extreme pressure to complete my routine inspections and other assignments and my DPV was only made, by my division director and branch chief, my second priority after my routine inspections. The following are more specific examples of where I disagree with the NRC policy specifically related to ECCS pumps' suctions.

I also disagree with the NRC allowing licensees to rely on the pressure in any LWR containment after a LOCA to provide adequate NPSH for the ECCS pumps. I feel this is incorrect and non-conservative because if the operators spray containment to reduce pressure after an accident they very easily could reduce the pressure below that counted on for NPSH for the ECCS pumps. Containment spray systems cannot be controlled adequately to ensure that the pressure is not reduced below that required. Operators can not control the spray down of containment to ensure that adequate NPSH is maintained and should not be burdened by being required to try to maintain the proper pressure after a LBLOCA.

I also disagree with NRR's assertion that the vapor pressure in the NPSHA equation is the only term that needs to be adjusted for hot water operation. I believe that the NRC and the industry have underestimated the NPSHR for the ECCS pumps after a LBLOCA. The Pump Handbook by Karassik et al (copy righted 1976, p. 2-157) stated "Field experience and carefully controlled lab experiments have indicated that pumps handling hot water or certain hydrocarbons may be operated safely with less NPSH than normally required for cold water." However p. 2-158 stated that "NPSH may have to be increased above the normal cold water value to avoid unsatisfactory operation when a) entrained air or other noncondensable gas is present in the liquid or (b) dissolved air or other noncondensable gas is present and the absolute suction pressure is low enough to permit release of the gas from solution." I believe that both of these statements are referring to NPSHR. I think these statements indicate that after blowdown the ECCS pumps will require even more NPSHR than what's listed on the vendor supplied pump curves and I do not believe that this increased NPSHR requirement has been considered in the post-LOCA NPSHR calculations. Neither the Standard Review Plan nor any other document I have reviewed considered that in the design.

CC: Darrell Schrum; James Caldwell; Jim Dyer; John Grobe; Ronald Langstaff; Roy Caniano

A14

From: Gerard O'Dwyer , R3
To: Kxr; Sar1 K O'Brien, R3
Date: Mon, Sep 17, 2001 10:25 AM
Subject: Fwd: I would like to change my DPV recommendation.

A/S

From: Gerard O'Dwyer / R3
To: DPV cc; Jim Dyer, R3
Date: Thu, Sep 13, 2001 5:00 PM
Subject: I would like to change my DPV recommendation.

In recognition that the devastating attacks on the United States of America on September 11, 2001, were acts of war as stated by President Bush, I would like to cancel my recommendation of July 24, 2001 that all operating reactors with Mark I or II containments be immediately shut down. The relative risk of a nuclear accident requiring mitigation by the ECCS pumps now seems to me to be relatively much less urgent compared to the present need for those reactors to provide power necessary for the defense of the United States and the prosecution of this war against the countries that support terrorism. I made my recommendation for immediate shut down when alternate power sources seemed stable and we were not at war. I now respectfully request to replace my previous recommendation with a recommendation that for all operating reactors with Mark I or II containments no more increases in licensed power be granted. I make this recommendation because I believe that running these reactors at higher power levels (some of them relatively old) will unacceptably increase the probability of an accident or transient that would shutdown the reactors and thereby interrupt the necessary power. Also, any ensuing accident would then be affected by all the issues in my DPV and would make our present situation significantly worse. If any uprates are granted, then my DPV issues should be addressed.

From: Darrell Schrum, R3
To: Kenneth Riemer; Steven Reynolds R3
Date: Thu, Sep 20, 2001 9:08 AM
Subject: Mr. O'Dwyer's Nuclear Safety Concerns

I believe we need answers to some of the attached questions to determine the risks to nuclear plants of Mr. O'Dwyers concerns. See attachment.....

CC: Gerard O'Dwyer

A/6

Here are some questions that may need answered before we can determine the risks of Mr. O'Dwyer's concerns.

1. What is the maximum percentage of air that can be trapped in water? At what percentage of air will the RHR Pumps and Core Spray Pumps become air bound? How long does the air stay in the water. Also, we need some data on how long it takes to get to various percentages of air during a LOCA.

2. We need calculations that show the number of seconds to blow the RV empty. In addition, we need the number of seconds for the blowdown of air and steam from the containment into the torus and the velocity of the steam through the downcomers into torus water. This would give a good indication of the penetration of the air/steam mix into the water. In addition, this would give a good indication of how long of time the steam and air are being mixed into the water. (These calculations are needed for all sizes of LOCA(s), because an intermediate size LOCA may be worse than a large break LOCA because of the additional time that gas is being mixed in torus water.)

3. How long are the pumps in minimum flow prior to opening the valves to the RV if off-site power is available? We need this time data from each nuclear plant to see which plant is most limiting. For Monticello it is around 10 seconds until injection time. The time to injection is important to compare with the timing of the air and steam injection into the torus at the same time. (Has a LOCA ever been evaluated with off-site power available?)

4. We need to know the configuration of the equipment in the torus. For example, some plants have the strainers in the same area as the downcomers. DAEC strainers and downcomers are in different zones of the torus. However, the licensee mentioned several plants with the strainers and downcomers located next to each other. These plants would appear to be very vulnerable to steam and air entrapment.

5. Is there a containment pressure that can prevent pumps from becoming air bound from air trapped in the water? Is more air trapped in water at higher pressures (Like CO2 in soda). Does the additional pressure help to keep the air in the water on its way through the pump?

6. Does power uprates with higher torus temperatures make the above conditions worse for air binding pumps? Has any licensee addressed this condition during their application for power uprate?

7. The action of the vacuum breakers is to release the air in the torus back into containment. How much of this air is recycled from the containment for a second and third time through the torus water. This could have a substantial effect on the amount of air trapped in the water. Has the recycled air condition been evaluated before?

8. We need a list of all of the weaknesses in the NRC's previous tests that simulated a LOCA into a test torus. Is this test a good/bad representative of the actual LOCA conditions? Did they simulate a pump start during the actual LOCA conditions? What was the configuration of the equipment in the simulated torus? What was the water temperature in the torus?

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

November 30, 2001

MEMORANDUM TO: Gerald O'Dwyer, Reactor Inspector
Division of Reactor Safety

FROM: Steven A. Reynolds, Deputy Director
Division of Reactor Projects

SUBJECT: DIFFERING PROFESSIONAL VIEW: ECCS SUCTION CONCERNS

The Ad Hoc Review Panel, for reviewing your Differing Professional View regarding ECCS suction concerns at BWRs with Mark I and Mark II containments, has reviewed your various emails and associated attachments, including your email of September 13, 2001. In your email dated September 13, 2001, you canceled your original issue and replaced it with a request that no more increases in licensed power be granted to operating reactors with a Mark I or Mark II containment. This new request was based on your belief that "running these reactors at higher power levels (some of them relatively old) will unacceptably increase the probability of an accident or transient that would shutdown the reactors and thereby interrupt the necessary power." You further stated that "If any uprates are granted, then my DPV issues should be addressed."

Based on our review of the information that you provided, the panel has developed the following issues that we need you to address before we can proceed:

1. Your concern appears to be that you believe, that for BWRs with Mark I and Mark II containments, there would be an ingestion of gas from the downcomer during some LOCAs that would result in subsequent gas binding of the LPCI pumps which would allow containment failure and the release of excessive radioactivity to the public. In light of your September 13, 2001 email, is this a clear statement of your concern? If not please provide the panel with a clear statement of your concern.

2. Assuming that your concern is correctly stated above, under what scenarios and conditions would your issue be of concern? (e.g., large, medium, or small break LOCAs; HPCI and RCIC suction from CST or other sources; LOOP LOCA or non-LOOP LOCA)

3. You reference Mark I containments with an ECCS suction header (where the velocity of water would increase substantially with initiation of the HPCI system); provide specific information concerning the header and why the substantial velocity increase because of having a header. Was the normal HPCI line-up (aligned to the CST) considered in formulating the concern?

A/7

4. Describe the reason or basis for disagreement with the NRR summary that "the swell (exclusion zone) will be limited to less than 1 diameter of the downcomer."

