DEC 14 1982

MEMORANDUM FOR: The Atomic Safety & Licensing Boards for:

Callaway Plant, Unit 1
Comanche Peak Steam Electric Station, Units 1/2
Midland Power Station, Units 1/2
Palo Verde Nuclear Generating Station, Units 1/2/3
South Texas Project 1/2
Waterford Steam Electric Station, Unit 3

The Atomic Safety & Licensing Appeal Boards for:

Comanche Peak Steam Electric Station, Units 1/2
Diablo Canyon Nuclear Power Plant, Units 1/2 50-275
Offshore Power Systems, FNP 1-8
San Onofre Nuclear Generating Station, Units 2/3
Virgil C. Summer Nuclear Station, Unit No. 1
Waterford Steam Electric Station, Unit 3

FROM: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing, NRR

SUBJECT: BOARD NOTIFICATION - ACRS EVALUATION OF PWR FLOW BLOCKAGE (Board Notification No. 82-125, 82-125A)

In accordance with present NRC procedures regarding Board Notifications, the enclosed information is being provided for your information as constituting new information relevant and material to safety issues. This information is applicable to all PWR's.

The notification relates to an evaluation concerning flow blockage during natural circulation which was performed by H. Etherington. Our assessment has concluded we are in general agreement with all of the points identified in Mr. Etherington's evaluation, and that all of his concerns regarding the phenomena of natural circulation flow blockage have been previously identified by the staff and provided to the boards in Board Notification BN-82-71. However, due to the interest in natural circulation and feed and bleed cooling in recent licensing proceedings, we believe it is in the best interest of the regulatory process to make the licensing boards aware of this recent evaluation. We do not believe that these results adversely impact our present staff position regarding reliance on natural circulation or the validity of feed and bleed cooling as a defense in depth measure.

Contact:
Suzanne Black, NRR
xt. 29788

OFFICE ▶							
SURNAME ▶							
DATE ▶							

DEC 1 4 1982

The staff is continuing to pursue with the B&W Owners the requirement for them to provide acceptable integral system experimental test data to aid in code verification and emergency operator procedure evaluation as part of TMI-2 action items II.K.3.30 and I.C.1 respectively. We will inform the boards of significant information if it causes us to change our technical position.

Original signed by:
Thomas M. Novak
Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: Licensee/Boards
Service List

OFFICE	DL:L	DL:L					
SURNAME	SBlack:ms	TMNovak					
DATE	12/10/82	12/10/82					

DISTRIBUTION LIST FOR BOARD NOTIFICATION

Before the Atomic Safety and Licensing Appeal Board

Diablo Canyon Units 1&2, Docket Nos. 50-275/323 OL
San Onofre 2&3, Docket Nos. 50-361/362
Waterford Unit 3, Docket No. 50-382

Ms. Elizabeth Apfelberg
E. Blake, Esq.
Mr. Richard E. Blankenburg
Mr. Glenn O. Bright
Mr. Herbert H. Brown
Dr. John H. Buck
A. S. Carstens
Mr. Brian Cassidy
Mr. Gary D. Cotton
Philip A. Crane, Jr., Esq.
Stephen F. Eilperin, Esq.
Mr. Frederick Eissler
David S. Fleischaker, Esq.
Mrs. Raye Fleming
Luke B. Fontana, Esq.
Dr. Harry Foreman
Phyllis M. Gallagher, Esq.
Arthur C. Gehr, Esq.
Mr. Byron S. Georgiou
Dr. Reginald L. Gotchy
Mr. Mark Gottlieb
Mr. Gary L. Groesch
William J. Guste, Jr., Esq.
Dr. Cadet H. Hand, Jr.
Mrs. Lyn Harris Hicks
Mr. Richard B. Hubbard
Mrs. Elizabeth B. Johnson
Dr. W. Reed Johnson
Dr. Walter H. Jordan
James L. Kelley, Esq.
Janice E. Kerr, Esq.
Dr. Jerry Kline
Charles R. Kocher, Esq.
Christine N. Kohl, Esq.
Ian Douglas Lindsey, Esq.
Mr. John Marrs
Charles E. McClung, Jr., Esq.
Thomas S. Moore, Esq.
Bruce Norton, Esq.
David R. Pigott, Esq.
Joel R. Reynolds, Esq.

Mr. James O. Schuyler
Mr. Gordon Silver
Malcolm Stevenson, Esq.
Paul C. Valentine, Esq.
Alan R. Watts, Esq.
Richard J. Wharton, Esq.
Harry M. Willis
John F. Wolf, Esq
Sheldon J. Wolfe, Esq.

Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Panel
Docketing and Service Section
Document Management Branch

ACRS Members

Dr. Robert C. Axtmann
Mr. Myer Bender
Dr. Max W. Carbon
Mr. Jesse C. Ebersole
Mr. Harold Etherington
Dr. William Kerr
Dr. Harold W. Lewis
Dr. J. Carson Mark
Mr. William M. Mathis
Dr. Dade W. Moeller
Dr. David Okrent
Dr. Milton S. Plesset
Mr. Jeremiah J. Ray
Dr. Paul C. Shewmon
Dr. Chester P. Siess
Mr. David A. Ward

DISTRIBUTION LIST FOR BOARD NOTIFICATION

Before the Atomic Safety and Licensing Board

Palo Verde Units 1,2&3, Docket Nos. 50-528/529/530
South Texas Project, Units 1&2, Docket Nos. 50-498/499
Waterford Unit 3, Docket No. 50-382

Charles Bechhoefer, Esq.
Lynne Bernabei, Esq.
Brian Berwick, Esq.
E. Blake, Esq.
Mrs. Peggy Buchorn
Dr. Dixon Callihan
Mr. Brian Cassidy
Dr. Richard F. Cole
Kim Eastman
Luke B. Fontana, Esq.
Dr. Harry Foreman
Arthur C. Gehr, Esq.
Mr. Rand L. Greenfield
Mr. Gray L. Groesch
William J. Guste, Jr., Esq.
Mr. Ernest E. Hill
Ms. Lee Hourihan
Dr. Walter H. Jordan
William S. Jordan, III, Esq.
Dr. James C. Lamb, III
Robert M. Lazo, Esq.
Ian Douglas Lindsey, Esq.
Jack R. Newman, Esq.
Mr. David Prestemon
Melbert Schwarz, Jr., Esq.
Mr. Lanny Sinkin
Malcolm Stevenson, Esq.
Sheldon J. Wolfe, Esq.

Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Panel
Docketing and Service Section
Document Management Branch

ACRS Members

Dr. Robert C. Axtmann
Mr. Myer Bender
Dr. Max W. Carbon
Mr. Jesse C. Ebersole
Mr. Harold Etherington
Dr. William Kerr
Dr. Harold W. Lewis
Dr. J. Carson Mark
Mr. William M. Mathis
Dr. Dade W. Moeller
Dr. David Okrent
Dr. Milton S. Plesset
Mr. Jeremiah J. Ray
Dr. Paul C. Shewmon
Dr. Chester P. Siess
Mr. David A. Ward

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

DEC 0 3 1982

22-(2)

MEMORANDUM FOR: Darrell Eisenhut, Director
Division of Licensing

FROM: Roger J. Mattson, Director
Division of Systems Integration

SUBJECT: BOARD NOTIFICATION CONCERNING A RECENT ACRS EVALUATION OF PWR FLOW BLOCKAGE

References: 1. TMI-1 Restart Appeal Board Notification, BN-82-71, containing letter from H. Denton, NRC, to H. Myers, congressional staff, "Dynamic Response of B&W Reactors to Small Break LOCAs.

2. Safety Evaluation Report, related to the operation of Midland Plant Units 1 and 2, NUREG-0793, Section 5.5, "Design Sensitivity of B&W Reactors", May 1982.

SUMMARY:
The purpose of this memorandum is to request that you inform all PWR Licensing and Appeal Boards of an evaluation by ACRS member H. Etherington titled "Flow Blockage by Steam During Natural Circulation in PWRs" and provided as enclosure (1). The Etherington evaluation discusses various mechanisms by which single phase natural circulation might be lost and regained. The feed and bleed mode of decay heat removal and the effect of high point vents in the B&W design on restoration of natural circulation are also discussed. The evaluation is primarily for plants with once through steam generators (B&W design), although some of the discussion relates to plants with inverted U-tube steam generators (Westinghouse and C. E. designs). The evaluation concludes that "the Committee (ACRS) may want to review the final disposition of this problem, and to be assured that the various possibilities (of core cooling) are reflected in sufficiently flexible and understandable operating procedures."

We recommend providing this information to the Boards due to recent interest in two phase natural circulation and the feed and bleed mode of cooling.

The staff is in general agreement with Mr. Etherington's evaluation. A similar evaluation was previously performed by the staff and documented in a letter which responded to questions from Dr. Henry Myers, Science Advisor to the House Committee on Interior and Insular Affairs. This letter is attached to Board Notification BN-82-71 (Ref. 1). In this letter the staff also expressed concerns relating to the understanding of plant response by operators in the event of natural circulation flow blockage, and has recommended that the phenomena be investigated by integral system tests.

The staff is pursuing resolution of the requirement for integral systems tests with the B&W Owners as part of TMI-2 Action Items II.K.3.30 and I.C.1. (see NUREG-0737). The status of this resolution is summarized in a letter recently sent to all licensees with B&W designed reactors. A copy of this letter is provided for the board's information as enclosure (2).

The staff has reviewed the Etherington evaluation and our assessment is discussed in some detail below. We request that our assessment be provided to the licensing boards concurrently with the Etherington evaluation (enclosure 1) and the letter to the B&W Owners (enclosure 2).

Background:
Recent licensing proceedings (in particular the TMI-1 Restart Hearing) have focused on the ability of PWRs to remove decay heat in various modes of natural circulation when feedwater is available and by feed and bleed in the event of loss of all feedwater. License applicants have not relied on feed and bleed cooling in meeting the Commission's regulations, but the staff and applicants recognize that such capability is available at many PWRs as a defense in depth for events beyond the design basis.

As such, feed and bleed cooling is addressed in present emergency procedures and is included in the emergency procedure guidelines now under development. Natural circulation, both in single phase and two-phase modes (including boiler-condenser), is the primary mechanism for decay heat removal when the reactor coolant pumps are not operational and feedwater is available. Reliance on natural circulation to remove decay heat from the reactor system, both with and without a small break LOCA, has always been considered acceptable to the staff. Single phase (liquid) natural circulation has been demonstrated extensively in operating reactors, and two phase natural circulation including the boiler condenser mode, has been justified by test for inverted U-tube steam generator plants. Two phase natural circulation, including the boiler-condenser mode, has been shown to be effective by analysis for all PWR reactor types. In addition, auxiliary feedwater systems are sufficiently reliable to provide the required heat sink for satisfactory comformance to the General Design Criteria.

Staff Comments:

I. The evaluation by Mr. Etherington deals primarily with the time required to condense a steam bubble which might be trapped at the top of the hot legs of a B&W designed reactor and therefore affect the period of time in which natural circulation, and hence decay heat removal, was interrupted. The evaluation does not address core cooling as a result of natural circulation interruption. The question of core cooling in such a situation was addressed by the staff in BN-82-71 (reference 1). In that reference the staff reported a similar evaluation of bubble condensation rates and concluded that the reactor core would be adequately covered and cooled regardless of the time required to condense the bubble and restore a liquid flow path between the vessel and the steam generators.

2. The Etherington evaluation postulates various heat transfer mechanisms for steam void condensation within the hot leg, assuming that the coolant loops are in a quiescent condition (little or no coolant flow). Bubble condensation times of between 3 and 65 hours are calculated, depending on which heat transfer mechanisms dominate the condensation process. The Etherington evaluation also makes note of a calculation performed by LANL using the TRAC computer code. The TRAC code predicted that the coolant loops would not be in a quiescent condition even in the presence of a steam bubble. Rather, it predicted an intermittent condition of slug flow causing rapid steam condensation. Using the RELAP-5 computer code the staff has also predicted slug flow in the coolant loops when steam voids were present.

