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

JAN 24 1980

REPORT OF INVESTIGATION

TITLE: Babcock & Wilcox NPGD/Possible Violation of 10 CFR Part 21

CASE NUMBER: 79-HQ-003

SUPPLEMENTAL: Vendor Number 99900400

PERIOD OF INVESTIGATION: October 25, 1979 - November 26, 1979

STATUS OF INVESTIGATION: PENDING

REPORTING INVESTIGATOR: William J. Ward JAN 24 1980
William J. Ward, Senior Investigator
Executive Office for Operations Support, IE:HQ

PARTICIPATING PERSONNEL: Howard A. Wilber JAN 24 1980
H. A. Wilber, Senior Reactor Inspection Specialist
Division of Reactor Operations Inspection, IE:HQ

REPORT APPROVED BY: George C. Gower 1-24-80
George C. Gower, Acting Executive Officer for Operations Support, IE:HQ

8004280149

SUMMARY

An investigation was conducted concerning B&W Nuclear Power Generation Division (NPGD) handling of two technical issues during the period November, 1978 - January, 1979. The two issues as described in B&W NPGD memoranda provided NRC by the Kemeny Commission, were whether a proper analysis had been done for small break LOCAs with Reactor Coolant (RC)Pumps powered and whether a proper analysis had been done for a 10-foot steam generator (SG) setpoint level. The investigation was also to determine whether B&W NPGD's handling of this information was in compliance with the reporting requirements of 10 CFR Part 21.

Investigation at B&W NPGD, including interviews of the authors of many of the documents provided by the Kemeny Commission, revealed that B&W's concerns about these two issues were occasioned by a request from the Toledo Edison Company (TECO) to reduce the SG level from 10-feet to 3-feet in November, 1978. As a result of that request, A B&W engineer noted that the original analyses had been calculated at a 32-foot level; he could find no formal documentation for the 10-foot level although he was aware that B&W's engineering judgment was that the 10-foot level was bounded by the original analysis. Other employees subsequently noted that RC pumps powered during small breaks was also unanalyzed. Various employees reported this in internal documents, urged that the formal analyses be done, and suggested that neither NRC nor TECO be informed of the lack of analyses. When interviewed, all these employees averred that they did not feel that either issue represented a safety hazard and that the recommendation to withhold this information from the NRC & TECO was an attempt to protect B&W from an NRC overreaction to what they perceived as a technicality. All employees interviewed stated, (two of them under oath) that they felt that the information was not of safety significance and was not reportable in accordance with the existing B&W NPGD Part 21 reporting procedures. (B&W Part 21 procedures revised in November, 1979 have been expanded to include matters which have "safety implications".) Two employees expressed their belief that TECO was aware of the lack of the formal analyses. The investigation continues in a Pending Status.

DETAILS

PREDICATION AND BACKGROUND

On September 10, 1979, the NRC/TMI Special Inquiry Group forwarded a number of copies of documents to the NRC Executive Director for Operations that had been subpoenaed by the President's Commission on the Accident at Three Mile Island (hereinafter referred to as the Kemeny Commission), and inquired as to what actions the NRC had taken from the standpoint of compliance with 10 CFR Part 21. These documents and the request were subsequently referred to the Office of Inspection and Enforcement (OIE) for appropriate action. Upon receipt in OIE, the Division of Reactor Operations Inspection (DROI) began a technical review to determine the significance of the subpoenaed documents and to ascertain whether an investigation was warranted. On October 24, 1979, the Assistant Director for Technical Programs, DROI, Mr. Edward L. Jordan, provided the results of this analysis to the Executive Officer for Operations Support and asked that a Part 21 investigation be conducted by the IE:HQ Investigative Staff in view of its apparent interrelationship with similar Part 21 investigations involving the B&W Nuclear Power Generation Division (NPGD) that had been previously conducted by the Staff.

The investigation was initiated on October 25, 1979 with a view to determining the circumstances surrounding B&W NPGD's handling of two issues. The first was whether B&W had properly analyzed the consequences of a small break Loss of Coolant Accident (LOCA) with the Reactor Coolant Pumps (RC pumps) running, and whether their handling of the issue indicated non-compliance with the reporting requirements of Part 21. The second issue was whether B&W performed an analysis of a 10-foot Steam Generator (SG) setpoint and whether the handling of this related situation represented a Part 21 violation.

INVESTIGATION AT LYNCHBURG, VIRGINIA

INTERVIEW OF DAVID NMN MARS

Mars, a B&W Licensing Engineer, when interviewed November 6, 1979 at B&W NPGD, Lynchburg provided the following information in substance: B&W NPGD has a set of written procedures which implement the provisions of 10 CFR 21. These procedures require, inter alia, that any B&W employee who identifies a substantial safety hazard, significant deficiency, or miscellaneous reportable item must file a B&W form called a Preliminary Safety Concern (PSC). Mars described the filing and review process in the following manner.

The PSC must be co-signed by the employee's manager who reviews it in terms of completeness and accuracy. (Mars was uncertain whether a manager had the authority to shortstop the PSC). The PSC is then sent to Mars' unit, Licensing, where it is logged and assigned to a licensing engineer. Copies are then distributed to twelve different units within B&W such as ECCS, Nuclear Service, Project Management, etc. The PSC is then evaluated by the appropriate technical people within B&W. There is no established time limit for such evaluation. An evaluation report is prepared and sent to the same distribution as the PSC had been sent originally. It is reviewed by the recipients who must meet an informal deadline of one week in providing comments to Licensing. There is an informal requirement for all such comments to be addressed.

The revised evaluation report is then sent to the Managers of Quality Assurance and Plant Integration for concurrence. Again there is an informal one week review period, but an extention can be requested. If they concur, they sign it and send it back to Licensing which in turn advises the Vice President who is the responsible officer as set forth in Part 21. He is not asked to concur, but he is asked to acknowledge the notification in writing.

The Licensing Manager either advises NRC telephonically or advises the customer who in turn has 48 hours to notify NRC (if the PSC is adjudged to be a safety hazard).

Prior to Three Mile Island, PSCs were submitted on the average of 20 per year. Only one of five turned out to be reportable. Mars indicated that there were several reasons why a PSC would be cancelled or determined to be not reportable. For instance, if the evaluation discloses that the item is not safety related, has already been reported, or the NRC is already aware, the concern would not be reported. A PSC can be cancelled if it is duplicated by another PSC, is already known by a customer who reports it to NRC in accordance with 10 CFR 50.55e, or if the originator changes his mind. In the latter case, the originator must document his actions.

When advised of the nature of this investigation, Mars stated that he was aware of both issues but was of the opinion that they had been properly handled in accordance with B&W procedures. He added that the issue of the Reactor Cooling pumps has been the subject of a PSC. He subsequently provided a copy of the file associated with PSC 79-16 (enclosure (1)) which deals with the issue of the effects of leaving RC pumps running for a period during small break LOCAs.

INTERVIEW OF NIRANJAN H. SHAH

Shah, a Senior Engineer when interviewed November 7, 1979, at B&W NPGD, Lynchburg, Virginia, provided the following information in substance: In early November, 1978, the Toledo Edison Company (TECO) raised a question as to whether they could establish a 3-foot steam generator (SG) low level setpoint for the Davis-Besse plant. In researching the answer he discovered that the analysis supporting the Topical Report provided NRC (BAW-10075A) was calculated on a 32-foot level. He then wrote a memo November 13, 1978 to Eric Swanson, B&W Plant Integration (enclosure 2) informing him of the foregoing and pointing out that although there have been scoping studies done at the 10-foot level that demonstrate its safety, these had not been reported to the NRC.

He continued to research this issue, and on December 13, 1978, wrote a memo to Lucius Cartin, Plant Integration in which he summarized the data base supporting a 10-foot auxiliary feedwater/SG level control. He noted in this memo (enclosure 3) that it was his opinion that the 10-foot level was safe but a minor cladding temperature excursion may occur. Shah explained that the temperature increase that he cited would be caused by a slight degree of core uncovery, but that the temperature increase would be less than 20 degrees. He pointed out that the increase was hypothetical, a product of the model that was being used at the time. He added that a more sophisticated model is in current use and that he was confident that this model would not show such an excursion at the 10-foot level. He stated that he did not see any reason to consider the filing of a PSC as he did not view this as a safety concern.

Similarly, about this same time, Shah became aware that BAW-10075 did not address the situation of RC Pumps running during small break LOCAs and thus was an unanalyzed issue. He related that he had been informed that such an analysis had been done only for large breaks. Shah emphasized, however, that he did not feel that the mere fact that an issue was not analyzed was sufficient to warrant submittal of a PSC as it did not appear to him to present a substantial safety hazard.

INTERVIEW OF ERIC W. SWANSON

Swanson, a Senior Supervisory Engineer, was interviewed on November 6 and 8, 1979 at B&W NPGD. On the latter occasion, Swanson provided oral testimony under oath in lieu of a written statement, this accomodation being reached after the Manager of the B&W NPGD Legal Department declined to approve the furnishing of a written statement. Swanson provided the following information in substance during those interviews:

Shortly after receiving the November 13, 1978 memo from Mr. Shah, Swanson sent a memo to W. H. Spangler, Nuclear Service. In this November 15, 1979 memo (enclosure 4), Swanson characterized both B&W and TECO as being in a "'risk' position" due to the fact that any indication that the 10-foot level was not analyzed, may precipitate re-analysis and re-licensing. He added, however, that the B&W ECCS Unit felt that the 10-foot level was adequate. The memo further pointed out that the ECCS unit had not done a small break analysis with RC pumps running, and suggested that if such were done, the results could be unfavorable. He further recommended that an analysis be done for the 10-foot SG level.