5. Explain why you believe that the phenomena (gas binding) would be greater for "no loss of off-site power" compared to "Loss of off-site power."

6. In your email with the heading "Supplement to Differing Professional View (DPV) of July 24, 2001", you state that "Gas binding or cavitation of the LPCI pumps would allow containment failure and the release of excessive radioactivity to the public". Explain the mechanism of how a loss of the LPCI pumps would lead to containment failure and the release of excessive radioactivity to the public.

I would appreciate your response by December 14, 2001.

cc: K. Riemer, DRS
D. Schrum, DRS
B. Berson, ORA

From: Gerard O'Dwyer , E3
To: Sar1 S Reynolds , E3
Date: Fri, Jan 11, 2002 3:49 PM
Subject: Response to your 11-30-01 questions.

Answer 1: The concern you stated is obviously one of my concerns, but is obviously not my only concern. There are many other concerns in my DPV and DPV supplements. For example, in my original DPV I included the attachments to my July 24, 2001 email which have additional concerns however you did not include the attachments in your statement of my DPV. Another example is that I added two concerns in my September 6, 2001 email.

Answer 2: As I have communicated repeatedly, one of my concerns is complete or partial vapor locking of the LPCI pumps. One of the potential sequences that I am most concerned about is a (large) medium break LOCA at one of the Dresden units (with no LOOP and no single active failure) occurs with blowdown (starting at about 5 seconds after the LOCA) forcing large amounts of nitrogen into the torus for about 15 seconds. The nitrogen will exhaust from the downcomer straight down into the bottom of the torus and then travel up the inside and outside of the torus, and completely or significantly displace the water near the entrance to the suction piping leading from the torus to the ECCS header. By five seconds after the LBLOCA, all LPCI and CS pumps would be running on min flow (500 gpm each) and large amounts of nitrogen would be pulled into those suction pipes from the torus to the ECCS header. These suction pipes are 2 ft. in diameter and there are four them. If the HPCI suction was lined up to the torus as allowed by the license, the HPCI pump would be running at full speed within five seconds which would be over 5000 gpm. The total ECCS pumps' flow would be 9000 gpm or 20 cubic feet/second. Even if only half of the 20 cubic feet/second sucked into the ECCS header suction piping during the 15 seconds would be entrained nitrogen, then in 15 seconds 150 cubic feet would be pulled into the suction piping. Then the LPCI pumps will go to full flow when the injection valves open. The LPCI injection valves will get a signal to open in 18 seconds after the LOCA and since they are gate valves they only need to be 7% open to allow full flow. The valves take 25 seconds to full open but full flow will be achieved in about 1.75 seconds since the reactor vessel will have blown down and therefore about 20 seconds after the LOCA occurs the LPCI pumps reach full flow and will rapidly pull the previously ingested nitrogen into the LPCI pumps. The LPCI pumps are vertical centrifugal pumps and very susceptible to gas binding. Thirty eight cubic feet per pump is more than enough to vapor lock the LPCI pumps completely or substantially. The LPCI pumps would be vapor locked and not work at all or not pump enough to meet safety function requirements. I am concerned because a large spectrum of LOCAs will also cause similar vapor locking because the initial blowdown causes huge amounts of nitrogen to be ingested. The ECCS pumps were inadvertently "designed" to fail because their suction source is designed to be a violent cauldron of gas, steam and water.

Answer three: See answer two. Even if HPCI lined up to CST, about 18 cu ft per punp would be ingested. Eighteen cubic feet per pump is still more than enough to vapor lock the LPCI pumps completely or substantially.

Answer 4: First, NRR is incorrectly using the word swell. Swell is where the the water in the torus is driven up and smashes into the top of the Torus. NRR stated for a large break LOCA that the swell would be less than one downcomer diameter which is about 2 ft. This is incorrect, for large break LOCA the swell is enough to drive the torus water up until it smashes into the top of the torus. what I am talking about is the penetration of the nitrogen blast from containment straight down filling the bottom of the torus and driving the torus water up until it smashes into the top of the torus. NRR stated that nitrogen blast from containment would go down into the torus water only to about 2 ft. and then take a right turn and spread out. This is contrary to experimental data and plants' design.

Answer 5: If a large break LOCA occurs with no loss of offsite power the ECCS pumps start in seconds. If a large break LOCA occurs with loss of offsite power the ECCS pumps' starts are delayed by about 37 seconds until they are loaded onto the diesels. The delay of 37 seconds gives some time for the nitrogen blown into the torus to rise to the top and not be available for ingestion.

A18

Answer 6:
If the LPCI pumps fail, then the core melts and there is more core damage than assumed in the present LOCA analysis. If the LPCI pumps fail, then they cannot spray down containment and the present analysis assumes containment spray. Higher containment pressure than analyzed will cause containment to fail or will require containment to be vented more than the present analysis.

CC: Darrell Schrum; James Caldwell; Jim Dyer; JMJ3; Kenneth Riemer; Ronald Langstaff; Tony D'Angelo

From: Darrell Schrum , L3
To: Kenneth Riemer , R2
Date: Tue, Jan 29, 2002 8:18 AM
Subject: Riemer Questions

Ken....Do the answers to the questions address your concerns? Or is additional clarification needed?

Gerry....Do the answers to the questions agree with what you were thinking?

CC: Gerard O'Dwyer; Steven Reynolds

A/9

1) The individual references Mark I containments with ECCS suction header (where the velocity of water would increase substantially with initiation of the HPCI system). I am not familiar with BWR ECCS suction headers (I've primarily seen individual suction taps for the ECCS). Please provide specific information concerning the header. Also please address whether the normal HPCI line-up (aligned to the CST) was considered in formulating the concern.

The information will be corrected. HPCI should be assumed to be in its normal line-up.

2) The referenced AEOD report assumes for the purpose of analysis that containment air was assumed to be the one and only source of gases dissolved in the pool water. This is not the case at BWR reactor sites. Please address the effects (if any) of inerting BWR containments with Nitrogen vice air.

Information Request will include nitrogen rather than air.

3) The submitter's reasons for disagreeing with the prevailing staff views are listed with one exception. Please list/describe the reason or basis for disagreement with the NRR summary that "...swell (exclusion zone) will be limited to less than 1 diameter of the downcomer.

Licensee's do have data on strainer loading during a LOCA. There is significant forces which can rip the flange off that is attached to the strainer. The licensee used this data when new larger strainers were installed. This certainly does not agree with the swell be limited to one diameter of the downcomer.

4) I'm unclear as to the basis or reason for the third concern listed on the second page of the submittal (concerned about other than the DBA). Please explain why the phenomena (gas binding) would be different for "no loss of off-site power" compared to "loss of off-site power".

The pumps start pumping sooner during a LOCA, since power is immediately available and there is no delay for the diesel coming up to speed and loading. Mr. O'Dwyer assumed much more air/steam mix during this early part of the LOCA.

5) Under "Assessment of Consequences" I agree that the loss (due to cavitation, gas binding, etc) of LPCI pumps is significant. However, I'm unclear as to the mechanism of how loss of the LPCI pumps would (emphasis added) lead to containment failure. Please provide specific information on how the loss of the LPCI pumps would lead to containment failure.

Mr. O'Dwyer assumed that the core would melt after the loss of all ECCS pumps. Do you have more information that Mr. O'Dwyer should know?

2.

From: Darrell Schrum /e3
To: Steven Reynolds /e3
Date: Thu, Jan 31, 2002 2:15 PM
Subject: Mr. O'Dwyers DPV Questions

I requested input from Mr. O'Dwyer and Mr. Riemer for the following list of questions. Mr. Langstaff said the questions were ok, with a minor change to the last question.

Let me know what you think. I am currently assigned to prep/inspect Braidwood P&IR, so I don't have too much time to invest in Mr. O'Dwyer's issue for the next few weeks. I think these questions will give us the information we need to make a decision. I believe that most of these questions can be answered in RIII, but NRR probably has more time. I have a lot of documentation from the internet that may take Mr. O'Dwyer closer to resolving his issue, including an international study for a suppression pool with a steam and air mix.