   But the staff's conclusion on the safety of interrupted natural circulation does not rest on the TRAC or RELAP calculations. Rather the staff evaluated the consequences of both rapid and slow bubble condensation in Ref. 1. For the limiting assumption of an infinitely slow condensation rate (i.e., no condensation) the staff concluded that the reactor core would still remain covered with water and adequately cooled.

3. The staff does not believe that any current method of predicting steam void condensation rates has been adequately verified. The staff has concluded that additional data needs to be obtained using an integral system test facility scaled and geometically similar to the B&W reactor design. Appropriate test data has already been obtained for Westinghouse and CE designs at the LOFT and Semiscale facilities. The staff concluded in reference 1 that for B&W designs such data was needed for operator training and evaluation of emergency operation procedures but was not required to demonstrate the adequacy of core cooling.

4. In reference 1, the staff evaluated the consequences of steam voids trapped in the hot legs of a B&W reactor following a small break (i.e., stuck open PORV) which was subsequently isolated. The evaluation by Mr. Etherington postulates that voids might be formed by PORV or pressurizer spray actuation. We agree that pressurizer PORV or spray actuation*, when the primary system is at or near saturation conditions, is a mechanism by which voids might form and

*In this case, we assume this is the auxiliary pressurizer spray, which is not derived from the main reactor coolant pump flow. If this was normal pressurizer spray, which is derived from main reactor coolant pump flow, then this pump operation would also serve to sweep any steam voids into the steam generators where they would be condensed.

interrupt natural circulation. The staff also evaluated the effect of reactor system overcooling in producing void formation and the loss of natural circulation for B&W reactors in Ref. 2. This evaluation indicated that anticipated overcooling events should not result in the loss of natural circulation, and even the more severe steamline break events would only tend to block circulation in one loop.

5. The evaluation by Mr. Etherington states "It appears possible that there is no direct recovery to single-phase natural circulation from the boiler-condenser mode." The staff agrees with this statement in the sense that rapid void condensation predicted by computer codes has not been verified by integral system tests and, in fact, may not occur. However, recovery of single phase natural circulation is not required for successful mitigation of a LOCA as discussed below.

   Following a loss of coolant accident, the ECCS systems of PWRs are not designed to deliver enough water to the reactor system to completely refill it except for very small break sizes. When the system refills above the break elevation, the ECC water will spill out of the break and prevent the coolant level in the primary system from rising higher than the break elevation. However, because all primary system piping is at an elevation above the top of the core, the system will always refill to above the top of the core, thus assuring the core will be covered. By maintaining a water level above the top of the core, core cooling is assured by nucleate pool boiling heat transfer. This condition will maintain the maximum fuel cladding temperatures slightly above the coolant saturation temperature. Small break LOCA operator guidelines for B&W designed PWRs also state that it is not necessary to refill the reactor system following a LOCA in order to assure long-term core cooling.

6. The evaluation by Mr. Etherington states that a "one-inch vent line at the top of a U-bend could easily eliminate a steam void in a subcooled system as fast as makeup could be supplied. But venting a steam space in a saturated system without makeup could be an exercise in futility." We agree with these statements, but we note that the high point vents of PWRs are designed to vent hydrogen, not steam, in accordance with the requirements of 10CFR 50.44. They are designed to be small enough in diameter so that their failure will not produce a LOCA in accordance with Item II.B.1 of NUREG-0737. Most high point vent sizes are smaller than one inch i.d. If the high point vents were opened by the operator in an attempt to restore natural circulation while the primary system hot leg coolant was near to or at saturation conditions, the pressure in the vicinity of the open vent would decrease. This would cause some of the saturated liquid to flash to steam. The steam formed from flashing along with additional steam formed from boiling in the core, would replenish any steam removed from the hot leg U-bend by the vent. Opening of the hot leg high point vents would only aid in reestablishing natural circulation if opening the vent removes steam at a faster rate than it is generated and if the

volume occupied by the steam being vented was being replaced with liquid (i.e., the system was being refilled).

7. The staff agrees with Mr. Etherington's statement that natural circulation "Blockage by non-condensible gas remains as a low-probability occurrence". This statement is consistant with previous staff evaluations. (See NUREG-0565, NUREG-0611 and NUREG-0635.)

8. The evaluation by Mr. Etherington states that feed-and-bleed requires use of non-safety-grade components and is not an NRC requirement. We point out that at those plants which can feed and bleed with the safety valves, the safety valves are safety grade. In addition, at some plants, the PORVs do meet safety grade requirements. Thus, we believe a more appropriate statement would be "feed and bleed operation may rely on non-safety components.

Conclusions

Based on our assessment of Mr. Etherington's evaluation, we do not believe it contains any relevant material for new information per the criteria of Office Letter Number 19. Thus, we do not believe we are required to notify Licensing Boards of either Mr. Etherington's evaluation, or the staff's assessment of this evaluation. In fact, our assessment has concluded we are in general agreement with all of the points identified in Mr. Etheringtons evaluation, and that all of his concerns regarding the phenomena of natural circulation flow blockage have been previously identified by the staff and provided to the boards in Board Notification BN-82-71. However, due to the interest in natural circulation and feed and bleed cooling in recent licensing proceedings, we believe it is in the best interest of the regulatory process to make the licensing boards aware of this recent evaluation. We do not believe that these results adversely impact our present staff position regarding reliance on natural circulation or the validity of feed and bleed cooling as a defense in depth measure.

The staff is continuing to pursue with the B&W Owners the requirement for them to provide acceptable integral system experimental test data to aid in code verification and emergency operator procedure evaluation as part of TMI-2 action items II.K.3.30 and I.C.1 respectively.

Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration

Enclosures:
1. Memorandum from R. Fraley ACRS to H. Denton NRR and R. Minogue RES, Transmitting Etherington Evaluation, November 10, 1982.

2. Letter from H. Denton NRC to W. Parker, Duke Power Company, November 16, 1982.

cc: See Next Page

cc: H. Denton
S. Hanauer
R. Minogue, RES
O. Bassett, RES
R. Landry, RES
N. Lauben
W. Hodges
W. Lyon
M. Keane
G. Lainas
E. Case
G. Knighton
D. Ross, RES
H. Sullivan, RES
G. D. McPherson
T. Marsh
G. Mazetis
R. Barrett
T. Novak
W. Jensen
H. Etherington, ACRS
R. Fraley, ARCS
P. Boehnart, ACRS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 10, 1982

Enclosure 1

MEMORANDUM FOR: Harold Denton, Director, NRR

Robert Minogue, Director, RES

FROM: R. F. Fraley, Executive Director, ACRS

R F Fraley

SUBJECT: FLOW BLOCKAGE BY STEAM DURING NATURAL CIRCULATION IN BWRs

The attached is being made publicly available in accordance with a request from the Science Advisor, House Committee on Interior and Insular Affairs. Copies are being provided for your information and use.

Attachment:
Memo from H. Etherington, ACRS Member, to P. G. Shewmon, ACRS Chairman and ACRS Mbrs. dated 9/7/82, Subject: Flow Blockage By Steam During Natural Circulation in PWRs

cc:
D. G. Eisenhut, DL
R. H. Vollmer, DE
H. L. Thompson, DHFS
S. Hanauer, DST
T. Novak, OR
G. Lainas, SA
R. Mattson, DSI
T. P. Speis, RS
B. Sheron, RSB
C. Kelber, RES
H. Sullivan, RES
G. A. Arlotto, RES
O. E. Bassett, RES
K. R. Goller, RES
R. Bernero, RES
W. J. Dircks, EDO
V. Stello, EDO
J. Auston, OCM
E. Abbott, OCM
J. Milhoan, OCM
D. Garner, OCM
S. Chestnut, OCM

8212130293

1p

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 7, 1982

MEMORANDUM FOR: P. G. Shewmon, Chairman
ACRS Members

FROM: H. Etherington, ACRS Member

SUBJECT: FLOW BLOCKAGE BY STEAM DURING NATURAL CIRCULATION IN PWRs

At the June 1982 ACRS meeting, there was a brief discussion of this subject. Questions have been asked by the Union of Concerned Sceintists, Dr. Henry Myers, and the ASLAB for Rancho Seco. The purpose of this memorandum is to explore and quantify some fundamentals of the problem.

1. In the absence of a heat sink, steam cannot be condensed, in any amount, by repressurization. When steam (or any other vapor) is compressed, it becomes superheated. For example, the Mollier chart shows that isentropic compression of saturated steam from 1000 psia to 1500 psia results in superheat of about 40°F; irreversible adiabatic compression results in greater superheat.

2. Condensation of a steam pocket is not a simple reversal of the steam formation process, i.e., it should not be assumed that steam formed during a pressure transient can be quickly condensed by restoring the original pressure.

   Steam separates by gravity and accumulates at high spots in the system, but the steam may be a product of flashing over a substantial part of the liquid system. The reverse process, steam condensation, proceeds by heat transfer processes that have no relation to the mass separation process.

3. Simple classical modes of heat transmission are inadequate for rapid condensation of a large steam void. When the system is repressurized, the steam quickly loses its slight superheat by contact with the steel boundary and surface water. Thereafter, the steam and the water surface remain at the saturation temperature corresponding to the new pressure. As steam condenses by contact with water, the latent heat of condensation is transmitted downwards by the very slow process of conduction into stratified water -- the temperature gradient is in the wrong direction for convection.

8212020048
13pp

Other modes of heat transfer are also investigated in an illustrative calculation which shows that a layer of steam, three feet high, might be condensed in the following times, each mode being treated separately.

1. Conduction into water 65 hr.

2. Conduction along a low-alloy steel pipe 18 hr.

3. Conduction through low-alloy steel pipe from the steam space to just below the surface of the water 3 hr. (?)

4. Heat loss through pipe to atmosphere 15 hr. (?)

   (The last two items are based on unsupported hypotheses -- both are calculable, and item 3 might be worth developing.)

5. At the assumed conditions, steam is a significant heat radiator, and if much of the radiant heat from the steam or pipe could pass through the thin layer of heated water, it would be possible to condense about 1/2 ft/hr of steam by this mode. However, whereas water is transparent to radiation in the visible part of the spectrum, it is relatively opaque to low-temperature heat radiation. Also, any internal radiation "from hot water to cold water" is presumably included in the experimentally determined conductivity. It appears unlikely that radiation could contribute importantly to condensation.

4. <u>Other modes of heat transfer may predict greater rates of steam condensation</u>, but these would have to be justified either generically or on a case-by-case basis.

   It might, for example, be demonstrated that the system is not sufficiently quiescent to sustain a fully stratified thin layer of heated water at the surface; or that alternately raising and lowering the level, by varying the system pressure or by surges, will permit effective heat transfer by alternately heating and cooling the steel pipe.

   On the other hand, sustained interruption of circulation could lead to intrusion of hotter water and even more steam into the hot leg pipes.

5. <u>A high repressurization pressure is strongly favorable to steam condensation</u>.

   The driving force for all modes of heat transfer, except heat loss to atmosphere, is the temperature difference between the steam and the water, i.e., the difference between the saturation temperatures corresponding to the repressurized pressure and the depressurized pressure.

The numbers given in Section 3 are for depressurization to 980 psia and repressurization to 2000 psia. If the system were repressurized to 1200 psia, the times for modes 1, 2, 3, and 4 would be increased to 1180, 324, 3.6, and 17.5 hr., respectively.

In the event of a small-break loss-of-coolant accident (SBLOCA), the attainable repressurization pressure may be limited by conflicting procedural requirements, or inadequacy of HPI pump head or capacity. With no repressurization and no natural recirculation, heat transfer ceases by all modes discussed in Sections 3 and 4 except by heat-loss through the pipe to the sourroundings.