Swanson could not recall during interview whether he knew at the time that he authored the memo that the 10-foot level was unanalyzed (notwithstanding the clear implication of the memo that such was the case), but he did realize that the RC pump issue was not analyzed. He explained that his remark concerning the possibly unfavorable results of such an analysis were based on his belief that the ECCS needed a high SG level for proper operation, and that the RC pumps would create a low SG level that may be unacceptable. He was subsequently informed by Bert Dunn, Manager of the ECCS Unit, that his assumption was wrong. Based on this assurance from Dunn, he felt that the RC pump running issue did not present a hazard, and thus did not warrant submission of a PSC.

Swanson went on to explain that his use of the term "risk" in his memo did not refer to a hazard, and that he regretted his choice of that word. He felt that the point that he was attempting to convey was that it would been have awkward or uncomfortable for either TECO or B&W if the NRC were to demand documentation of the safety of the 10-foot SG level or the RC pumps running in that B&W would be forced to develop such documentation on a crash basis. Swanson averred that he had no reason to believe that either condition had any safety significance and thus he had no reason to take action in accordance with Part 21. As indicated above, Swanson expressed considerable dismay at having his remarks in an internal B&W memorandum subjected to such intense scrutiny a year after they were written, and characterized the effect upon him as being "extremely demotivating".

INVESTIGATOR'S NOTE: Prior to questioning Swanson, the reporting investigator identified himself by display of credentials and informed Swanson that he was conducting an investigation of a matter within the jurisdiction of the NRC. Swanson was further advised that although he had the right to not answer questions, knowingly and willfully providing false information could constitute a criminal offense. Swanson signified his understanding of the foregoing.

INTERVIEW WITH LUCIUS R. CARTIN

CARTIN, a Senior Engineer, was interviewed November 7 and 8, 1979 at B&W NPGD, Lynchburg. On the latter occasion, Cartin provided oral testimony under oath in lieu of a written statement, this accomodation being reached after the Manager of the NPGD Legal Department declined to approve the furnishing of a written statement. Cartin provided the following information in substance during these interviews:

He acknowledged that he had authored a December 19, 1978 memo to Ray Luken (enclosure 5) that described both the 10-foot SG level and RC pumps powered issues as not being analyzed and suggested that such information be withheld from both the NRC and the customer (TECO). Cartin explained that he felt that neither issue was a substantial safety hazard falling under Part 21/PSC reporting requirements. He was assured by Bert Dunn that RC pumps powered during a small break LOCA would result in less severe effects than would loss of offsite power. Notwithstanding his lack of concern regarding the safety implications of these issues, he was aware that there was a need for further documentation if for no other reason than the NRC may insist upon it. His comments regarding trying to keep this information from NRC were meant in the context that the NRC might demand such an analysis in an unreasonably short time, and that B&W's inability to respond could result in NRC shutting down or derrating plants. Similarly, to notify the customer would be tantamount to notification of the NRC due to the more stringent reporting requirements that apply to licensees. He emphasized that he was not suggesting, nor did he feel, that information of safety significance should be withheld from the NRC. Cartin stated that the same explanation would pertain to similar references made by him in a handwritten memorandum of January 9, 1979 (not enclosed).

Cartin asserted that Mr. Fred Miller of TECO was aware that the 10-foot level was not analyzed. On the other hand, Miller was aware that it was B&W's position that it was bounded by existing analytical assumptions.

INVESTIGATORS NOTE: Prior to questioning Cartin the reporting investigator identified himself by display of credentials and informed Cartin that he was conducting an investigation of a matter within the jurisdication of the NRC. Cartin was further informed that although he had a right to not answer any questions, knowingly and willfully providing information that he knew to be false could constitute a criminal offense. Cartin indicated his understanding of the foregoing.

INTERVIEW OF RAYMOND C. LUKIN

Lukin, Service Manager, was interviewed November 7, 1979 at B&W NPGD, Lynchburg, at which time he provided the following information in substance: He has been involved with activities concerning Davis-Besse since May 1978, and assumed service cognizance over the facility in August 1978. Upon assumption of those duties, he recalled that the 10-foot SG level was considered to be the authorized level. He was then unaware that 32-feet had been used in the Appendix K analysis. He learned that upon receiving copies of memoranda written by Eric Swanson and Lou Cartin in December 1978. Although he became aware of both the SG level and RC pumps powered issues at this time, neither raised any safety concerns in his mind. He was aware that the former had been the subject of scoping studies and that the latter had been looked at at some time in the past. He characterized his feelings at the time as being, "a warm glow" concerning the safety of these two issues, a feeling imparted to him primarily by Bert Dunn, the ECCS Manager. Consequently, he saw no need to intiate a PSC regarding either issue.

Lukin stated that he agreed with the concerns that Lou Cartin expressed in his December 19, 1978 memo regarding possible NRC action being taken if it became known that neither issue had been analyzed in accordance with Appendix K. He felt that NRC, lacking the assurances that he had gotten from Bert Dunn, might overreact to what appeared to him to be a technicality. He reiterated his belief that this did not represent a significant safety issue and thus was not the appropriate topic of a PSC under then existing guidelines. Luken added, however, that if the same facts were to present themselves in today's climate, i.e., post-TMI, he would certainly submit a PSC. Luken also stated that it was his belief that TECO was aware that the 10-foot SG level was unanalyzed and recalled that a TECO representative named Fred Miller was present during some meetings during which this matter was discussed.

INTERVIEW OF ROBERT C. JONES

Jones, a Supervisory Engineer, when interviewed November 7, 1979 at B&W NPGD provided the following information in substance: When questioned regarding his December 11, 1978 memo to Lou Cartin (enclosure 6), Jones explained that the issue of whether RC pumps powered during a small break LOCA has been analyzed came up during a conversation, he could not recall any such analysis although he has since been told by Bert Dunn that an analysis had been done. He then wrote the memorandum in question in order to get the issue analyzed. He claimed that he deliberately wrote the memo in a vague and negative fashion in order to assure funding for the analysis. He said that the analysis was subsequently done by Niru Shah and that although it showed some uncovery, the fuel cladding was adequately cooled by high velocity steam. He stated that he did not perceive this as a safety issue and added that he requires some documentation of a hazard before submitting a PSC in order not to waste time.

When shown Cartin's December 19, 1978 memo, Jones said that he agreed with the implications of the document, i.e., that at the time of the memo, both the RC pumps powered and the 10-foot SG level were unresolved issues. He agreed that the purpose in withholding the information from TECO was to avoid NRC harassment which would be occasioned by TECO's mandatory reporting of the information to NRC. He felt that B&W was in the process of doing the very analyses that would be requested by NRC without having to adhere to an arbitrary deadline. Jones emphasized in conclusion that he did not at any time see that either issue warranted reporting in accordance with Part 21.

INTERVIEW OF BERT M. DUNN

Dunn, Manager, ECCS, when interviewed November 6, 1979 at B&W NPGD, at Lynchburg provided the following information in substance: Although a full Appendix K analysis has not been done for a 10-foot SG level, he is of the opinion that the B&W model supports the 10-foot level was well as the 32-foot that was used in the topical. He described the model as being relatively simple with 32-feet being a rather arbitrary point. Nonetheless, his scoping studies suggested that TECO's request for a 3-foot level could not be supported without further analysis. Dunn verified that this issue came to light as a result of TECO's request for a lower than 10-foot level. He further stated that it appeared obvious that RC pumps powered was a less severe condition than loss of offsite power, an opinion reinforced by his study of the problem both prior and subsequent to the TMI accident. Dunn emphasized that he at no point felt that either issue represented a safety hazard less a substantial safety hazard and for that reason saw no reason to initiate, or have initiated, a PSC.

INTERVIEW OF EDGAR ALLEN WOMACK

Womack, Manager of Plant Design when interviewed at B&W NPGD Lynchburg on November 8, 1979 provided the following information in substance: He was aware of the memoranda generated by Lou Cartin and others concerning the issues of the 10-foot SG level and RC Pumps powered during small break LOCAs. He did not feel that either situation warranted the issuance of a PSC based on comments by members of his staff as well as his own technical perceptions of the subject areas. He noted for instance that the key issue regarding the SG level was not the level per se, but the parameters of temperature and heat transfer. Similarly, the matter of RC pumps powered did not suggest a safety hazard to him even though he had some concern about the resulting void fraction. He added that he also relied on Bert Dunn's judgment that RC pumps powered was not a PSC.

Womack stated that Cartin's remarks concerning keeping this information from the NRC did not represent an attempt to conceal safety information from the NRC. Rather, they were geared to sparing B&W from a possible NRC overreaction to a matter that was essentially a technicality inasmuch as B&W was already confident that no hazard existed and had already initiated steps to do the appropriate analysis.

INTERVIEW OF HENRY A. BAILEY, JR.

Bailey, a Principal Licensing Engineer when interviewed November 7, 1979, at B&W NPGD, Lynchburg, provided the following information in substance: He recalled having received Robert Jones' December 11, 1978 memo regarding the need for a small break analysis with RC pumps powered, but indicated that it did not have much impact upon him at the time. Although he attributed that lack of impact to possibly bad judgment, he averred that he saw no need to take action and did not see that it was of significant safety interest. He stated that if it was important, he would have expected it to be in the form of a PSC which it was not.