CC: John Jacobson

A/10

See Attachment #1 to this Memo for an Overview of Mr. O'Dwyer's Concerns With a LOCA in Mark I and Mark II Containments. Mr. O'Dwyer's main concern is that the ECCS pumps will become inoperable from nitrogen and steam in the suppression pool water during a LOCA

The following list of questions need to be answered to determine the significance of Mr. O'Dwyer's concern.

1. We need calculations that show how many seconds it will take for the reactor vessel to empty during a LOCA,. [The data we have indicates this time is approximately 20 seconds for a LBLOCA.]

2. We need the calculated velocity of nitrogen and steam in the suppression pool downcomers during the LOCA. This calculation should include the number of gallons of water that flashes to steam, the volume of the steam, and the cross sectional area of the downcomers. The velocity would give a good indication of the penetration and turbulence of the nitrogen/steam mix into the torus water.

[Licensee's do have LOCA strainer loading calculations for the new larger strainers which may be useful in determining the area covered by the steam and nitrogen in the suppression pool. Some of these calculations show that the LOCA force can rip the flange off the piping that is attached to the strainer (See NRC INSPECTION REPORT 50-237/01-09(DRS); 50-249/01-09(DRS)).

3. What is the maximum nitrogen solubility for water at various temperatures? At what percentage of nitrogen will the ECCS Pumps become inoperable?

[Use LOCA conditions for the nitrogen: containment pressure, torus water temperatures, significant water turbulence, etc.] [The centrifugal pump data we have indicates that there is a 10% reduction in pump capacity at 2% air, a 43% reduction in pump capacity at 4% air, and the destruction of the pump at 6% nitrogen.] [The data we have indicates 2.5% nitrogen per volume of water at 1 atmosphere pressure. According to Henry's Law there would be 7.5% nitrogen at 3 atmospheres pressure (45 pounds in containment)]
[
3. What is the time from the start of a LOCA until the start of the injection of water into the reactor vessel (data from different plants).

[The data we have indicates around 10 seconds with offsite power available and 20 seconds if the emergency diesel generator is required]

4. How far apart are the strainers and downcomers for those plants that have this equipment in same zones of the torus.

[These plants would appear to be very vulnerable to steam and nitrogen entrapment].

5. Did licensee's address the LOCA blowdown torus turbulence issue during their applications for power uprates? Some of these plants were permitted to have torus water temperatures above the boiling temperature of water (atmospheric pressure). Will some of the water turn to steam in the ECCS pumps and contribute to making the ECCS pumps inoperable?

6. We need a list of reasons why NRC's previous tests was an adequate simulation of a LOCA. Did the test simulate an actual pump start during the LOCA conditions? What was the configuration of the equipment in the simulated torus? What was the water temperature in the torus?

From: Darrell Schrum , E3
To: Kenneth Riemer; Steven Reynolds , E3
Date: Fri, Feb 1, 2002 7:03 AM
Subject: Fwd: Mr. O'Dwyers DPV Questions

Here is additional facts I found on the internet:

The solubility of gases in water drops off very rapidly above 70 degrees Fahrenheit. If the suppression pool heats up rapidly during a LBLOCA (power uprate may be an advantage) the nitrogen will rapidly escape to the surface of the water.

The condensation of steam mixed with air is very rapid. The international study used steam mixed with air at approximately 60 miles per hour. The zone of condensation was about 10 diameters from the downcomer (NRR gave Mr. O'Dwyer 1 diameter, which certainly doesn't seem possible). It appears that if the water stays in the bottom of the torus (steam blast doesn't keep a lot of water at top of the torus) that the steam will be condensed before it can be sucked into the strainer.

The international study (performed in 2000) indicates that all the questions associated with suppression pools were never (and still not) completely addressed. So NRR may not be able adequately answer all of the questions we propose to send to them.

I believe that if the blast of steam is 60 mph or less that we can guess that ECCS pumps will be operable. If the blast of steam is greater than 100 mph, no one can quess what will happen to the pumps.

CC: John Jacobson

A/11

From: Gerard O'Dwyer, C3
To: Darrell Schrum; Kenneth Riemer; Steven Reynolds C3
Date: Thu, Apr 11, 2002 1:59 PM
Subject: Re: DPV panel recommendation memo

The DPV author respectfully does disagree with the approach you have recommended. All Mark I and Mark II containment reactors should be shut down or at least all the power uprates granted in the last five years should be revoked. You state that it's not an immediate safety concern because the SDP only requires low-pressure injection of only one RHR pump or one core spray pump, however, the SDP is based on licensee studies that have not been verified or validated by the NRC and I differ with those studies, e.g., General Electric previously did not require core spray after a LOCA because of the assumption that steam cooling would cool the upper core for breaks allowing reflooding of the core to only 2/3 coverage. This was based on the old bottom peaked power distribution, however DAEC USAR, SECTION 6.3.1.1.2, "Reliability Requirements", pg 6.3-3, rev 15 - 5/00, Reliability requirement 6 required: for the LOCA analysis of reference 4b, long-term core cooling requires core reflood above TAF OR (core reflood to top of jet pump and one core spray pump operating). Reference 4b is: GE company, DAEC GE12 Fuel Upgrade Project, NEDC-32915P, Rev 0, Nov 1999. DAEC engineer Brad Hopkins told me that he was a former reactor physics engineer for General Electric and that the new longer life cores have a power distribution that is concentrated in the top of the core producing a spectral shift and producing more plutonium in the top of the core therefore steam cooling would no longer work and GE no longer takes credit for it. Therefore the SDP should at least require as a minimum one RHR pump and one core spray pump. The power uprates make core spray even more necessary. The DAEC core spray pumps will probably be unrecoverable because cavitation alone is very detrimental to mechanical seals and would be aggravated because the Duane Arnold core spray pump Seals are already operating beyond design (see Mr. Schrum's SSDI findings at DAEC). The SDP also assumes that the seals will operate correctly which is an incorrect assumption. The SDP further assumes that nothing else will go wrong which is a non-realistic assumption. The SDP has no defense in depth. At plants other than DAEC, there is no reasonable assurance that the operators will realign the core spray pump to the CST because they'll have no idea to do that because they will not know why the pump is not operating correctly because nowhere have I seen this issue addressed before certainly not in the EOPs. Operators will also be loathe to stop a pump in the middle of an accident and troubleshoot. Plus I believe some plants cannot even lineup core spray to the CST. By the time the operators lineup feed water, condensate or other sources the core will have overheated with no RHR or CS flow causing core damage. Containment spray will be needed to save containment and none of those sources can supply it only RHR. With no method or an inadequate method to cool the cores, the heat would result in substantial damage to the reactor cores. The radioactivity from the damaged core would probably escape from containment through the main steam isolation valves because the main steam isolation valves sometimes leak in excess of the TS maximum allowed leakage value. An example of such leakage is documented in DRESDEN NRC INSPECTION REPORT 50-237/00-13(DRP); 50-249/00-13(DRP). This would be worse than a nuclear accident as defined in the NRC Severe Accident Policy Statement (50 Federal Register 32138, August 8, 1985). The Statement defines a nuclear accident as those accidents which result in substantial damage to the reactor core, whether or not serious offsite consequences occur. Such an accident would violate licenses; the NRC's performance measures; release radioactivity in excess of license requirements and accident analyses; and destroy public confidence in nuclear safety.

The memo also does not address my DPV concerns with cavitation causing catastrophic failure of the seals especially in light of Mr. Schrum's findings at DAEC (see attachment one).

The memo also does not address my other DPV concerns. the following is verbatim from my 9/6/01 email:

I also disagree with the NRC allowing licensees to rely on the pressure in any LWR containment after a LOCA to provide adequate NPSH for the ECCS pumps. I feel this is incorrect and non-conservative because if the operators spray containment to reduce pressure after an accident they very easily could reduce the pressure below that counted on for NPSH for the ECCS pumps. Containment spray systems cannot be controlled adequately to ensure that the pressure is not reduced below that required. Operators

A/12

can not control the spray down of containment to ensure that adequate NPSH is maintained and should not be burdened by being required to try to maintain the proper pressure after a LBLOCA.