6. Feasibility of developing a large steam void. It is not the purpose of this memorandum to discuss how steam voids may form in a system during natural circulation, but it is pertinent to inquire whether the illustrative example is reasonable.

The calculation is based on depressurization to 980 psia of a system whose saturation pressure is 1000 psia (a negative temperature margin of 2.44°F). Such a pressure loss might be associated with an open valve in the pressurizer or actuation of a pressurizer spray. For the assumed conditions, a three-foot high steam layer forms at the high points of a loop -- more exactly, three cubic feet of steam in the U-bend region for each square foot of pipe cross section. For two 3 ft. diameter loops, the total volume of steam (at actual conditions) is 42.4 cu. ft. This quantity of steam is associated with expulsion of water to the pressurizer which causes a 12 in. increase in pressurizer level. Since the calibrated height of the pressurizer is 400 in., it appears that much larger steam voids could form without generating strong self-limiting tendencies.

7. An important one-step reduction in steam volume. Repressurization, if permissible, raises the temperature of the steam above that of the pipe or vessel, and the latter becomes a heat sink, causing fairly rapid partial condensation of steam. (This rapid one-step partial condensation is distinct from the slow continuing condensation described in Section 3).

The fraction of steam condensed depends on the initial water temperature, the final pressure, and the rate of repressurization. An illustrative calculation, improbably favorable in these respects, shows that 45% of the steam could be condensed by this mechanism.

8. Steam voids can be quickly dispersed by forced circulation. Problems arise only when conditions or procedures require shutdown of the reactor coolant pumps. Natural circulation is assumed in following discussions unless otherwise stated.

9. Blockage of an inverted U-bend ("candy cane") in a B&W system.

   (1) If the void does not completely block flow, it will not stop natural circulation.

   (2) If the void completely blocks flow, the heat sink (steam generator) is isolated and the temperature of a subcooled system rises until saturation is reached; cooling then proceeds in the boiler-condenser mode.

   The boiler-condenser mode requires that the steam void extend over several feet of the riser pipe, over the entire U-bend and the upper plenum ("channel") of the steam generator, and down into the steam generator tubes far enough to provide sufficient heat transfer surface to condense the steam. Complete blockage implies that the water level in the riser pipe, allowing for static and dynamic effects of steam bubbles, is low enough to prevent two-phase flow or slug flow over the bend.

10. It appears possible that there is no direct recovery to single-phase natural circulation from the boiler-condenser mode. Replenishment of inventory by make-up pumps will compress the steam void, raising the level of water in the tubes, and probably into the plenum, thereby decreasing the heat transfer surface or isolating the heat sink. There will be some steam condensation by mechanisms described in Sections 3 and 7, but the temperature of the water will slowly rise until it reaches saturation at the increased pressure, boiling will start again and lower the level of the water in the steam generator plenum and tubes until the boiler-condenser mode is re-established at the new pressure. It appears that the recirculation pumps must be started for re-establishment of single-phase recirculation. (The basis for the B&W admonition to "bump the pump"?)

    Note: A LANL draft report "Small-Break LOCA Recovery in B&W Plants" was distributed with a memorandum dated July 19, 1982, T. M. Novak to ASLAB for Rancho Seco. This report is based on a TRAC analysis and concludes that natural circulation can be reestablished by restoring the inventory. The analysis shows an intermediate condition of slug flow, apparently associated with oscillation of water level. I don't know whether this condition can be demonstrated and quantified in a real system.

11. Non-condensible gas in a void strongly inhibits all modes of steam condensation.

(1) There is no effective mechanism for absorption of a non-condensible gas in a non-flowing sytem. Most of the gas originates, like steam, from regions remote from the surface. Return of gas to the water proceeds by slow mass transfer analogous to heat transfer by conduction, but in the case of a gas, there are no alternative faster modes of mass transfer.

(2) The presence of gas reduces the partial pressure of the steam, and therefore the saturation temperature which provides the driving force for steam condensation. When the saturation temperature of the steam is reduced to that of the water, the water and steel are no longer available as heat sinks.

(3) Hydrogen, in normal concentrations, is not likely to cause a problem. At 1000 psi, the volume of added hydrogen in the system is only 2 to 3 cubic feet. This is small compared with the volume of a steam void that could cause trouble.

(4) Hydrogen from a metal-water reaction or nitrogen from core flooding accumulators could lead to large quantities of non-condensible gases in the steam voids.

12. Water must be supplied to fill a void! If water is not supplied, the system cannot be repressurized except by objectionable increase of bulk water temperature and additional boiling.

Water may be supplied by the pressurizer, by transfer from another voided region (the vessel head), by makeup pumps, or by starting the circulating pumps to disperse the void throughout the system. In a very small break LOCA, these processes may suffice to eliminate a void, possibly becoming effective only after partial depressurization. With a slightly larger break, voids may persist at least until the low pressure emergency cooling system can function.

13. A one-inch vent line at the top of a U-bend could easily eliminate a steam void in a subcooled system as fast as makeup could be supplied.

    But venting a steam space in a saturated system without makeup could be an exercise in futility.

14. The main concern is a B&W System with a SBLOCA that is too small to depressurize the system sufficiently for early operation of the low pressure safety systems, yet too large to permit the inventory to be maintained by the high pressure pumps. Loss of inventory leads to formation of steam voids at high spots, and the possibility of "degraded" modes[1] of heat transfer as discussed in Sections 9 and 10.

15. Other Possible Concerns. Other conditions that could cause concern are possible but not likely.

    (1) Steam blockage of the U-bends in a B&W system as a result of an operating transient is conceivable. In this case the reactor coolant pumps (RCP) would probably be available to disperse the steam. If the RCPs were not available, it might be possible by repressurization to reduce the steam volume sufficiently to permit passage of water; or it might be possible to depressurize the system and sufficiently reduce the inventory to permit cooling in the boiler-condenser mode. If these procedures cannot be relied on, it may be necessary to review the adequacy of heat transfer modes discussed early in this memorandum.