Bailey stated that he had only a vague recollection of the SG level issues that were the subject of several memoranda. He claimed that B&W Licensing was apparently only peripherally involved in that issue whereas they had been much more so on the RC pump issue. As an example of the latter, he called attention to the PSC which was described by David Mars during his interview.

Bailey commented that he felt that the prospect of a PSC ending up in the NRC Public Document Room has had a chilling effect upon the use of the PSC system. He explained that there was a reluctance to raise an issue to the level of a PSC without doing some sort of evaluation first. Bailey added, however, that B&W NPGD has initiated new Part 21 procedures that in effect lower the threshold for PSCs to encompass anything that affects safety. These draft procedures are enclosure (7) to this report. The draft procedures were scheduled to become effective on November 20, 1979.

STATUS OF INVESTIGATION

The investigation will remain in a PENDING status awaiting review of this interim report by NRC management.

ENCLOSURES

Item		Dissemination
1.	B&W PSC 79-16	Copy all
2.	Shah to Swanson memo of 11/13/78	Copy all
3.	Shah to Cartin memo of 12/13/78	Copy all
4.	Swanson to Sprangler memo of 11/15/78	Copy all
5.	Cartin to Luken memo of 12/19/78	Copy all
6.	Jones to Cartn memo of 12/11/78	Copy all
7.	B&W Draft Part 21 Procedures	

1

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To
DISTRIBUTION

From
J. H. Taylor, Manager, Licensing (2817)

BDS 663-5

Cust.	File No. or Ref. 205 T4.4 PSC 16-79
Subj. Preliminary Report of Safety Concern PSC 16-79	Date May 31, 1979

This letter to cover one customer and one subject only.

Distribution

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In accordance with Procedure NPG-1707-01, "Processing of Safety Concerns," I am forwarding herewith a reported concern on a small break LOCA should the RC pumps go off the line by any means such as by operator action or loss of offsite power. PSC 16-79 has been assigned to this case.

When my staff has completed its evaluation as to whether a reportable concern exists, I will communicate their findings to you. The point of contact within Licensing on this matter is H. A. Bailey, Ext. 2678.

JHT/fw
Attachment

JHTaylor

J. H. Taylor

MAY 29 1979

ERK

BWNP-20208 (2-77)

BABCOCK & WILCOX
PRELIMINARY REPORT OF SAFETY CONCERNS

PSC 16-79

TO: MANAGER, LICENSING, NPGD

FROM: Charles E. Parks

ORGANIZATION: Plant Design

ATTACH AND IDENTIFY, BY PAGE NUMBER, ANY SUPPORTING INFORMATION/DOCUMENTS

☐ SIGNIFICANT DEFICIENCY

☐ SUBSTANTIAL SAFETY HAZARD

☐ MISC. REPORTABLE ITEM

DATE

FILE NO. T4.4

CONTRACT NO.

PAGE 1 OF

WHEN, HOW AND ON WHICH PLANT WAS THE SAFETY CONCERN IDENTIFIED?

Analysis performed by ECCS Analysis Unit in January, 1979 for the 205FA standard plant.

3 TO YOUR KNOWLEDGE IS CUSTOMER AWARE? ☐YES ☑NO

WHEN & HOW

4 TO YOUR KNOWLEDGE IS NRC AWARE? ☐YES ☑NO

WHEN & HOW See attachment 2

OTHER AFFECTED CONTRACTS (CUSTOMER NAME AND LOCATION)

Possibly All (177, 205 & 145 FA plants)

DESCRIPTION OF SAFETY CONCERN-IDENTIFY AFFECTED COMPONENT(S), SYSTEM(S) OR ACTIVITY/SUPPLIER, AND IMPACT ON SAFETY OF PLANT OPERATIONS

See attachment 1

DESCRIBE CORRECTIVE ACTION COMPLETED/TO BE INITIATED

RESPONSIBLE UNIT ECCS Analysis/Safety Analysis/PS & Controls

SIGNATURE AND DATE

Charles E. Parks 5/25/79

ORIGINATOR DATE

E.W. Womack Jr. 5/29/79

MANAGER DATE

ATTACHMENT 1

In the ECCS Analysis Unit's and Plant Design's progress report for January, 1979, the following item was reported as resolution of a concern over the RC pump status during a small LOCA.

> Small Break Analysis with No Loss-of-Offsite Power - The 0.05 ft^2 break was studied on the 205 F.A. Plants to determine the impact of keeping the R.C. pumps on. Results show a much more rapid loss of R.C. inventory relative to a case with tripped R.C. pumps. While it has been determined that the liquid inventory situation is worse for a pumps running case, hand calculations have been performed which show that, due to the pumps running, a forced flow, steam cooling situation will exist in the core and will result in cladding temperatures of less than 670F. Thus, the pumps tripped case remains a worse situation for small LOCA evaluations. This position will be documented during February.

Examining this case from the standpoint of being able to withstand multiple failures brings about this concern. While the statement above may be true if the RC pumps remain in operation, the case that was run also shows that the reactor vessel would contain only ~ 550 ft^3 of water in 10 minutes after the break should the RC pumps go off line by any means such as by operator action or loss of offsite power.

Since the RV lower head is ~ 900 ft^3, it would take several minutes just to fill the head with only 1 HPI pump. (RC pressure ~ 1300 psia at 10 min) The core temperature transient would probably be unacceptable. Rough hand calcs predict a temperature rise of 300-400 F/min for the hot pin. Assuming a starting cladding temperature of 700F and a 300 F/min rise, clad temperature would reach 2200 F in 5 minutes. The lower head cannot be filled in 5 minutes.

One solution to this problem is to develop a signal to trip the RC pumps such as a low system pressure signal or some new signal such as a low level signal which currently does not exist.

In any case, should a trip signal be installed, a great deal of safety and ECCS analyses would have to be performed or re-examined. On the other hand, if the pumps are not tripped, unacceptable results would probably occur if the RC pumps should go off line. Further study of this situation is warranted.

ATTACHMENT 2

BABCOCK & WILCOX COMPANY
R GENERATION GROUP

E.R. KANE, LICENSING

B.M. DUNN, MANAGER, ECCS ANALYSIS (2138)

BDS 563-5

File No.
or Ref.

Date
May 29, 1979

Telephone Conversation with Zoltan Rosztoczy on May 15, 1979 on Stuck Open PORV With Pumps Running and No AuxFeed

This letter to cover one customer and one subject only.

Mr. Rosztoczy phoned on the afternoon of the 14th with a request that B&W supply information on the expected results of stuck open PORV small break with loss of ~~no~~ auxiliary feedwater and RC pumps running. This was to cover the concern that the analyses provided over the weekend which considered the same case with the RC pumps off may not have been the worst case. I responded that in my opinion they had studied the worst case but that the scenario of events would be altered by the different pump assumption and said that I would consider these in more detail in the afternoon and call them back with a position. At 5PM I was unable to make contact with them and actually made the call at approximately 10:30AM, May 15th. As Mr. Rosztoczy was not available, a discussion was held with Paul Norian, NRC, Bob Jones of our staff also listened in. I described the analysis as follows: A typical small break evaluation of a stuck open PORV without auxiliary feedwater with RC pumps running and with one HPI and realistic decay heat power levels (1.0 ANS). Scenario of events: The system would involve on a homogoneous as opposed to a separated fluid condition and approach high void fractions; at some time between one and two hours it is conceivable that the void fraction could be as high as 75%. (75% is the equilibrium void fraction at a decay heat power corresponding to 3000 seconds. However, the evolving system can probably not reach this void fraction by two to three hours as evidenced by the TMI-2 transient. To allow for the fact that TMI-2 had operating steam generators whereas this event is without operating steam generators I concluded that 75% could be obtained within the RCS somewhere between one to two hours.) If at that time the RC pumps are tripped, the available, 25%, water would fall into two locations, approximately 50% into the RC vessel and approximately 25% each to each steam generator. This would create a solid water level in the reactor vessel of 7 feet or a core mixture level of approximately 8-1/2 feet. If the RC pumps did not coastdown instantaneously, I stated that in my opinion the HPI flow occurring during the pump coastdown would be preferentially distributed to the reactor vessel rather than dispersed throughout the RC system and that this flow would fill the remaining 3-1/2 feet within the core region. Thus it would be my expectation that no core uncovery would take place even if the reactor pumps would trip at the most unfavorable time. Further, should the HPI flow not fill the reactor vessel, the cladding temperature heatup would be minimum and not result in core damage. The heatup would be limited to between 400 and 500°F and the resulting peak temperature could not be in excess of 1300°F. This situation

would last for only about five (5) minutes and after that time core covery would again be maintained. As an over-riding concern, I pointed out that there is no intention within the operating guidelines to cause an RC pump trip during the transient and that this is true regardless of pump performance variables. In other words, I restated our position that at least one pump per loop will run until it dies. I confirmed that my experience with RC pumps running in high void systems has shown no problems with their performance and that our pump experts indicate no concern in pumping a two-phase fluid.

Our phone call ended with Mr. Norian to pass this information on to Mr. Roxztoczy and have followup telephone calls as necessary. I have not, at this time, had further contact on this issue.