I also disagree with NRR's assertion that the vapor pressure in the NPSHA equation is the only term that needs to be adjusted for hot water operation. I believe that the NRC and the industry have underestimated the NPSHR for the ECCS pumps after a LBLOCA. The Pump Handbook by Karassik et al (copy righted 1976, p. 2-157) stated "Field experience and carefully controlled lab experiments have indicated that pumps handling hot water or certain hydrocarbons may be operated safely with less NPSH than normally required for cold water." However p. 2-158 stated that "NPSH may have to be increased above the normal cold water value to avoid unsatisfactory operation when a) entrained air or other noncondensable gas is present in the liquid or (b) dissolved air or other noncondensable gas is present and the absolute suction pressure is low enough to permit release of the gas from solution." I believe that both of these statements are referring to NPSHR. I think these statements indicate that after blowdown the ECCS pumps will require even more NPSHR than what's listed on the vendor supplied pump curves and I do not believe that this increased NPSHR requirement has been considered in the post-LOCA NPSHR calculations. Neither the Standard Review Plan nor any other document I have reviewed considered that in the design.

Finally, making this a candidate generic issue is not a properly timely response to my urgent concerns.

>>> Steven Reynolds 04/04/02 10:02AM >>>
Based on comments from Mr. Dyer, the panel recommendation memo has been modified. Concern 1 now has the previous concern #4 incorporated. The memo also has some assessment in it.

Please let me know if you have any comments.

thanks,
Steve

CC: Jim Dyer

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

Introduction

E1.4 10 CFR 50.59 Safety Evaluations and Screenings

a. Inspection Scope (37001)

The team reviewed completed SE(s) for design changes to the plant.

b. Observations and Findings

(1) RHR Mechanical Seal Failure

On August 2, 1999, the licensee performed a 10 CFR 50.59 evaluation (Safety Evaluation 99-041) to not inspect or flow test the RHR mechanical seal heat exchangers. Due to the configuration of the piping the licensee could not perform a flow test. The four RHR Heat Exchangers had not been tested or cleaned in 28 years. Following SE 99-041, the licensee changed the FSAR and the Technical Specification Basis to eliminate the need for heat exchanger cooling. The licensee assumed it could operate the RHR Pumps with the heat exchangers plugged for the remaining life of the plant.

The licensee based SE 99-041 on the vendor manual, "Installation, Operation, and Maintenance Instructions for Type 'U' Mechanical Seals", 1969, which stated that the mechanical seal components were designed for temperatures up to 450 degrees Fahrenheit (F). The licensee failed to understand that the critical temperature for operating the mechanical seals is the temperature allowed for the actual sealing surface face. The vendor data indicated that the maximum water temperature for these seal faces was 150 degrees F. Previously, the licensee had performed two heat exchanger flow calculations to keep the seal faces at 150 degrees F during pump operations above this temperature. These older calculations should have been a warning to the licensee that seal cooling was necessary.

Without cooling water to the mechanical seal the water in the seal will not support the sealing surfaces as a lubricant. The inspectors determined that, for seals of this type, temperatures above 180 degrees F that the water lubricating film is not thick enough to separate the sliding surfaces of the seal faces. This is caused because the viscosity of water decreases rapidly at higher temperatures. Above this temperature significant wear and temperatures occur at the seal faces. The frictional temperatures generated can exceed the temperature limits of the seal components. In addition, the water will vaporize between the seal faces and cause a destructive form of vibration called slipstick. The seals can fail in relatively short period time depending on the original condition of the seal. The licensee stated that leakage would be 40 to 60 gallons per minute with seal failure. All RHR and Core Spray Pumps have valves to isolate the pumps from the reactor vessel and/or torus.

The two limiting plant conditions that the seals would be operated at would be 358 degrees F for plant shutdown and 215 degrees F for torus cooling following a large break LOCA. The seals are designed for these temperatures only with seal cooling. The RHR pumps would be expected to operate at both of these temperatures. The Core Spray Pumps would only be expected to operate for torus cooling. During the inspection the licensee attempted to obtain design information from several vendors to allow operation of these seals without seal cooling. The licensee did eventually receive a memorandum "Alliant Energy Containment Spray Pumps - Upset Conditions", March 28, 2002, from Flowserve Corporation that stated that the seals could survive torus cooling temperatures following a LOCA without total failure, but with damage and several gallon per minute leakage. The vendor stated that accelerated mechanical seal face wear could be expected at approximately 200 degrees F. The vendor conditioned this statement with the fact that the amount of wear was difficult to predict and the seals not failing was conditional on corrosion, shaft run-out, and varying product characteristics. The licensee and vendor did not provide any test data for high temperature seal operation to support any of its statements that the mechanical seals would not fail in high temperature applications.

After receiving the above information from the vendor, the licensee stated that they would put the heat exchangers back into the Generic Letter 89-13 Program and take credit for seal cooling for the RHR Mechanical Seals. The seal cooling had never been actually isolated from flowing through the RHR heat exchangers, so no actual damage would have occurred to the RHR Pump Mechanical Seals. The licensee stated that the seal heat exchangers would be opened and cleaned during the next RHR pump outage. In, addition the heat exchangers would be put into the preventive maintenance program for periodic cleaning. The inspectors considered the pending corrective actions for the RHR Pumps acceptable.

The licensee performed an Operability Evaluation (AR#30414), March 28, 2002, for continued plant operations. The licensee used the Core Spray Pumps evaluation to bound the RHR and Core Spray Pumps because the Core Spray Pumps do not have seal coolers and these pumps operate at a higher RPM (more frictional heat in the seals). The inspectors considered the Operability Evaluation acceptable, however it contained significant weaknesses. One significant weakness was that the licensee used the original design specifications to determine the acceptability for continued use of the RHR and Core Spray Pump Seals without seal cooling. The original GE design specification for pump seals specifies that the seals be capable of operation at temperatures of 212 degrees F for 1 day and operation at temperatures of 200 degrees F for 6 months. The licensee did not receive a seal that meets these design requirements during plant construction. The current seals can only survive at these temperatures for a short time period without cooling. The licensee stated that additional information would be provided for this condition. Also, the licensee stated that this was a potential Part 21 issue. This generic condition of the seals appears to be applicable to other BWR plants. A second weakness in the Operability Evaluation was that shutdown cooling temperatures were not addressed.

The inspectors considered the RHR pump seals operable based on the fact that water could still be seen flowing in the RHR Heat Exchanger Sight Glasses. In addition, no RHR Seals were damaged during the last shutdown of the plant when the seals were subjected to temperatures above 300 degrees F. This indicated that the seals were still being cooled and the heat exchangers probably had not been plugged during this cycle of plant operations.

The Core Spray Pump Seals design do not meet the temperature requirements (215 degrees F) for torus cooling following a LOCA without being damaged. The vendor stated that several gallon per minute leakage could be expected from these seals after operating at higher temperatures. These seals were never tested at these temperatures, so final seal condition is an engineering judgement. The vendor recommended that these seals be replaced with seals with that have a reduced amount of wear at high temperatures. However, the vendor stated that leakage of several gallons a minute could still be experienced. The inspectors believe that seal cooling appeared to be necessary to meet the original design requirements for the Core Spray Pumps.

Several evaluations are needed if the licensee decides to continue to operate the Core Spray Pumps with the current seals installed. On April 30, 2001, the licensee submitted analysis "Safety Evaluation Input For Alternative Source Term Technical Specification Change Amendment For Duane Arnold Energy Center" to support Amendment 240. This analysis only considered the equivalent of 1.5 gallons per minute unfiltered ECCS leakage during a Design Basis Accident. The licensee received Amendment 240, "Duane Arnold Energy Center - Issuance of Amendment Regarding Alternative Source Term", July 31, 2001. Using the current mechanical seals, the leakage from one Core Spray Pump during an accident could exceed this contribution to the source term. This additional source term should be included in the analysis. The inspectors believe that control room habitability could also be effected by the change in the source term. In addition, several gallons per minute of 215 degree F water/steam will contribute significant heat to the Corner Rooms during an accident. Currently, the heat load calculations have not been performed for significant heat loads identified by the inspectors as being included in the Monticello Corner Room Heatup Calculations but not included in DAEC Calculations. AR 30302 "Review NE & SW Corner Rooms, RHR, Core Spray, Heat Load Calculations 466-M-003", March 21, 2002, was written by the licensee to perform this calculation. The additional heat load from leakage will need to be included in these calculations. The EQ qualification of 140 degrees F for the corner room equipment may be exceeded with the Core Spray Pump potentially making the RHR pumps inoperable. In addition, room flooding will need to be considered an issue for long term cooling following an accident. Also, the additional heat from not have cooling to seals would allow significant heat to travel through the shaft from the pump to motor potentially exceeding the motor winding temperature or motor bearing oil temperature. The licensee did not address any of the above issues in the Operability Evaluation.