    (2) In U-tube steam generators (W and CE), the level of the secondary system water is normally above the U-bend, and a steam pocket could not form so long as the temperature of the secondary system is below that of the primary system. Departure from the normal condition could lead to conditions similar to those described for a B&W system.

    (3) Blockage by non-condensible gas remains as a low-probability occurrence.

[1] The NRC Staff considers single-phase natural circulation and boiler-condenser heat transfer both acceptable.

16. Feed-and-bleed offers an alternative mode of heat removal. This requires use of non-safety-grade components and is not an NRC requirement. Licensing Boards, however, appear to give some weight to this capability.

17. Conclusion. The Committee may want to review the final disposition of this problem, and to be assured that the various possibilities are reflected in sufficiently flexible and understandable operating procedures.

ATTACHMENT A

ILLUSTRATIVE CALCULATION

Initial hot-leg water temperature: 544.6°F (corresponding to saturation at 1000 psia), and at some satisfactory overpressure.

Depressurization, caused by pressurizer malfunction (e.g., open PORV or spray actuation) to 980 psia at high point in the system (saturation temperature 542.2°F).

Calculated void fraction (cu. ft. of steam, at actual temperature and pressure, per cu. ft. of water): 0.097 at high point, decreasing linearly to zero 62 ft. below the final surface of the water.

Water expelled to pressurizer: 87% of steam volume.

Steam formation and condensation in pipe over 62 ft. high.[1]

Height of steam void	3.0 ft.
Associated heat of condensation per sq. ft. of water surface	3690 Btu/ft^2
Difference between saturation temperatures	93.6°F
Condensation time by conduction to water	65 hr.
Condensation time by conduction to carbon steel:	18 hr.

---

1 The quantity of steam formed is greater if the pipe extends less than 62 ft. above the vessel outlet, because boiling then also occurs in the much larger volume of the reactor vessel; but much of the extra steam will collect in the vessel head.

Effect of pressure.

Difference between saturation temperatures, t

2000 psia/980 psia	93.6°F
1200 psia/980 psia	25.0°F

Heat of condensation of steam

2000 psia	561 Btu/lb.
1200 psia	641 Btu/lb.

Condensation time by conduction to water or steel is greater at 1200 psia by the factor:

$$(641/561)^2(93.6/25.0)^2 = 18.3$$

Heat capacity of water. If a layer of water could be heated uniformly from the depressurized temperature to the repressurized temperature, condensation of 3 ft. of steam would heat a layer 0.72 ft. thick at 2000 psia, 2.9 ft. at 1200 psia, and 20 ft. at 1000 psia.

ATTACHMENT B

CALCULATION IN SUPPORT OF MEMORANDUM

Steam Table Data

		spec. vol, v		enthalpy		density, 1/v	
psia	°F	liq	vap	liq	vap	liq	vap
980	542.17	.0215	.4557	539.3	1192.6	46.51	2.194
1000	544.61	.0216	.4456	542.4	1191.8	46.30	2.244
			Average	540.85		46.405	

α = steam fraction formed by volume
f = volume fraction of water expelled at average density and enthalpy

Material Balance per cu. ft. of initial liquid:

$$(1-\alpha)46.51 + \alpha 2.194 + f46.405 = 46.30$$

Heat Balance per cu.ft.

$$(1-\alpha)46.51 \times 539.3 + \alpha 2.194 \times 1192.6 + f46.405 \times 540.85 = 46.30 \times 542.4$$

$f = 0.086$ $\qquad$ $\alpha = 0.097$

Rough Check

$$46.3(542.4 - 539.3) = \alpha(1192.6 - 539.3) \times 2.194$$

$$\alpha = 0.100$$

Depth corresponding to 20 psi

$$d = 20 \times 144/46.51 = 62 \text{ ft.}$$

Height of steam void

$$62 \times 1/2 \times 0.097 = 3.0 \text{ ft.}$$

Condensation Times
(calculation based on 1 sq ft of water surface)

1 Pressure, psi	980	1000	1200	2000
2. Saturation temperature t_s, °F	542.17	544.61	567.2	635.8
3. Water enthalpy, saturated, Btu/lb	539.3	542.4	571.7	671.7
4. Steam " , superheated, "		1194	1213	1233
5. Density of water, lb/ft³	46.57	46.30	44.84	38.41
6. " " steam, "	2.194	2.244	2.763	5.325
7. Heat of condensation, Btu/lb, (#4 − #3)		652	641	561
8. Enthalpy increase of water, from sat. at 980 psi to t_s, (#3 − 539.3)		3.9	32.4	132.4
9. Heat of condensation per ft of steam height referred to 980 psi sat., (#7 × 2.194)		1430	1406	1231
10. Depth of water, hypothetically heated to saturation, to condense 1 ft of steam, (#9 ÷ #5 ÷ #8)		7.92	0.968	0.239
11. Time T to condense 1 ft of steam by conduction into water (see below), hr		14200	331	7.97
12. Ditto by conduction into steel pipe		3920	36	1.98

Time (#11 and #2) varies as square of steam heigh

Heat Conduction Properties

	Cond. k	Sp. Ht c_p	Density w	Diffus. $a = k/c_p w$	$\alpha = k/\sqrt{a\pi}$	
Water, 550°F (sat.)	.3246	1.21	45.9	.00540	2.49	Use 2.46
" 600°F "	.2419	1.57	42.4	.00456	2.44	
Low-alloy Steel, 575°F	25	0.14	490	.364	23.4	

Steel Cross-Section per Sq Ft of Water Surface, Ratio R

Design pressure 2500 psi, design stress 25,000 psi, thickness b, radius r

$b = 2500r/25000 = 0.1r$

$R = 2\pi r b/\pi r^2 = \underline{0.2}$

For semi-infinite medium at uniform initial temp t_0 raised to constant surface temp. t_1