BMD/lc

cc: R.C. Jones
E.A. Womack
C.E. Parks

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To	B. A. Karrasch, Manager, Plant Integration E. V. DeCarli, Manager, Quality Assurance
From	J. H. Taylor, Manager, Licensing (2817)
Cust.	
Subj.	Preliminary Report of Safety Concern PSC 16-79

BDS 663-5

File No. or Ref. 205 T4.4 PSC 16-79

Date Oct. 24, 1979

This letter to cover one customer and one subject only.

PSC 16-79 presents a concern for a Small Break LOCA combined with a trip of all RCP at some time after about 2 minutes in the accident.

Pursuant to Procedure No. 1707-01, Licensing has completed its evaluation of the subject PSC and concludes that this is not reportable under the requirements of 10 CFR 50.55(e) or 10 CFR 21.

The Manager, Plant Integration, and the Manager, Quality Assurance are requested to review the attached report, signify concurrence or non-concurrence, sign, date, and return this sheet to Licensing within one week of the above date; a detailed explanation should accompany any non-concurrence. Should you require additional information, H. A. Bailey (Ext. 2678) is the contact in Licensing.

James D. Agar / fw
J. H. Taylor

JHT/fw
cc: H. A. Bailey
G. O. Geissler
Record Center

Plant Integrator Manager Action

Concurrence__________ Non-concurrence__________

Signature__________ Date__________

Quality Assurance Manager Action

Concurrence ✓ Non-concurrence__________

Signature Earl DeCarli Date 11/1/79

H. A. Bailey

ACTION

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To	B. A. Karrasch, Manager, Plant Integration E. V. DeCarli, Manager, Quality Assurance		
From	J. H. Taylor, Manager, Licensing (2817)		BDS 663.5
Cust.		File No. or Ref.	205 T4.4 PSC 16-79
Subj.	Preliminary Report of Safety Concern PSC 16-79	Date	Oct. 24, 1979

This letter to cover one customer and one subject only.

PSC 16-79 presents a concern for a Small Break LOCA combined with a trip of all RCP at some time after about 2 minutes in the accident.

Pursuant to Procedure No. 1707-01, Licensing has completed its evaluation of the subject PSC and concludes that this is not reportable under the requirements of 10 CFR 50.55(e) or 10 CFR 21.

The Manager, Plant Integration, and the Manager, Quality Assurance are requested to review the attached report, signify concurrence or non-concurrence, sign, date, and return this sheet to Licensing within one week of the above date; a detailed explanation should accompany any non-concurrence. Should you require additional information, H. A. Bailey (Ext. 2678) is the contact in Licensing.

James D. Ayers / for
J. H. Taylor

JHT/fw
cc: H. A. Bailey
G. O. Geissler
Record Center

Plant Integrator Manager Action

Concurrence ✓ Non-concurrence ______

Signature B A Karrasch Date 10/26/79

Quality Assurance Manager Action

Concurrence ______ Non-concurrence ______

Signature ______ Date ______

Evaluation of
Small Break With No Loss-of-Offsite
Power Concern

This report documents the evaluation of a concern wherein it is postulated that unacceptable results would probably occur if the RC pumps were tripped after running for some period of time during a small break LOCA.

Identification

The affected plants include all with the B&W NSS. These are:

Oconee 1, 2, and 3
Three Mile Island 1, 2
Arkansas Nuclear One - 1
Crystal River 3
Midland 1, 2
Rancho Seco
Davis Besse 2, 3
North Anna 3, 4
Bellefonte 1, 2
WNP 1/4
Pebble Springs 1, 2
Erie 1, 2
Greenwood 2, 3

Analysis of Occurrence

Recent evaluations have examined the response of the primary system during small breaks with the RC pumps operative. During the transient with the RC pumps operative, the forced circulation of reactor coolant will maintain the core at or near saturation temperatures (no cladding temperature excursion). Small breaks evolve to high RCS void fractions due to high liquid (low quality fluid) discharge through the break as a result of the forced circulation of reactor coolant (Figure 1). The RCS void fraction will increase in excess of 90% in the short term. In the long term, the system void fraction would decrease as the RCS depressurizes, HPI increases, and decay heat diminishes.

The RCS evolution to a high void fraction raises the concern as to the ability of the plant to successfully sustain a RC pump trip by any means (i.e., loss-of-offsite power, manual action, etc.) at the worst possible time during the small break transient. That is, if an RC pump trip is postulated at a time when the system void fraction is greater than approximately 60-70% a core heatup would occur because the residual liquid would not be sufficient to keep the core covered. A cladding temperature excursion would ensue until core cooling is reestablished by the HPI system.

A preliminary estimate of the impact of the pump trip assumption for some of the cases analyzed shows core uncovery times in excess of 500 seconds will occur. Based on previous small break analyses, assuming an adiabatic heatup of approximately 5^{o}F/sec during the uncovery period, the expected peak cladding temperatures for a range of 500 seconds core uncovery will exceed LOCA PCT limit. Table 2 summarizes the core uncovery period for a spectrum of breaks analyzed. For continuous pump operation, the core will be covered and the PCT remains near the saturation temperature during the transient.

Corrective Action

A spectrum of analyses has been performed as shown in Table 1. The results from these preliminary analyses indicated the following:

a. Small breaks with continuous RC pump operation can be mitigated safely.
b. If an arbitrary RC pump trip at the worst time must be assumed, compliance to 10 CFR 50.46 cannot be shown with present plant equipment, realistic operator actions, and a single failure.
c. If an early pump trip is utilized, this action must be completed quickly (1-2 minutes after ESFAS actuation). If a pump trip is not initiated within the specified time frame, the RC pumps should not be secured. Under this circumstance, the operator should concentrate on achieving maximum HPI by initiating an immediate cooldown and depressurization of the primary system.

As a result of the above preliminary analyses, B&W has recommended to the wners Group that the RCP's be tripped immediately upon receipt of an ESFAS ctuation caused by low reactor coolant pressure.

Reportability

Dr. Zoltan Rosztoczy called B&W on May 15, 1979, to request some additional small break analyses related to reactor coolant pump operating assumptions. This request of Dr. Rosztoczy and a repeat of this request on June 8, 1979, is documented in a letter from J. H. Taylor to Dr. R. J. Mattson (NRC) of June 8, 1979, Subject: NRC Request for Additional Small Break Analyses.

Dr. Rosztoczy was briefed again by B&W on this concern by a telephone call on July 5, 1979. J. H. Taylor explained that B&W believed that unacceptable results would occur if the RCP's trip later during the accident for some breaks in the .025 to .2 ft^2 range. B&W requested a meeting with the NRC Staff at this time. July 18, 1979 was proposed by Dr. Rosztoczy for the meeting.

J. H. Taylor called T. H Novak of the NRC on July 10, 1979, to confirm that July 18 would be acceptable as a meeting date. During this conversation, Mr. Taylor reiterated what was told to Dr. Rosztoczy to be certain he understood the purpose for the meeting on July 18.

The NRC was completely informed of the results of the above requested analyses in a meeting with B&W on July 18, 1979. The minutes of this meeting have been distributed by memo from H. Bailey to file 20A3.2 on July 23, 1979. Following this above meeting, the NRC issued IE Bulletin Nos. 79-05C and 79-06C on July 26, 1979. This Bulletin specifies both short-term and long-term actions to be taken by Licensees regarding this concern. Further reporting of this concern is therefore not required, since the Commission has been adequately informed.

Table 1. Analysis Scope With AFW Available

Break[1] size, (ft^2)	Continuous RC pump operation with no SG BWN: 2 HPI[2]	Continuous RC pump operation with no SG BWN: 1 HPI[3]	RC pump trip @ 90% void with no SG BWN: 2 HPI[4]	RC pump trip @ 90% void with 2 HPI's & SG BWN via ADV's: 1 SG	RC pump trip @ 90% void with 2 HPI's & SG BWN via ADV's: 2 SG[4]	RC pump trip @ 90% void with BWN of both SG's via ADV's: 1 HPI[4]
0.025	X					
0.05	X	X	X		X	X
0.075	X	X	X	X	X	X
0.10	X		X		X	
0.2	X		X			

[1]All breaks are located at the RC pump discharge.

[2]With 2 HPI's available, 25% of the total HPI flow is assumed to be lost out the break.

[3]With 1 HPI available, 50% of the total HPI flow is assumed to be lost out the break for the first 10 minutes; after 10 minutes 30% is assumed to be lost out the break.