SE 99-041 was not only inadequate but contained a significant error for assuming the RHR pumps could be operated at 450 degrees F. As a result of the error, the above mentioned risks to the plant were not evaluated. In addition, the evaluation contributed significantly to shutdown risk because the RHR seals would fail if they were used at 358 degrees F. without cooling.

The licensee stated in the conclusion of the operability evaluation that a more exhaustive evaluation and analysis would be required to conclude that RHR Pump Seals would perform their design function. The inspectors believe that the licensee needs to address all of the above mentioned issues for both the Core Spray Pumps and the RHR Pumps. The significance of RHR pump issue is dependent on whether the Seal Cooler Heat Exchangers are found plugged and not capable of performing their function. The licensee issued AR 30234, dated March 17, 2002, to resolve the issues related to the evaluation of the seals. These issues will be tracked as Unresolved Item (50-331/2002-011-01). Pending the outcome of the above evaluations and inspections, these issues will be evaluated in the Significance Determination Process.

Conclusions

The team determined that a significant error occurred during the preparation of Safety Evaluation 99-041. As a result of the error, the licensee increased the risk to the plant during shutdown and LOCA conditions. The licensee must perform additional evaluations for continued use of the Core Spray Pump seals without cooling. In addition, additional analysis and calculations are required to support Operability Evaluation (AR#30414), These issues will be tracked as Unresolved Item. Pending the outcome of the above evaluations and inspections, these issues will be evaluated in the Significance Determination Process.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/249 2000003-03	URI	The licensee must perform additional evaluations for continued use of the Core Spray Pump seals without cooling. In addition, additional analysis and calculations are required to support Operability Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

April 8, 2002

MEMORANDUM TO: J. E. Dyer
Regional Administrator

FROM: Steven A. Reynolds, Deputy Director
Division of Reactor Projects

SUBJECT: AD HOC REVIEW PANEL RECOMMENDATION REGARDING A DIFFERING PROFESSIONAL VIEW ON BWR ECCS SUCTION CONCERNS

The purpose of this memorandum is to provide you with the Ad Hoc Review Panel's recommendation regarding the Differing Professional View (DPV) on BWR (with Mark I or II containments) ECCS suction concerns. In general terms, the prevailing staff's view is that the potential for gas binding of low pressure ECCS pumps after a Loss of Coolant Accident (LOCA) is low. The DPV is that the potential for gas binding of the pumps is not low. Furthermore, the basis for the prevailing staff view appears to come from a March 31, 1982, AEOD memorandum (AEOD/E218) titled: Engineering Evaluation - Potential for Air Binding or Degraded Performance of BWR RHR System Pumps During the Recirculation Phase of a LOCA. This evaluation addresses the recirculation phase of an accident, whereas the DPV is that the blow-down phase of the LOCA is a significant contributor to the potential gas binding or vapor locking of the pumps. The AEOD evaluation states: "The large bubbles generated by the initial blow-down of drywell air into the torus pool was neglected in this assessment since this air source would be expected to rise quickly out of the pool water and become part of the pressurized air-vapor space trapped above the pool surface." The DPV is that these bubbles can readily be transported into the low pressure ECCS piping and cause gas binding of the pumps.

Members of the Ad Hoc Review Panel spent considerable effort to fully understand and better characterize the concerns of the author of the DPV. Working with the DPV author, the Ad Hoc Review Panel has characterized the concerns as follows:

CONTACT: Steven A. Reynolds/DRP
630/829-9601

A/13

The DPV is that the possibility of gas binding/vapor locking for BWRs with Mark I or Mark II containments in low pressure ECCS pumps is higher than postulated by the prevailing staff view, and that the basis for the prevailing staff view does not adequately address the concern. Specific issues related to this DPV are as follows:

a. The prevailing staff's view is that the potential for gas binding or degraded performance of BWR RHR pumps during/after a LOCA is low. The DPV is that the low pressure pumps could fail to perform their safety function due to gas ingestion, vapor locking, or cavitation after a LOCA. (The DPV author did not provide any technical justification to support this position.)

The prevailing staff's view is that for Mark II containments, the gas discharge was looked at for the design basis accident (a loss of off-site power with a LOCA (LOOP/LOCA)). The DPV author's concern is that the most limiting scenario would not be the design basis LOOP/LOCA but rather, a LOCA <u>without</u> a loss of off-site power. Based on the panel's engineering judgement, the DPV author's opinion that the large break LOCA (LLOCA) <u>without</u> a loss of off-site power is of more concern than a LOOP/LOCA seems possible in that the ECCS pumps will start sooner without a LOOP and that this could happen before the bubbles generated by the blow-down had a chance to rise out of the pool water. Additionally, based on the panel's engineering judgement, some plants in response to NRC Bulletin 96-03: Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors may have installed larger strainers that may be closer to the down-comers and thus be able to more readily transport the air bubbles to the pumps.

b. The prevailing staff's view is that the AEOD evaluation (AEOD/E218) calculations assume that the volume of gas that is of concern is limited to the assumed volume of RHR suction piping. The DPV author does not agree with this assumption because it does not include the volume of water in the torus, which also contains dissolved gasses. The DPV author's opinion is that the total volume of available water (water in the piping and torus) needs to be used to determined the amount of gas that may result. (The DPV author did not provide any technical justification to support this position.)

c. The prevailing staff's view is that the swell/exclusion zone in the torus post-LOCA will be limited to less that one diameter of the containment down-comer. The DPV author's position is that the intrusion of non-condensable gasses into the torus (and hence available to impact the low pressure ECCS pump suction) would be greater than assumed in the analysis. The DPV author did not provide a technical basis, but just believes that it might be more than 2 or 3 diameters.

As previously communicated to you, the panel concluded that these concerns did not constitute an immediate safety concern. The reasoning for that conclusion is as follows:

> For a BWR, a large break LOCA (LLOCA) has an initiating frequency (generic plant value) of approximately 3E-5 (NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995"). Evaluating the dominant sequences for LLOCA results in the following sequence: LLOCA * LPI, where LPI is either one of four RHR pumps or one of two Core Spray (CS) pumps.
>
> Assuming a LLOCA with just one of any of the four RHR/LPCI or just one of the two CS pumps available to inject water into the reactor vessel, the issue would result in a change in CDF of approximately 1E-7. This would result in an item with very low safety significance (GREEN). If the assumption is that all six low pressure ECCS pumps were lost, giving the operators reasonable credit to recover one core spray pump by realigning it to the condensate storage tank would result in a change in CDF of 1E-6, which is considered an item of low to moderate safety significance (WHITE).
>
> Though not credited in the above discussion, the following systems are referenced in licensees' EOPs and would be available to mitigate the consequences of the event: feedwater, condensate, condensate booster pumps, heater drain pumps, service water, fire water, and control rod drive water. For a small break LOCA under the same set of assumptions (i.e., complete loss of all low pressure ECCS pumps), the availability of HPCI and RCIC results in a change in CDF of approximately 1E-8.

The panel members have conducted an evaluation of the DPV that involved reviewing NRC generated documents, Nuclear Energy Agency, Committee on the Safety of Nuclear Installations (NEA/CSNI) generated documents, and other publicly available information. (Attached is a list of the material reviewed by the panel.) Through a dozen or more meetings and email messages, the panel members periodically discussed the DPV and the status of the panel's review with the DPV author. The panel members also conducted informal phone calls with NRR technical staff and discussed the DPV. The panel was unable to find any evidence to support the statement in the AEOD evaluation that the air bubbles from initial blow-down can be neglected. The panel was unable to find any evidence that an evaluation was performed for plants that have installed larger strainers and the possible resultant effect of transporting more air to the pumps. The panel was also unable to find evidence that would show that the DPV was correct. Based on engineering judgement, the panel concluded that the DPV warrants further review by NRC staff with more technical expertise in this area.