(* - Estimate, detailed steam tables not available) HE telecon 11/8/82

B3

Heat flux per sq ft of water surface at time T:

$$h = k(t_1 - t_0)/\sqrt{\pi a T} = \alpha(t_1 - t_0)/\sqrt{T}$$

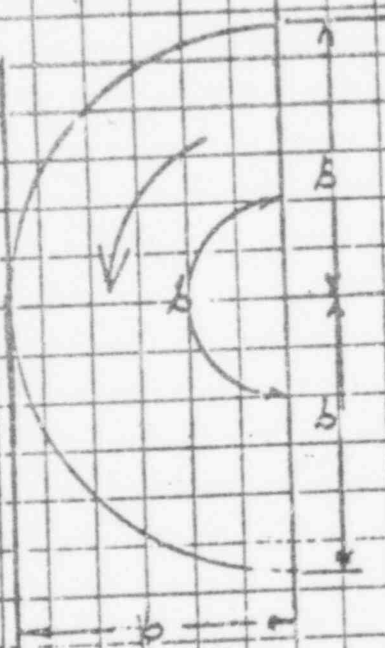
Total heat flow per sq ft to time T:

$$H = \int_0^T h\,dT = 2\alpha(t_1 - t_0)\sqrt{T}$$

$$T_w = H^2/[2\alpha(t_1 - t_0)]^2 = (\#9)^2/[4.92(\#2 - 542.17)]^2$$

For steel: $T_s = H^2/[0.5\alpha_s(t_1 - t_0)]^2 = 0.276\,T_w$ ✓

Hypothetical rate of heat flow from steam to water through steel



2000 psia
$t_1 = 635.8$ °F
Ht trans. coeff
$h_s = \infty$

Steam

Water

$t_i = \frac{1}{2}(635.8 + 542.2)$
$= 589.0$°F
$h_w = 500$

$t = 542.2$ °F

Pipe wall thickness

For pipe radius 1.5 ft, $b = 0.15\frac{?}{3}$ ft

For mean path length b, heat flow per ft of pipe circumference

$$= \frac{t_1 - t_0}{b/kb + 1/500b}$$

$$= 46.8/(0.04 + 0.013)$$

$$= 883 \text{ Btu/hr}$$

Heat flow per sq ft of water surface

$$= 883 \times \frac{2\pi r}{\pi r^2}$$

$$= 1177 \text{ Btu/ft}^2\text{-hr}$$

Time to condense 3 ft of steam

$$= 1231 \times 3/1177 = 3.1 \text{ hr}$$

Steam Condensation by Heat Capacity of Pipe

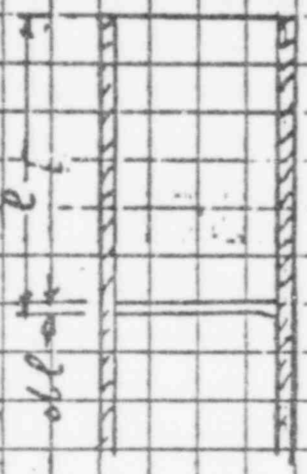
Based on 1 sq ft of water surface

t = steam saturation temperature

C = heat capacity of pipe per ft of length per °F

$= 0.2 \times 490 \times 0.141 = 13.8$ Btu/ft-°F

w = density of steam, lb/ft³



Assume very slow repressurization, no axial heat flow

Weight of steam $\quad W(t) = w\ell$

Steam condensed $\quad dW = w\,d\ell + \ell\,dw$

Heat balance:

$$(w\,d\ell + \ell\,dw)\,h_{ev} + C\ell\,dt = 0$$

$$-\frac{d\ell}{\ell} = \frac{C}{w h_{ev}}\,dt - \frac{dw}{w}$$

More directly:

$$h_{ev}\,d(w\ell) + C\ell\,dt = 0$$

$$\frac{d(w\ell)}{w\ell} = -\frac{C}{h_{ev} w}\,dt$$

$$\ln(\ell w) = -\int \frac{C}{w h_{ev}}\,dt$$

$$\frac{\ell_2 w_2}{\ell_1 w_1} = \exp\left[\int_{t_1}^{t_2} \frac{C}{w h_{ev}}\,dt\right]$$

Fraction of steam condensed $= \dfrac{\ell_1 w_1 - \ell_2 w_2}{\ell_1 w_1}$

$$= 1 - \exp\left[-C\int_{t_1}^{t_2} \frac{1}{w h_{ev}}\right] dt$$

Integrate by Simpson's Rule

psia	t	h_{ev}	$v = 1/w$	$1/wh_{ev} = v/h_{ev}$	
980	542.2	653.3	0.4557		.000698
(1120.5)	589.0	682	0.2959	.000434	.001735
2000	635.8	561	0.1878		.000335
				S =	.002768

$$\int = \frac{1}{6} \times .002768\,(635.8 - 542.2) = 0.04325$$

Fraction of steam condensed $= 1 - e^{-13.8 \times 0.04325} = 0.44$

HE 8-1-82

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

NOV 16 1982

Enclosure 2

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Sir:

We have recently received two separate letters from each of the licensees with B&W reactors regarding actions and resolution plans for integral systems testing and TMI Action Plan Item II.K.3.30.

The staff has reviewed these letters and has met on two separate occasions with the B&W owner representatives in the recently formed Test Advisory Group. Based on these efforts, we are now in a position to advise you as to the extent we find your proposals acceptable and what further actions need to be taken in order to resolve TMI Action Plan Item II.K.3.30. In addition, we offer our general comments on your perception of the history of this matter as documented in your letters.