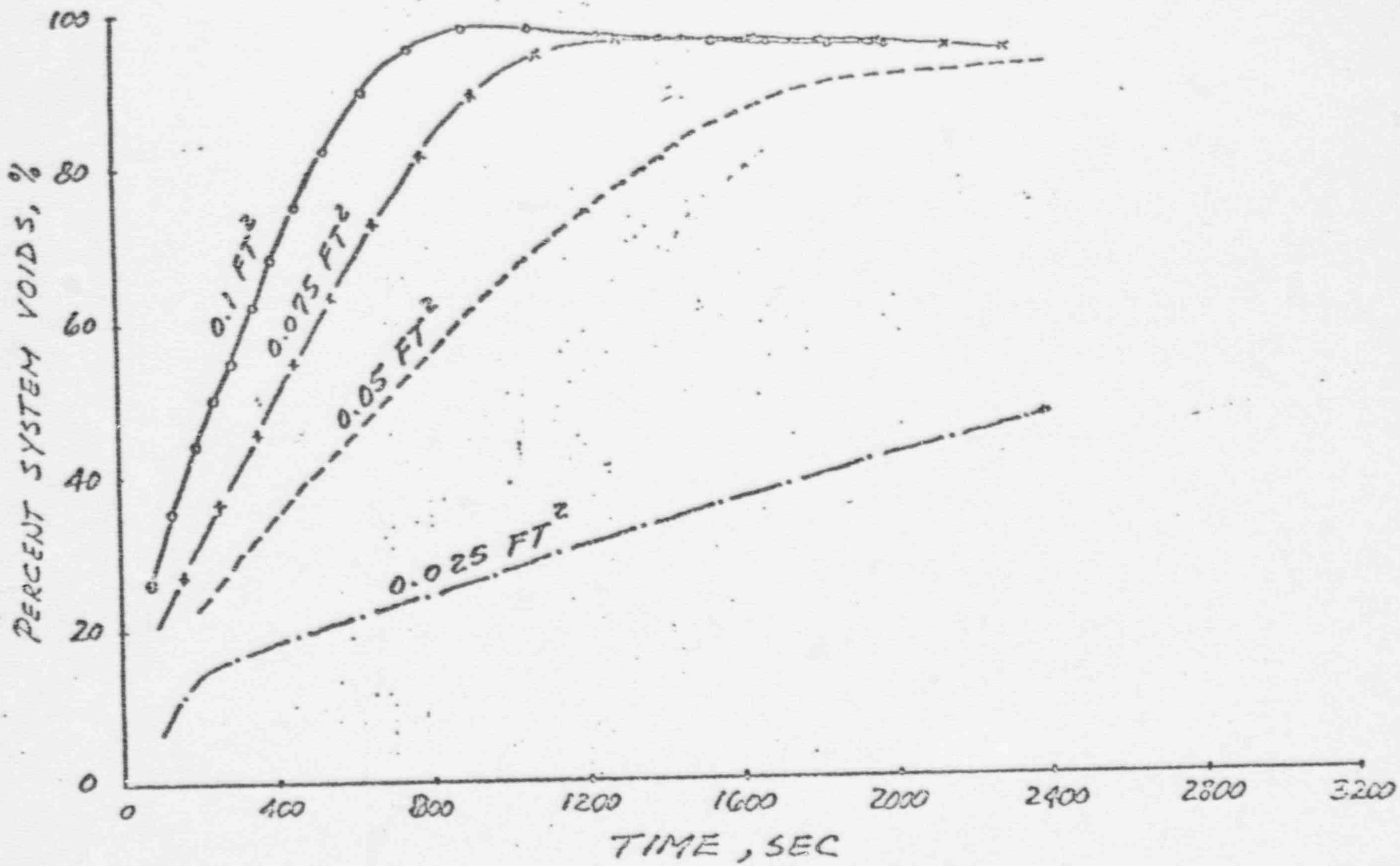
Table 2. Impact Assessment of Break Spectrum With RC Pump Trip at 90% Void

Break Size (ft^2)	Core Uncovery Time (sec)
0.10	550
0.075	625
0.05	575
0.025	0

Notes: 1. Two HPIs available during the transient.

2. Core uncovery time is the time period following pump trip required to fill the inner RV with water to an elevation of 9. ft in the core which is approximately 12. ft when swelled.

FIGURE 1 : Break Spectrum - Average System Void Fraction With the RC PUMPS OPERATIVE and 2 HPI PUMPS



cock&Wilcox

Power Generation Group

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Telephone: (804) 384-5111

June 8, 1979

Dr. R. J. Mattson, Director
Division of System Safety
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Mattson:

Subject: NRC Request for Additional Small Break Analyses

On May 15, 1979, Dr. Rosztoczy called B&W to request some additional small break analyses related to reactor coolant pump operating assumptions. This request was repeated in a telephone conversation on June 8. The requested analyses consisted of the following:

1) Perform an analysis for the worst case break with the RC pumps running, AFW available, with normal Appendix K assumptions (single failure). If the RC pumps trip at the worst time, what are the consequences?

2) Same as case/s 1) except no AFW should be utilized.

It should be noted that "worst case break" and "worst time" are undefined, and the analyses should demonstrate that all breaks and times are covered. It is my impression that Dr. Rosztoczy is concerned about the possibility of high system void fractions evolving in this scenario and that subsequent loss of the RC pumps, for some undetermined reason, will lead to unacceptable consequences.

We intend to discuss this work further with our utility customers on June 13 and will advise you of the outcome of that meeting.

Very truly yours,

James H. Taylor

James H. Taylor
Manager, Licensing

JHT:dsf

cc: Zoltan Rosztoczy (NRC)
Tom Novak (NRC)
R. B. Borsum (B&W)

ER GENERATION GROUP

FILE	G. O. Geissler JUL 9 1979 Licensing
H. A. BAILEY - LICENSING H. Bailey	BDS 663.5
ALL	File No. or Ref. 20A3.2 205 T4.4 PSC 16-79
SMALL BREAK WITH RCP'S OPERATING	Date JULY 5, 1979

This letter to cover one customer and one subject only.

A telecon was held on Thursday, July 5, between Zoltan Rosztoczy of the NRC and J. Taylor, et al. The call was initiated by B&W to update Zoltan on B&W's followup on our June 8, 1979 letter to R. Mattson concerning additional small break analyses.

J. Taylor started the telecon by explaining that some work had been done which showed that breaks in the range of .025 to .2 ft^2 develop large void fractions if RCP's are allowed to continue operation. He went on to explain that B&W believes, based on preliminary calculations, that unacceptable results (PCT) would occur if the RCP's trip later during the accident for the larger breaks in the above size range of concern.

A meeting was requested for the week of July 23rd to brief the NRC Staff prior to any further work or revision by B&W of the operator guidelines for small breaks.

Zoltan then launched into a discussion of the items the NRC wanted to discuss with B&W and the Owners Group in their scheduled meeting on July 19 and 20. These items are:

a) Small Breaks - a few outstanding items

b) Core Uncovery Procedure - What indications are available? What would the operator do? No mechanism for uncovery was specified.

c) Other Safety Analyses - Look at procedures and guidelines for such things as LOFW, SSLB, Overpressure Transients, Stuck Open Secondary Safeties, etc.

Zoltan added that a NUREG Report was scheduled to be out before the end of July, 1979, and it was to discuss the above items (with resolutions if possible). For this reason, he asked for the meeting (RCP's on with small breaks) to be held on July 18, 1979. B&W agreed to check with the Owners Group and report back to the NRC.

HAB:dsf

cc: J. H. Taylor*
E. A. Womack
G. O. Geissler*
R. E. Ham
B. M. Dunn
R. B. Davis*
C. E. Parks
L. R. Cartin
H. A. Bailey*

*Participants in Telecon

ER GENERATION GROUP

DISTRIBUTION

J. H. TAYLOR - MANAGER, LICENSING (2817) JHTaylor

E. O. Geissler

JUL 10 1979

Licensing

BDS 663.5

File No. 20A3.2
or Ref. 205 T4.4 PSC 16-79

SMALL BREAKS WITH REACTOR COOLANT PUMPS OPERATING - MEETING WITH NRC

Date JULY 10, 1979

This letter to cover one customer and one subject only.

DISTRIBUTION

H. A. Bailey
L. R. Cartin
R. B. Davis
B. M. Dunn
E. O. Geissler
R. C. Jones

R. E. Ham
D. W. LaBelle
D. Mars
C. E. Parks
E. A. Womack

REFERENCE: Memo, H. A. Bailey to File, Same Subject as Above, dated July 5, 1979.

As a followup to the telephone conversation recorded in the above referenced memo, I called Mr. Tom Novak of the NRC to confirm that July 18, 1979 would be an acceptable time to meet with the NRC and review the results of our calculations relative to tripping the pumps at the worst time after a LOCA. Mr. Novak indicated that it would be desirable to have this meeting in the Phillips Building starting at 1:30 PM on July 18. Presently it is scheduled for Room P-422.

During this conversation, I also took the opportunity to reiterate what had been passed on to Dr. Rosztoczy in the referenced telephone call stating that certain cases involving break ranges in the .025 to .2 ft^2 range can lead to unacceptable peak clad temperatures if the reactor coolant pumps are tripped after high void fractions develop in the loop. The purpose of mentioning this to Mr. Novak was to be certain that he understood the purpose for the meeting and that Dr. Rosztoczy had passed on this message to him.

By copy of this memo, Mr. Geissler is requested to arrange for a dry run for this meeting this week - preferably Friday.

JHT:dsf

GENERATION GROUP	
FILE	
H. A. DAILEY - LICENSING (2678) H. A. Dailey	BDS 663.5
OWNERS GROUP	File No. or Ref. 20A3.2/LS-7
SMALL BREAK WITH RCP'S OPERATING NRC MEETING	Date JULY 24, 1979

This letter to cover one customer and one subject only.

A meeting was held with the NRC Staff on July 18, 1979, in Bethesda. The Owner's Group was also represented by several attendees. The purpose of the meeting was to report to the NRC Staff on the work done on the NRC request for additional small break analyses with RCP's tripped at the worst time. This request was documented in J. H. Taylor's letter to Dr. R. J. Mattson of June 8, 1979 (attached). Those noted (*) on copy distribution were present from B&W. The slides used in B&W's presentation are attached.

The 6-Node model and the HPI assumption of loss of all ECC to the broken leg after RCP trip was explained.