The panel recommends that the DPV be considered a Candidate Generic Issue in accordance with Management Directive 6.4: Generic Issues Program.

Attachment: List of Documents Reviewed

cc: J. McDermott, HR/OD
J. Caldwell, RIII
K. Riemer, RIII
D. Schrum, RIII

Attachment

Documents and Material Reviewed by the DPV Ad Hovc Review Panel

Management Directive 10.159: "Differing Professional Views or Opinions"

AEOD/E218, dated March 31, 1982: "Engineering Evaluation - Potential For Air Binding Or Degraded Performance Of BWR RHR Pumps During The Recirculation Phase Of a LOCA"

NUREG/CR-2792: "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions"

NUREG /CR-5750: "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995"

NRC Bulletin 96-03: "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors"

Various generic General Electric BWR system descriptions, diagrams, and EOP guidance

NEA/CSNI/R(2001)4 C1-15 One and Two Phase Pump Behavior (B1.12) / C1-16 Non-Condensable Gas Effect (B1.14) .

NEA/CSNI/R(2001)4 Validation Matrix for the Assessment of Thermal-Hydraulic Codes for VVER LOCA and Transients, June 2001

NEA/CSNI/R(2001)3 Bubbler Condenser Related Research Work, February 2001 SBLOCA Oscillatory Condensation and Chugging More Likely and Duration Longer

Meier, M., G. Yadigaroglu, *Numerical and Experimental Study of Large Steam-Air Bubbles Injected in a Water Pool*, Swiss Federal Institute of Technology, Züürich, Switzerland. (http://www.lkt.mavt.ethz.ch/~meier/nse.pdf)

Specialized Test Facilities (www.nuc.berkeley.edu/design/abwr.spec-test.htm)

Predicting NPSH For Centrifugal Pumps (www.pump.zone.com/articles/00/dec/feature1/.htm)

Multiphase Flow R&D (www.swri.org/3PUBS/BROCHURE/D04/MULTIFLO/multiflo.HTM}

Cavitation and NPSH (www.pricepump.com/pumpschool/psles3.htm)

Vaporization Cavitation/Vortexing Liquid/NPSHA Calculating Net Positive Suction Head Available in USCS Units/Flow Turbulence Cavitation/Suction Specific Speed (http://www.menallyinstitute.com/CDweb.u-toz-html.v026.htm)

Documents and Material Reviewed by the DPV Ad Hovc Review Panel (con't)

Meier, M., Andreani, M., Yadigaroglu, G. (1998), *Experimental Study of Large Steam-Air Bubbles Condensing in a Suppression Pool*. HTD-Vol. 361-5, Proceedings of the ASME Heat Transfer Division, Volume 5, ASME 1998 (1998 ASME International Congress and Exhibition, November 15-20, 1998, Anaheim, CA), 489-500. (http://www.lkt.iet.ethz.ch/~meier/work.html)

High Speed Fluid Dynamics (http://t3.lanl.gov/secondlevel/pubs/biblio55-80.htm) This is list of all known publications by Group T-3 for numerical computing methods for hydrodynamics problems. LA and LAMS numbers refer to reports issued by the Los Alamos National Laboratory.

The Gas Laws (http://oak.cats.ohiou.edu/ds106488/The%20gas%20laws.htm)

Henry's Law (http://www.nidlink.com/-jfromm/chapt7.htm)

Dissolved Oxygen (http://www.vcnet.com/koi_net/do.htm)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

April 29, 2002

MEMORANDUM TO: Gerard O'Dwyer, Mechanical Engineer
Division of Reactor Safety

FROM: J. E. Dyer JEDyer
Regional Administrator

SUBJECT: RESOLUTION OF DIFFERING PROFESSIONAL VIEW ON BWR ECCS SUCTION CONCERNS

I have reviewed the report of the Differing Professional View (DPV) panel concerning BWR ECCS Suction Concerns which you raised in your July 24, 2001 e-mail and supplements. A copy of the panel's report and recommendation is attached. I agree with the panel's conclusion that this issue could benefit from further review as a candidate generic issue, but does not appear to be an immediate safety issue.

Therefore, by copy of this memorandum and in accordance with NRC Inspection Manual Chapter 0970, "Potentially Generic Items Identified by Regional Office," I have directed DRS to generate the appropriate referral memorandum to NRR by COB May 10, 2002, and provide you a copy of the referral.

I appreciate and commend your willingness to utilize the DPV process. I am aware that we did not meet the timeliness goals for resolution of your DPV specified in Management Directive (MD) 10.159, but I understand that you were advised of the reasons for the delay, i.e., the complexity of the issue, the need to coordinate with headquarters, and schedule difficulties. In accordance with the MD, a summary of the issue and its disposition will be included in the Weekly Information Report to advise interested employees of the outcome. DPVs are not normally made available to the public. However, if you would like to have your DPV case file made public, with or without the release of your name, please contact Bruce Berson.

Our review of your DPV will be considered complete following issuance of the referral memorandum to NRR. Should you wish, you may then initiate the Differing Professional Opinion process as described in Management Directive 10.159.

Attachment: As stated

cc w/att.: J. Grobe, DRS

CONTACT: Bruce Berson/ORA
630/829-9653

A/14

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

April 8, 2002

MEMORANDUM TO: J. E. Dyer
Regional Administrator

FROM: Steven A. Reynolds, Deputy Director
Division of Reactor Projects

SUBJECT: AD HOC REVIEW PANEL RECOMMENDATION REGARDING A DIFFERING PROFESSIONAL VIEW ON BWR ECCS SUCTION CONCERNS

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Members of the Ad Hoc Review Panel spent considerable effort to fully understand and better characterize the concerns of the author of the DPV. Working with the DPV author, the Ad Hoc Review Panel has characterized the concerns as follows:

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The panel recommends that the DPV be considered a Candidate Generic Issue in accordance with Management Directive 6.4: Generic Issues Program.

Attachment: List of Documents Reviewed

cc: J. McDermott, HR/OD
J. Caldwell, RIII
K. Riemer, RIII
D. Schrum, RIII

Documents and Material Reviewed by the DPV Ad Hovc Review Panel

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AEOD/E218, dated March 31, 1982: "Engineering Evaluation - Potential For Air Binding Or Degraded Performance Of BWR RHR Pumps During The Recirculation Phase Of a LOCA"

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NEA/CSNI/R(2001)3 Bubbler Condenser Related Research Work, February 2001 SBLOCA Oscillatory Condensation and Chugging More Likely and Duration Longer

Meier, M., G. Yadigaroglu, *Numerical and Experimental Study of Large Steam-Air Bubbles Injected in a Water Pool*, Swiss Federal Institute of Technology, Züürich, Switzerland. (http://www.lkt.mavt.ethz.ch/~meier/nse.pdf)

Specialized Test Facilities (www.nuc.berkeley.edu/design/abwr.spec-test.htm)

Predicting NPSH For Centrifugal Pumps (www.pump.zone.com/articles/00/dec/feature1/.htm)

Multiphase Flow R&D (www.swri.org/3PUBS/BROCHURE/D04/MULTIFLO/multiflo.HTM}

Cavitation and NPSH (www.pricepump.com/pumpschool/psles3.htm)

Vaporization Cavitation/Vortexing Liquid/NPSHA Calculating Net Positive Suction Head Available in USCS Units/Flow Turbulence Cavitation/Suction Specific Speed (http://www.menallyinstitute.com/CDweb.u-toz-html.v026.htm)

Documents and Material Reviewed by the DPV Ad Hovc Review Panel (con't)

Meier, M., Andreani, M., Yadigaroglu, G. (1998), *Experimental Study of Large Steam-Air Bubbles Condensing in a Suppression Pool*. HTD-Vol. 361-5, Proceedings of the ASME Heat Transfer Division, Volume 5, ASME 1998 (1998 ASME International Congress and Exhibition, November 15-20, 1998, Anaheim, CA), 489-500. (http://www.lkt.iet.ethz.ch/~meier/work.html)

High Speed Fluid Dynamics (http://t3.lanl.gov/secondlevel/pubs/biblio55-80.htm) This is list of all known publications by Group T-3 for numerical computing methods for hydrodynamics problems. LA and LAMS numbers refer to reports issued by the Los Alamos National Laboratory.