General Comments

From the letters we have received, we understand it is your belief that the staff has expanded the issues which you believed you were to address. The original staff concern, as stated in our memorandum to William Parker of Duke Power Company on April 1, 1981, "... was with regard to the need for overall model verification against integral system experimental data, ..." Specifically, we stated "... that integral system two phase natural circulation test data (i.e., representation of small break conditions) applicable to the B&W primary system design would be required for overall model verification." As we have progressed in trying to understand the two-phase performance of the B&W reactor system under transient and accident conditions, we have identified other issues (e.g., steam generator tube rupture response) related to the same original concerns with small break LOCA phenomena. However, we do not consider that the general issue as originally identified has been expanded. We recognized the desire by the B&W owners to try to "narrow the scope" and reduce the concern to one of "bubble dynamics." We have never agreed that the subject of concern was simply "bubble dynamics."

Secondly, you indicate that you provided many reports to the staff which were in support of resolution of the issue. We point out that most of

8211230137 XA

these reports did not address the subject of concern, namely verification of the two phase performance of the B&W reactor system. Rather, they addressed such issues as leak discharge modelling, surge line modelling, auxiliary feedwater penetration and axial flow distribution and core heat transfer model comparison to ORNL data. While these topics are of definite interest to the staff, and indeed address some of the original nine areas identified by the staff as needing additional justification, they were of little use regarding the primary subject of concern.

Finally, your letters imply that you believe that you have provided sufficient information on the additional areas of concern (not related to integral system testing needs) originally identified by the staff as part of II.K.3.30. This belief was also expressed by you at the September 28, 1982 meeting of the ACRS subcommittee on ECCS. We do not agree with your conclusion. In your recent letters you have only provided a schedule for the formal submittal of this additional information. While some reports were informally transmitted to us by B&W in a letter dated September 30, 1982, we still do not have a formal submittal. Until this information is submitted, reviewed, and found acceptable, this aspect of the II.K.3.30 requirements cannot be resolved. We recommend you provide a more accurate status of II.K.3.30 information submittals in future documents and presentations.

Further Actions Needed to Resolve II.K.3.30

As the staff has previously told you, it is our desire to treat the need for integral systems test data as long-term confirmatory research, not directly related to present licensing issues*. However, we believe our closing of the present outstanding licensing issues must be predicated on the expectation that the technical judgments we make today will be tested by a longer-term confirmatory research program. This approach is not unlike that previously used by the NRC to justify the acceptability of Appendix K to 10 CFR Part 50, pending completion of longer-term confirmatory research (e.g., LOFT and FLECHT).

Thus, in order to close on the current licensing issues, we require that acceptable progress be made in determining the need for a longer-term research program beyond that presently proposed by the B&W owners. We need to see that serious study is being made of the costs and benefits of the various facility options for obtaining the data. As you know, a Test Advisory Group (TAG) has been formed to fulfill this need.

---

*Present licensing issues include

1. II.K.3.30
2. Midland SER
3. Vessel head vent exemption request
4. ATOG
5. TMI-1 RESTART

At present we understand the B&W owners' proposal is that the decision to define additional testing needs should wait until the completion of the planned GERDA and SRI-II test programs. The basis for this proposal is that there is uncertainty as to whether the GERDA facility will perform as expected and that no safety issues are involved, and thus no urgency is indicated.

While we agree that no immediate safety issues are involved, we believe that there are numerous indirect safety issues which warrant a more rapid decision to proceed with additional integral systems tests. Moreover, since you propose delaying any decision on future testing until the usefulness of the GERDA data is established, it is not clear to us how we could approve II.K.3.30 or resolve other licensing issues prior to determining if the GERDA data is acceptable. Therefore, we do not accept the plan you have proposed.

Finally, the GERDA facility lacks active pumps and is not expected to adequately address steam generator tube ruptures, and other asymmetric effects involving two loops. The SRI-II facility lacks the elevation scaling that is important in gravity dominated phenomena. For these reasons, we have concluded that the GERDA/SRI-II testing program will most likely not satisfy the confirmatory research needs for the B&W design.

This conclusion is shared by the ACRS as shown in its letter to NRC Mr. William J. Dircks dated October 13, 1982 (attached).

We reiterate our earlier position that progress by the TAG in developing recommendations to senior management in utilities, EPRI and NRC on how to best meet these longer term confirmatory testing needs is necessary before II.K.3.30 and other current licensing issues can be resolved separate from the longer term, integral systems test. Thus it is our conclusion that the owners must address these issues expeditiously in order to avoid delays in resolving present licensing issues.

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

William T. Orders
Senior Resident Inspector
U.S. Nuclear Regulatory Commissio..
Route 2, Box 610
Seneca, South Carolina 29678

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

J. Michael McGarry, III, Esq.
DeBevoise & Liberman
1200 17th Street, N.W.
Washington, D. C. 20036

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 13, 1982

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS COMMENTS ON NRC PROGRAM TO ADDRESS CONCERNS WITH THERMAL HYDRAULIC BEHAVIOR OF BABCOCK AND WILCOX PLANTS DURING TRANSIENTS AND ACCIDENTS

During its 270th Meeting, October 7-8, 1982, the Advisory Committee on Reactor Safeguards met with the NRC Staff and representatives of the Babcock and Wilcox (B&W) Owners Group to discuss NRC Staff concerns regarding the dynamic thermal hydraulic behavior of B&W plants during transients and accidents, particularly small break loss of coolant accidents.

For some time, the NRC Staff has identified a need for experimental data for investigation of specific plant phenomena and for assessment of analytical calculations of B&W plant response to transients and accidents. Recently, a Test Advisory Group composed of NRC Staff members and representatives of the B&W Owners Group was formed to evaluate alternatives available for obtaining the desired test data. The Owners Group has proposed use of two industry test facilities (GERDA and SRI-II) in response to the NRC Staff's concerns.

While we support the cooperative effort between NRC and the Owners Group, it appears that the GERDA and SRI-II facilities as now proposed will be inadequate to satisfactorily address the NRC Staff concerns in this matter. Although the data obtained from these facilities may be useful, we believe that a more adequate facility, similar to the proposed Semiscale MOD-5 configuration, is necessary to address the major operational questions of concern. We also wish to emphasize that the timely acquisition of such data and associated analyses are required in order that NRR can make use of B&W plant accident analyses confidently.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

~~8212130305~~

Dupe of
~~8212130305~~