The void fraction γ required prior to RCP trip to uncover the DB-1 core was discussed. B&W has done no specific calculations for DB-1, but feels it would be higher than the 63% required to lower the level to 9 feet of collapsed liquid in the 177FA lowered loop plants.

The NRC Staff (Zoltan Rosztoczy) asked if B&W had taken liquid carryover into account. Answer was no, but we would expect better cooling if we did. Zoltan then asked that B&W look at low flooding rate FLECHT tests and extrapolate from that and see if it increases uncovery time.

The ability of the pumps to run during high γ was mentioned. B&W cited the previous submittals on this subject.

Dr. B. Sheron of the Staff suggested the window of break sizes might be larger due to the separation of water downstream of the RCP. He also asked if a partial loss of RCP's had been looked at. Answer was no. He asked if pump degradation effects were considered. B&W - the CRAFT Code has a degradation model, but these effects are not significant.

The size of the breaks was noted to be nonmechanistic except for the .025 ft^2 (spray line). Zoltan asked about the Code Safety size. B&W - about .02 ft^2. Zoltan then noted that two open code safeties would put the break in the window.

The uncertainty of the calculations and how much margin are present were Zoltan's chief concerns. He pointed out that the 1.2 ANS on decay heat was really the only Appendix K conservatism working here. The 1.2 ANS was shown to shorten core uncovery in one case from 600 sec to 400 sec.

Zoltan pressed B&W for a recommendation, but was told the owners would make any recommendation. B&W went on to explain we were looking at the feasibility of a coincident RCP trip based on low RCS pressure ESFAS and void fraction.

B&W promised the Staff an official submittal of the presentation by July 27th. An SER for the B&W operating plants has been written, but this new information will be factored into the SER by the Staff.

HAB:dsf
Attach.
cc: D. H. Roy
J. H. Taylor
E. A. Womack*
C. D. Morgan
J. J. Cudlin
B. M. Dunn
R. C. Jones

w/o attach.
L. R. Cartin*
C. E. Parks*
G. O. Geissler*
M. V. Bonaca*
D. F. Hallman*
M. A. Haghi*
R. E. Ham*
E. W. Swanson*

*Attended Meeting

K. Giessler

8 pgs

BETHESDA OFFICE

JUL 27 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

JUL 27 '79 AM

July 26, 1979

IE Bulletin Nos. 79-05C & 79-06C

NUCLEAR INCIDENT AT THREE MILE ISLAND -- SUPPLEMENT

Description of Circumstances:

Information has become available to the NRC, subsequent to the issuance of IE Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06A (Revison 1) and 79-06B, which requires modification to the "Action To Be Taken By Licensees" portion of IE Bulletins 79-05A, 79-06A and 79-06B, for all pressurized water reactors (PWRs).

Item 4.c of Bulletin 79-05A required all holders of operating licenses for Babcock & Wilcox designed PWRs to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with reactor coolant pumps (RCPs) operating, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in Item 7.c of Bulletin 79-06A (for Westinghouse designed plants) and in Item 6.c of Bulletin 79-06B (for Combustion Engineering designed plants).

Prior to the incident at Three Mile Island Unit 2 (TMI 2), Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the loss of coolant accident (LOCA) situation. This Westinghouse position, has led to a series of meetings between the NRC staff and Westinghouse, as well as with other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the NRC. Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI 2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs.

Action To Be Taken By Licensees:

In order to alleviate the concern over delayed tripping of the RCPs after a LOCA, all holders of operating licenses for PWR facilities shall take the following actions:

Short-Term Actions

1. In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

2. Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200 degrees F is identified.

3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidlines developed under Item 3 above.

5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

Long-Term Action

1. Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

Schedule

The schedule for the short-term actions of this Bulletin is:

Item 1: Effective upon receipt of this Bulletin,

Item 2: Within 30 days of receipt of this Bulletin,

Item 3: Within 30 days of receipt of this Bulletin,

Item 4: Within 45 days of receipt of this Bulletin,

Item 5: October 31, 1979 (as noted in Table B-2 of NUREG-0578, under Item 3).

A schedule for the long-term action required by this Bulletin should be developed and submitted within 30 days of receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office with copies forwarded to the Director, Office of Inspection and Enforcement and the Director, Office of Nuclear Reactor Regulation, Washington, D. C. 20555.

Approved by GAO (B0072): clearance expires 7/31/80. Approval was given under a blanket clearance specifically for generic problems.

From:

Bob

Boisux

2

BABCOCK & WILCOX COMPANY
ER GENERATION GROUP

E.W. Swanson, Integration

N.H. Sh[redacted] (2136) RHShah

BOS 663.3

TECO-1

File No. or Ref. NSS-14/T3.4

Auxiliary Feedwater Level Control

Date [redacted]

This letter to cover one customer and one subject only.

A question is asked whether a low SG level set point for the TECO-1 plant would be adequate. The ECCS Unit has reviewed the impact of low level on LOCA analysis.

The auxiliary feedwater level control is significant for small breaks only. The analysis for the presently approved small break topical report, BAW-10075A, Rev. 1, was based on a 32 ft AFW level. Subsequent scoping studies done, but not reported to NRC, have shown that a 10-ft AFW level control is adequate to assure core safety for a small leak transient. However, a level set lower than 10-ft would require additional computer analyses and possible model and/or hardware changes needing NRC approval.

NHS/lc

cc: [redacted]
R.C. Jones
G.E. Anderson
B.A. Karrasch
R.C. Luken
W.H. Spangler
E.A. Womack

Enc (2)

3

BABCOCK & WILCOX COMPANY
ER GENERAT'ON GROUP

L.R. CARTIN, PLANT INTEGRATION

m

NHS

BDS 663-5

N.H. SHAH, ECCS ANALYSIS (2136)

t.

File No.
or Ref.

TECO NSS 14

j.

Date

10 FEET AUXILIARY FEEDWATER LEVEL CONTROL

DECEMBER 13, 1978

This letter to cover one customer and one subject only.

The following small break analysis base exist for TECO plants using auxiliary feedwater level control of less than 32 feet.

In these analyses it was assumed that the loss of offsite power was coincident with the reactor trip thus initiating the RC pump coastdown, main feedwater coastdown and isolation of secondary side of steam generators from the secondary loop.

(1) CFT line break:

AFW level used: 10 feet.

Status of Calc File: A formal calc file number does not exist as the work was never intended for a formal submittal to NRC. The analysis was done for sensitivity studies for new analytical techniques. A loose compilation of calculational input is available.

Status of Design Memo: A Q/A'd memo exist (no design memo number) which describes the analysis and results. R.J. Salm to H.A. Bailey, "Re-analysis of CFT Line Break," September 14, 1976.

Status of Results: Core always remains covered by a mixture thus no cladding temperature excursion occur and core remains in safe condition.

(2) HPI line break:

AFW level used: 10 feet.

Status of Calc File: Calc 32-4194-00 supports the anal[illegible]s. It is fully Q/A'd and released via a DRN.

Status of Design Memo: A Q/A'd memo exist, without a design memo number, which describes the analysis and results. W.L. Bloomfield to D.B. Tulodieski, "HPI Line Break," T3.4, NSS-14,25,26, January 4, 1977.

Status of Results: With an operator action within 30 minutes, the analysis shows compliance with the Acceptance Criteria of 10 CFR 50.46. The core is always covered by a mixture during the transient.

Enc (3)

(3) 0.5 ft^2 at pump discharge:

AFW level used: 20 feet.

Status of Calc File: Calc File 32-4518-00 is Q/A'd and released.

Status of Design Memo: A Q/A'd memo, without design number, exist which describes the analysis and the results. M. DiQuarto to D.R. Tulodeski, "Davis-Besse 1, 0.5 ft^2 Small Break Analysis," T3.4, February 10, 1978.

Status of Results: CRAFT and FOAM code analysis were needed to show that the core remained covered by a mixture thus maintaining core safety.

Note: It is my opinion that with a 10 feet AFW level control, the core safety will comply with the Acceptance Criteria of 10 CFR 50.46 but a minor cladding temperature excursion may occur.

If we use the presently approved small break analysis model modifications (i.e., two node inner vessel and appropriate phase separation multipliers for all reactor vessel nodes), I feel that the core will remain covered by a mixture and no cladding temperature excursion will occur.

NSH/lc

cc: B.M. Dunn
R.C. Jones
G.E. Anderson
H.A. Bailey
E.W. Swanson
E.A. Womack

4

E BABCOCK & WILCOX COMPANY
WER GENERATION GROUP

C

To	W. H. Spangler, Nuclear Service	
From	E. W. Swanson, Plant Integration	BDS 663.5
Cust.	Toledo-NSS-14	File No. or Ref.
Subj.	Auxiliary Feedwater Setpoints	Date November 15, 1978

Our recent discussions with Toledo personnel regarding their need to reduce the steam generator level setpoint for natural circulation, and B&W's need to maintain a high level because of ECCS small break have led to an impasse.

Both B&W and Toledo are in a "risk" position because the Toledo small break topical was based on a 32' level position; any change to that position may require re-analysis and re-licensing. Nevertheless, a steam generator level value has not been reported to NRC, and the ECCS Unit believes that a 10' level setpoint will be adequate.