The Gas Laws (http://oak.cats.ohiou.edu/ds106488/The%20gas%20laws.htm)

Henry's Law (http://www.nidlink.com/-jfromm/chapt7.htm)

Dissolved Oxygen (http://www.vcnet.com/koi_net/do.htm)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

May 10, 2002

MEMORANDUM TO: S. F. Newberry, Director
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

FROM: John A. Grobe, Director
Division of Reactor Safety

SUBJECT: POTENTIALLY GENERIC SAFETY ISSUE - BWR ECCS SUCTION CONCERNS

As discussed with L. B. Marsh, NRR, Roy Caniano, and John Jacobson of my staff on May 10, 2002, this is to inform you of a potentially generic safety issue identified by a Region III inspector. The issue pertains to the possible failure of low pressure emergency core cooling systems (ECCS) due to unanticipated, large quantities of entrained gas in the suction piping from boiling water reactor suppression pools. The issue is applicable to Mark I or Mark II containments during large or medium break loss of coolant accidents and could potentially cause pump failure or degraded performance due to gas binding, vapor locking, or cavitation.

Prior AEOD evaluation of this issue addressed the recirculation phase of this issue; however, did not address the large air bubbles generated during initial blow-down because the gas was expected to rise quickly out of the pool water. (Reference AEOD memorandum dated March 31, 1982, (AEOD/E218), "Engineering Evaluation - Potential for Air Binding or Degraded Performance of BWR RHR System Pumps During the Recirculation Phase of a LOCA.") However, due to the violent nature of the blow-down during the above accidents, there is a question regarding the gas that may become entrained in the suction flow to the ECCS pumps as a result of turbulence or mixing effects. The question is whether sufficient gas will get entrained in the suction flow to the ECCS to cause failure or degraded performance of the pumps.

There are several specific aspects that pertain to the above general concern for air entrainment in the suction flow to the ECCS pumps:

a. One of the bounding design basis accidents is a loss of off site power combined with a loss of coolant accident (LOOP/ LOCA). While this may be bounding from an ECCS performance perspective, it may not be bounding from a gas entrainment perspective. Because the pumps will start sooner during a LOCA without a LOOP, bubbles generated during the initial blow-down may not have risen to the surface and more may become entrained in the ECCS suction piping. Since a LOCA without a LOOP was not considered, this aspect should be considered for further evaluation.

CONTACT: John Jacobson, DRS
(630) 829-9736

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b. The AEOD evaluation, for potential air binding or performance degradation of RHR pumps, only used the volume of water in the RHR suction piping to determine the amount of dissolved gas. However, the amount of gas that is potentially available to affect pump performance is the total volume of water in the suction piping and the suppression pool. The potential for pump air binding or performance degradation may need to consider the total volume of available water in determining the volume of gas.

c. The swell/exclusion zone in the torus after a LOCA is considered to be limited to less than one diameter of the down-comer pipe. There does not appear to be a technical basis for this limitation, and it may not be conservative. The intrusion of non-condensable gas into the torus may be greater and the effect will potentially be worse due to the larger suction strainers installed in response to NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors." Adequate bases to limit the exclusion zone to less than one diameter of the down-comer pipe should be established, especially with respect to the recently installed larger suction strainers.

The above issues were identified during the course of several inspections by researching available information from licensee and the NRC. No known equipment or component failures led to the identification of this issue. We believe the issue does not pose an immediate safety concern, however, could benefit from further review as a potential generic issue.

cc: G. O'Dwyer, DRS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 28, 2002

MEMORANDUM TO: Farouk Eltawila, Director
Division of Systems Analysis & Regulatory Effectiveness
Office of Nuclear Regulatory Research

FROM: Scott F. Newberry, Director
Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research

SUBJECT: POTENTIALLY GENERIC SAFETY ISSUE - BWR ECCS SUCTION CONCERNS

I am forwarding to you a memorandum from Jack Grobe, Region III on a BWR ECCS suction concern involving gas entrainment during a LOCA. They note that the issue does not pose any immediate safety concern, but are requesting review as a potential generic issue.

Attachment: As stated

cc: A.Thadani/J. Strosnider, RES
J. Dwyer, Region III
J. Grobe, Region III
S. Collins, NRR
G. Holahan, NRR
D. Matthews, NRR
H. Vandermolen, RES

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

May 10, 2002

MEMORANDUM TO: S. F. Newberry, Director
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

FROM: John A. Grobe, Director
Division of Reactor Safety

SUBJECT: POTENTIALLY GENERIC SAFETY ISSUE - BWR ECCS SUCTION CONCERNS

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The above issues were identified during the course of several inspections by researching available information from licensee and the NRC. No known equipment or component failures led to the identification of this issue. We believe the issue does not pose an immediate safety concern, however, could benefit from further review as a potential generic issue.

cc: G. O'Dwyer, DRS

Potential for Air Binding or Degraded Performance of BWR Residual Heat Removal System Pumps During the Recirculation Phase of a Loss-Of-Coolant Accident

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ABSTRACT

An AEOD report regarding the potential for air binding or degraded performance of boiling-water reactor residual heat removal (RHR) system pumps during the recirculation phase of a loss-of-coolant accident is presented. This potential, which is due to air bubble generation in the torus pool during the blowdown phase, has been studied, with the concerns being identified as 1) the degraded capability of the RHR system pumps due to air bubble entrainment, and 2) attendant pumping of a water-air mixture through the RHR torus-to-pump suction piping. Air binding of a pump due to bubble rise coalescence potentially could be an associated concern and has also been assessed.

Related NRC Generic Communications: IN83-77, IN87-57, IN88-23, IN88-23S1, IN88-23S2, IN88-23S3, IN88-23S4

[NRC Home Page] [Nuclear Reactors] [PWR Studies] [BWR Studies] [Activity & Human Factors Studies] [Topical Index]

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NUREG/CR-5750
INEEL/EXT-98-00401

Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995

Idaho National Engineering and Environmental Laboratory

U.S. Nuclear Regulatory Commission
Office for Analysis and Evaluation of Operational Data
Washington, DC 20555-0001

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EXECUTIVE SUMMARY

This report presents an analysis of initiating event frequencies at United States (U.S.) nuclear power plants. The evaluation is based primarily on the operating experience from 1987 through 1995 as reported in Licensee Event Reports (LERs). The objectives of the study are: (1) provide revised, historical frequencies for the occurrence of initiating events in U.S. nuclear power plants, (2) compare these estimates based on operating experience to estimates used in probabilistic risk assessments (PRAs), individual plant examinations (IPEs), and other regulatory issues; and (3) review the operating data from an engineering perspective to determine trends and patterns of plant performance on a plant-type [i.e., pressurized water reactor (PWR) or boiling water reactor (BWR)], plant-specific, and industry-wide basis.

This study used as one of its sources of data the operating experience from 1987 through 1995 as reported in LERs. The Sequence Coding and Search System (SCSS) database was used to identify LERs for review and classification for this study. Each LER was reviewed from a risk and reliability perspective by an engineer with nuclear power plant experience. Based on the LER review, approximately 2,000 reactor trip events were analyzed with regard to their effect on plant performance.

For some initiators whose frequency is low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate their frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs).

Major Findings

This report provides information on frequencies, trends, and between-plant variation for initiating events. An evaluation of the results indicates that:

- Combined initiating event frequencies for all initiators calculated from the 1987–1995 experience are lower than the frequencies used in NUREG-1150, *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, and IPEs by a factor of five and four, respectively.

- General transients constitute 77% of all initiating events. Events that pose a more severe challenge to the plant's mitigation systems (nongeneral transients) constitute the remaining 23%.

- Over the nine-year span considered by this report, either a decreasing or constant time trend was observed for all categories of events. A decreasing trend was identified in approximately two-thirds of the more risk-significant categories that had sufficient data for trending analysis. The overall initiating event frequency decreased by a factor of two to three during the nine-year span. Most risk-significant initiator frequencies (such as total loss of feedwater flow, loss of instrument or control air, inadvertent closure of all main steam isolation valves (MSIVs), and total loss of condenser heat sink for BWRs) decreased at a faster rate than the overall initiating event frequency.