Toledo's needs to lower the setpoint are genuine and I offer the following suggestion which you should pursue with Toledo:

1. Alter the control logic of the SFRCS so that it will provide two setpoints. Since a control function cannot be readily placed in an ESFAS system, the SFRCS must be modified. In the presence of an ESFAS signal, the ESFAS sets a priority for operation over any SFRCS signal and directs the SFRCS to provide a high setpoint level control. In the absence of an ESFAS signal, but with an SFRCS generated signal, the SFRCS control setpoint is directed to a low level. A general schematic is attached; other methods of implementing are possible, but this purveys the concept.

2. ESFAS could also initiate auxiliary feedwater and isolate main feedwater. Further investigation needs to be made as to the actual sequence of events. I believe it is now possible for two conditions to exist because the TECo systems do not initiate AFW by ESFAS. These are:

<u>Current Design</u>

Site Condition	Systems Sequence	Control Setpoint
1. Offsite Power Available	ESFAS → ICS	2' (Main Feedwater)
2. Offsite Power Unavailable	ESFAS → SFRCS	10' (Aux. Feedwater)

If my reasoning is correct, the first condition will only provide a 2' control (of main feedwater); no SFRCS signal will occur and the ICS will control. The other condition will cause the SFRCS to respond to a loss of level (most likely) or to a loss of pump power. At any rate, SFRCS will initiate AFW and control to the high setpoint.

Encl (4)

The first condition may or may not be acceptable to [illegible] Analysis; they have not investigated small breaks with RC pumps running. If such an analysis were to be made, the results would probably be unfavorable.

I suggest that TECo confirm that the above sequences are correct before a decision is made to initiate AFW with ESFAS.

3. I believe that further analytical effort will probably be needed by ECCS to confirm that the 10' setpoint is acceptable even though their judgement says it is. I think that some documentation on file will be required to substantiate their claim, but I do not recommend analyses at this time.

4. An additional thought might be considered for limiting the pressurizer draining. Recent investigations for the 205 plants have shown us that the rate of addition of feedwater has a substantial effect on RC temperature drop. The Toledo plant power level only requires about 300 gpm (at about 30-40 seconds after trip) to remove decay heat. Yet the pumps are capable (at design) of about 800 gpm each; with reduced steam generator pressure the addition rate increases by about 25% to 30%. The total flow rate possible tends to introduce subcooled water into the generator, fill to a preset level (possibly as a subcooled inventory--I don't know the effect of heat pickup as the water falls through the tube nest), and then heat up to boiling. A more preferable mode would be to introduce flow at a rate more equal to the decay heat load. An investigation into rate limiting (valve opening restrictions, cavitating venturis) may be worthwhile. Rate limiting may be a full or partial tradeoff for level limiting.

5. Further discussions with TECo about these suggestions are desirable; we will support efforts in this area.

EWS:dh
Attach
cc: H. A. Bailey
L. R. Cartin
B. M. Dunn
B. A. Karrasch
D. E. Leinhart
R. C. Luken
F. R. Faist
N. H. Shah
C. W. Tally
R. O. Vosburgh
R. W. Winks
E. A. Womack

5

POWER GENERATION GROUP

R. C. LUKEN, NUCLEAR SERVICE — LUKEN

L. R. CARTIN, PLANT INTEGRATION

Work: K Dep. 29

File No. or Ref.

[redacted]

Date DECEMBER 19, 1978

Reference: 1. L. R. Cartin to B. A. Karrasch, "TECo - B&W Meeting Minutes", dated November 29, 1978

2. N. H. Shah to L. R. Cartin, "Ten-Feet AFW Control", NSS-14, dated December 13, 1978

3. R. C. Jones to L. R. Cartin, "Small Break Analysis with R.C. Pumps Powered", dated December 11, 1978

Since issuance of Reference 1, several activities have been initiated to assist TECo in their attempts to resolve NRC concerns relative to AFW actuation and control. Attachment 1 is a TECo letter requesting B&W analysis support; our progress or position to date to supply the seven items requested is as follows:

Items 1 - 3

Control Analysis performed the requested analyses and a preliminary report was submitted to TECo on December 5, 1978. Attachment 2 is a summary of the conclusions drawn. The Control Analysis work has been subsequently Q.A.'d and the preliminary conclusions presented to TECo were found to be valid. A formal report for submittal to TECo is scheduled to be completed by December 22, 1978.

Item 4

No plant specific (Davis-Besse 1) ECCS analysis is available or planned to address steam generator water levels when the R.C. pumps are operating. The customer is to be advised at this time to automate the present site instruction which requires auxiliary feedwater control to a ten-foot steam generator level following an ESFAS actuation. Under these conditions, B&W's position is that small break analyses (BAW-10075A, Revision 1) remains valid for Davis-Besse. Further, discussion is presented below on this general subject.

Item 5

ECCS has compiled a listing of available analyses which support the use of a ten-foot auxiliary feedwater control setpoint (see Reference 2). The customer is to be advised that a portion of

End (5)

these [illegible] upon request to support the B&W/TECo position that the topical remains valid. No additional ECCS analysis [illegible] ECCS, if required, plans to defend the validity of topical report via arguments that emphasize that the injection of auxiliary feedwater in the upper part of steam generator tube bundle would be continuous (at least intermittent) and that the simulation of primary to secondary heat transfer process (not steam generator level) in the topical is valid. This memo is also to advise Nuclear Service that approximately 100 manhours and a span time of three weeks would be required to assimilate the analysis results [illegible] auxiliary feedwater control setpoint into suitable form for an NRC submittal.

Item 6

B&W position to TECo is that the status of R.C. pumps should not be included in the dual setpoint control logic at this time. If questioned by the NRC, however, B&W must be in a position to state that the small break topicals have considered the worst possible conditions (i.e., loss of offsite power). Our inability to respond conclusively to such an inquiry could result in the NRC derating or shutting down all of B&W's 177 F.A. operating plants (except SMUD) until the issue is resolved. ECCS (Reference 3) has proposed that a generic study with the R.C. pumps powered be initiated now on the 205 F.A. plant, ECCS model. I agree that this analysis work should take place and be performed at B&W's expense. This course of action will require identification of funding (B. M. Dunn to secure) to resolve this [illegible] small break. If this work effort is completed and results are acceptable, B&W may then be in a better position to support TECo's request to include the status of the R.C. pumps into the dual setpoint logic.

Note: The customer should not be [illegible] analysis [illegible]. It is imperative that B&W be [illegible] to defend an ECCS analysis of this type or to have a planned course of action if results are unacceptable.

Item 7

B&W position is that no new analysis is required. Commitments to perform additional analyses either in the LOCA or long-term safety analysis area are not to be made unless specifically requested by the NRC.

In addition to the above, [illegible] which I feel was a better defended conclusion drawn from B&W supplied analyses. TECo refused to use the

B&W write-up, but they did agree to put less emphasis on ECCS. TECo's submittal to the NRC is given in Attachment 3.

A letter to TECo will be prepared by R. C. Luken of Nuclear Service consistent with the above and/or modified position as soon as concurrance within B&W is obtained.

for LRC

LRC/dww

cc: B. A. Karrasch
E. W. Swanson
E. A. Womack
R. C. Jones
N. H. Shah
[redacted]
R. W. Winks
R. O. Vosburgh

Attachments

9

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E BABCOCK & WILCOX COMPANY
WER GENERATION GROUP

L.R. CARTIN, INTEGRATION	
m R.C. JONES, ECCS ANALYSIS (2066) R.C. Jones	BDS 663-3
t. ALL	File No. or Ref.
j. SMALL BREAK ANALYSES WITH RC PUMPS POWERED	Date DECEMBER 11, 1978

This letter to cover one customer and one subject only.

Presently approved small break analyses have all been performed assuming that offsite power is lost coincident with reactor trip. This results in tripping of the RC pumps, actuation of auxiliary feedwater, and control of the liquid level on the secondary side of the SG at high levels. Should offsite power remain available during the transient, the RC pumps will remain powered and the ICS will control the secondary side water level to approximately 30 inches by throttling the main (or auxiliary) feedwater control valve. Because of the importance of the SG water level to mitigation of small breaks, ECCS recommends an analysis of the RC pumps powered case.

In making this recommendation, ECCS has performed a review to determine what previous analyses are available with the RC pumps powered and also has made an engineering assessment of what would happen under this situation. [redacted] However, it is expected that if any were performed, they were not done at break sizes of interest. With the RC pumps powered, it is expected that natural circulation would be maintained for a longer period of time and would aide system depressurization. Also, maintainance of the system flow would decrease T_{hot} after reactor trip and would result in a lower pressure for system flashing to occur. However, these positive influences would be offset by the decreased ability of the SG to condense steam, following the loss of natural circulation, due to the lower SG level control. It is also expected that, with the RC pumps running, a "steam pocket" will not form in the cold legs and lower quality fluid will exit through the break and thus shorten the time for the system to reach the "boiling pot" mode of the transient. An unknown factor at this time is whether or not the steam generated in the core will separate in the hot leg U-bend or be carried down into the SG and condensed. [redacted] ECCS proposes that the impact of this phenomena be examined in a sensitivity study.

As illustrated above, [redacted] running results in an enhanced ECCS situation. Thus, ECCS recommends an analysis be performed to examine this case. ECCS proposes that this analysis be run on the 205 FA plants for the following reasons:

cont'd (6)

1. The present 205 FA small break topical (BAW-10074) assumes heat removal typical of that for the 177 FA plants. Therefore, a comparison of the pumps on case to BAW-10074 will provide an assessment of the impact on the 177 FA plants.