- Loss-of-coolant accident frequencies are lower than those used in NUREG-1150 and industry-wide IPEs.

- The frequencies (per critical year) estimated from the 1987–1995 experience for the risk-significant categories and general transients are the following. All but the first show a decreasing trend, and the values presented here apply to 1995.

–	Loss of Offsite Power (PWR and BWR)	4.6E-2
–	Total Loss of Condenser Heat Sink: PWR	1.2E-1
–	Total Loss of Condenser Heat Sink: BWR	2.9E-1
–	Total Loss of Feedwater Flow (PWR and BWR)	8.5E-2
–	General transients: PWR	1.2
–	General transients: BWR	1.5

For LOCA categories, the frequencies were evaluated using data and information prior to 1987 due to their relatively low frequency and the corresponding sparseness of data. No pipe break LOCA events were found in the U.S. operating experience. For the small pipe break LOCA frequency, the estimate from WASH-1400, *Reactor Safety Study,* was updated using U.S. reactor experience. For medium and large pipe break LOCAs, frequency estimates were calculated by using the frequency of leaks or through-wall cracks that have occurred which challenge the piping integrity. Further, conservative estimates were used for the probability of break given a leak (based on a technical review of information on fracture mechanics, data on high energy pipe failures and cracks, and assessment of pipe break frequencies estimated by others since WASH-1400). The pipe-break LOCA frequencies (per critical year) estimated from the experience are:

	Small LOCA	Medium LOCA	Large LOCA
PWR:	5E-4	4E-5	5E-6
BWR:	5E-4	4E-5	3E-5

No interfacing system loss-of-coolant accident (ISLOCA) events were identified in the U.S. operating experience.

Table 3-1. Frequency estimates of functional impact categories: mean, percentiles, and trends. (See text for detailed explanation.)

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles: 5th %ile	Percentiles: 95th %ile	Model Used: Trend	Model Used: Plant Difference[j]
Loss-of-Coolant Accident (LOCA)	G						
Large Pipe Break LOCA: PWR	G7	0	5E-6[d]	1E-7	1E-5	Constant[e]	No
Large Pipe Break LOCA: BWR	G7	0	3E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: PWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: BWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Small Pipe Break LOCA	G3	0	5E-4[d]	1E-4	1E-3	Constant[e]	No
Very Small LOCA/Leak	G1	4	6.2E-3	2.3E-3	1.2E-2	Constant[e]	No
Stuck Open: Pressurizer PORV	G4	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: PWR	G2	2	5.0E-3	1.2E-3	1.1E-2	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: BWR	G2	10	4.6E-2	2.5E-2	7.1E-2	Constant[e]	No
Stuck Open: 2 or More Safety/Relief Valves	G5	0	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Reactor Coolant Pump Seal LOCA: PWR	G8	2[d]	2.5E-3[d]	5.6E-4	5.4E-3	Constant[e]	No
Steam Generator Tube Rupture: PWR	F1	3	7.0E-3	2.2E-3	1.4E-2	Constant[e]	No
Loss of Offsite Power	B1	33	4.6E-2	8.2E-3	1.1E-1	Constant[e]	No
Total Loss of Condenser Heat Sink (combined):[f] PWR	L	75[f]	1.2E-1[c,f]	2.3E-2[i]	3.2E-1[i]	Decrease[f]	Yes[j]
Total Loss of Condenser Heat Sink (combined):[f] BWR	L	122[f]	2.9E-1[c,f]	2.0E-1	3.9E-1	Decrease[f]	No
Inadvertent Closure of All MSIVs: PWR	L1	35	3.8E-2[c]	1.9E-2	6.5E-2	Decrease	No
Inadvertent Closure of All MSIVs: BWR	L1	74	1.7E-1[c]	6.0E-2[i]	3.6E-1[i]	Decrease	Yes[j]
Loss of Condenser Vacuum: PWR	L2	35	6.9E-2	2.9E-5	3.0E-1	Constant[e]	Yes[j]
Loss of Condenser Vacuum: BWR	L2	46	2.0E-1	4.3E-2	4.6E-1	Constant[e]	No
Turbine Bypass Unavailable	L3	10	4.1E-3[c]	6.1E-4	1.2E-2	Decrease	No
Total Loss of Feedwater Flow	P1	159	8.5E-2[c]	1.3E-2[i]	2.5E-1[i]	Decrease	Yes[j]
General Transients (combined):[f] PWR	Q	1184[f,g]	1.2E+0[c,f]	6.1E-1[i]	2.1E+0[i]	Decrease[f]	Yes[j]
General Transients (combined):[f] BWR	Q	541[f,g]	1.5E+0[c,f]	8.5E-1[i]	2.5E+0[i]	Decrease[f]	Yes[j]
High Energy Line Steam Breaks/Leaks (combined)[h]	K	9[h]	1.3E-2	7.0E-3	2.1E-2	Constant[e]	No
Steam Line Break/Leak Outside Containment	K1	7	1.0E-2	5.0E-3	1.7E-2	Constant[e]	No
Steam Line Break/Leak Inside Containment: PWR	K3	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Feedwater Line Break/Leak	K2	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles: 5th %ile	Percentiles: 95th %ile	Model Used: Trend	Model Used: Plant Difference[j]
Loss of Safety-Related Bus	C						
Loss of Vital Medium Voltage ac Bus	C1	13	1.9E-2	1.1E-2	2.8E-2	Constant[e]	No
Loss of Vital Low Voltage ac Bus	C2	3	4.8E-3	1.5E-3	9.7E-3	Constant[e]	No
Loss of Vital dc Bus	C3	1	2.1E-3	2.4E-4	5.4E-3	Constant[e]	No
Loss of Safety-Related Cooling Water	E						
Total Loss of Service Water	E1	1[d]	9.7E-4[d]	1.1E-4	2.5E-3	Constant[e]	No
Partial Loss of Service Water	E2	6	8.9E-3	4.0E-3	1.5E-2	Constant[e]	No
Loss of Instrument or Control Air: PWR	D1	15[c]	9.6E-3[c]	3.9E-3	1.9E-2	Decrease	No
Loss of Instrument or Control Air: BWR	D1	21[c]	2.9E-2[c]	1.3E-2	5.5E-2	Decrease	No
Fire	H1	39	3.2E-2[c]	1.7E-2	5.2E-2	Decrease	No
Flood	J1	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No
		Total — PWR	1.4E+0[c]	6.9E-1[i]	2.4E+0[i]	Decrease[f]	Yes[j]
		Total — BWR	1.8E+0[c]	9.5E-1[i]	2.9E+0[i]	Decrease[f]	Yes[j]

a. Reactor trip events from 1987 through 1995, inclusive, except when noted for certain rare events.

b. Frequencies are presented in per critical year (8,760 critical hours per critical year).

c. For categories with a decreasing trend, the frequencies reported are based on the endpoint of the trend line (i.e., 1995, the last year of the study). The Medium and Large Pipe Break LOCA estimates were based on review of current literature and fracture mechanic analyses

d. No failures were identified in the 1987–1995 operating experience. Frequency estimates for Small Pipe Break LOCA, Reactor Coolant Pump Seal LOCA, Stuck Open: 2 or
and using world-wide experience. (Appendix J contains the results of the LOCA analysis.)
More Safety/Relief Valves, and Total Loss of Service Water categories were based on total U.S. operating experience (1969–1997).

e. Any evidence for a trend was weak, not statistically significant. The trend, if any, is too small to be seen in the data. Therefore, no trend is modeled.

f. Combined number of occurrences of all categories for each plant type (BWR, PWR) under this heading was used to calculate this frequency and trend.

g. Total number of initial plant-fault occurrences for this plant type.

h. The frequency was based on the combined number of occurrences in the categories under this heading.

i. The interval includes variability from plants with events early in life (for example, learning periods) and are wider than the plants' current performance. See Appendix G for results with the early-in-life events excluded.

... between-plant variation was evaluated with the first four months from date of commercial operation (early-in-life