2. There presently exists an FAC model on the 205 FA plants.

3. The new SG model in CRAFT has been exercised on the 205 FA plants. This model more properly accounts for SG performance during the transient for the 205 FA and VEPCo plants. Comparison of the pumps on case to recent 205 FA plant studies will provide information on the impact for 205 FA and VEPCo plants.

It is expected that this work could be scheduled into the ECCS Unit workload now because of slippage in the NRC small break standard problem and the 205 FA small break work. It is estimated that this work will require 300 mh and 10 CDC hours and a span time of 3 months. Niru Shah will be the contact in ECCS for this issue. If any questions arise, please contact him, or call me on extension 2066.

RCJ/lc

cc: B.M. Dunn
H.A. Bailey
E.W. Swanson
E.A. Womack
S.H. Duerson
B.A. Karrasch

1. This satisfies the curiosity of the small break guys in ECCS, but does not help them establish a setpoint based on pumps
2. It does no good in [illegible] because we [illegible] [illegible]
3. It does not reflect the [illegible] plant
4. [illegible] no assurance that [illegible] [illegible] the test — [illegible] the many variables.
5. [illegible]

7

Olw

NUMBER
NPG-1707-01 (Rev 7)

SECTION	SUBJECT
QUALITY STANDARDS .	PROCESSING OF SAFETY CONCERNS

COMPLETE REVISION

I. APPLICABILITY

ALL NPGD PERSONNEL

II. PURPOSE

To provide an orderly and visible process for identifying, evaluating and initiating the resolution of safety concerns related to or affecting NPGD-supplied components, systems and services.

To assure compliance with NRC regulations (e.g., 10CFR21, 10CFR50, etc.).

II. EFFECTIVITY

All safety concerns identified after the issue date of this procedure. Safety concerns identified prior to the issue date of this procedure may be processed in accordance with this procedure.

IV. REFERENCES

NPG-1703-01 - Preparation and Processing of Internal Deficiency Report/ Restraint Order/Corrective Action Request
1716-A1 - Policy for Reporting of Defects and Noncompliance as Required by 10CFR21

V. FORMS PROCESSED (See Forms Section Manual)

BWNP-20208 - Preliminary Report of Safety Concerns (PSC)

VI. GENERAL

A. If guidance in interpretation of the following definition or other aspects of the procedure is needed, consult with Licensing.

B. The general definition of a safety concern as used in this procedure is:

Any item which has been discovered during design, analysis, fabrication, installation, testing, inspection, training, and operations activities of a nuclear power plant and which has or may have safety implications.

C. Reporting of safety concerns to the customer or the NRC is not required if NPGD Licensing has documented evidence that the concern is adequately known to the affected NPGD customers in the case of potential significant deficiencies, or is adequately known to the NRC in the case of potential substantial safety hazards.

D. Recurrence of a previously reported safety concern shall be reported as a new safety concern.

End (7)

REV STATUS	REV	7	7	7	7	7	7
OF PAGES	PAGE	1	2	3	4	5	6

I. GENERAL (cont'd)

E. Once a PSC is issued, the originator may rescind it by documenting the basis in a memo to the Manager, Licensing, and attaching supporting documentation as necessary. The originator's manager shall indicate concurrence on the memo. Appropriate action shall be taken by the Manager, Licensing.

I. RESPONSIBILITIES FOR REPORTING

All NPGD personnel are responsible for originating form BWNP-20208 when they discover potential safety concerns that are suspected of falling within the definition given in Section VI.B. above.

I. PROCEDURE

Refer to flowchart, Exhibit A, for the procedure to process safety concerns.

- E N D -

EXHIBIT "A"

PROCESSING OF SAFETY CONCERNS
(SEE NOTE 1)

PSC INFORMATION COPY

1 ORIGINATOR

Upon discovering or receiving a report of a potential safety concern suspected of falling within the definition of Section VI.B of the procedure, prepare form BWNP-22 , "Preliminary Report of Safety Concerns (PSC)".

Review with and obtain manager's signature on form indicating accuracy and completeness of information.

1A QUALITY ASSURANCE

PSC

2 LICENSING

STEP 2A

Review for adequacy; if necessary, request additional information from originator. See Note 2. Assign PSC number and log in. Prepare and distribute cover memo and information copy of PSC to distribution listed in Note 3.

STEP 2B

As required, convene meeting(s), requesting attendance from the following persons:

a) Originator of PSC
b) Licensing Engineer assigned to handle safety concern
c) Project Management function
d) Others, as necessary, such as Licensing Section Manager, Unit Manager or designee; representative of Plant Integration; Quality Assurance.

Purpose of the meeting:

a) Promote understanding of concern
b) Determine applicability of concern to operating plant and plants in construction
c) Determine safety significance
d) Establish priority for evaluation of concern
e) Scope evaluation and follow-up action requirements (see Note 6) including assignment of responsibilities and schedule. Identify lead organization responsible for follow-up actions.
f) Determine funding requirements and sources

The Licensing Engineer shall prepare a summary of meeting results and distribute to attendees and those listed in Note 3.

STEP 2C

Obtain information from the organizations as necessary to complete the evaluation.

Determine whether or not the reported safety concern is reportable under federal regulations.

If PSC is not to be evaluated by NPGD, forward safety concern to customer, as deemed necessary, via Project Manager function with memo to distribution listed in Note 3. (Procedure ends here.)

Prepare evaluation report and cover memo. Distribute in accordance with Note 3. See Note 4 for content of report.

PSC

3 PROJECT MANAGER (FUNCTION)

Advise customer of potential safety concern if required by contract or customer agreement. Send copy of letter to Manager, Licensing, and Records Center.

STEP 2A — PSC, COVER MEMO

STEP 2C — EVALUATION REPORT, COVER MEMO

4 LEAD MANAGER

Prepare Follow-up Action Plan (FAP) (see Note 7) as required by Step 2. Obtain approval of FAP from managers responsible for actions, and from Manager, Licensing, and Project Management function. Distribute FAP to Note 3 distribution.

Provide information as necessary to Licensing for evaluation report. Document progress as necessary with a final disposition to Licensing (Step 7B) and distribution to those listed in Note 3.

FAP

5 QUALITY ASSURANCE

Review FAP to determine if further action is required per NPG-1703-01. Advise Licensing of action taken.

6 RECIPIENTS

Review evaluation report and notify the Manager, Licensing, of any suggested corrections, additions and other clarifications.

COMMENTS

A

EXHIBIT "A" (cont'd)

A

EVALUATION REPORT
COVER MEMO

7 MANAGER, LICENSING

STEP 7A

Incorporate comments or resolve with originator. Revise evaluation report, if required, and obtain concurrence (signature on cover memo) from Managers, Quality Assurance and Engineering (see Note 5).

Determine reporting strategy, i.e., to whom, format, timing, etc. If the safety concern is identified as a:

1. Substantial Safety Hazard for Equipment, Information or Services Delivered to an NPGD Customer (10CFR21 reportable), promptly provide evaluation report to the Division head and applicable Project Manager with copy to distribution listed in Note 3. If written confirmation is not available within 48 hours from the customer that the NRC has been notified, notify the NRC of the safety concern by telephone, TWX or telegram.

   Substantial Safety Hazard for Equipment, Information or Services Delivered to NPGD or any Supplier Supplier (10CFR21 reportable), promptly provide evaluation report to the Division Head and applicable Project Manager with copy to distribution listed in Note 3. Notify the NRC of the safety concern within 48 hours by telephone, TWX or telegram with concurrent notification to the supplier via the procurement responsibility.

   Prepare cover memo and forward with record of pertinent telephone information or copy of TWX or telegram to the Division Head and distribution listed in Note 3.

   Submit written report within 5 days either directly to the NRC, or via applicable Project Manager to customer for submittal to the NRC, with copy to distribution listed in Note 3.

2. Significant Deficiency (10CFR50.55(e) reportable), prepare cover memo and forward with evaluation report to the distribution listed in Note 3 and Division Head.

3. Other Reportable Item, prepare cover memo and forward with evaluation report to the distribution listed in Note 3. Advise customer or NRC as required.

4. Non-Reportable Concern, prepare memo stating disposition, attach evaluation report, and forward to distribution listed in Note 3.

STEP 7B

Maintain PSC file active until follow-up actions are completed.

8 RECORDS CENTER — Process and file.

EVALUATION REPORT
COVER MEMO

SUBSTANTIAL SAFETY HAZARDS ONLY

9 DIVISION HEAD — Review report. Indicate time and date of receipt and sign cover memo. Return copy of signed memo to Manager, Licensing.

NOTE: Time of receipt by Division Head initiates 48-hour reporting period required by 10CFR21.

EVALUATION REPORT
COVER MEMO

AS REQUIRED

10 PROCUREMENT RESPONSIBILITY — Notify supplier with copy of notification to distribution listed in Step 11.1.

EVALUATION REPORT
COVER MEMO

11 PROJECT MANAGER FUNCTION

1. For Substantial Safety Hazards, promptly notify the customer with copy of notification to the Division Head and Managers, Quality Assurance, Project Management, Licensing, Engineering, Customer Service and the Records Center. Where customer has notified the NRC, provide the Manager, Licensing, with written confirmation that the NRC has been notified by the customer.

2. For all other reportable safety concerns, advise customer by letter of final determination with copy to Licensing.

WRITTEN EVIDENCE
LETTER TO CUSTOMER

12 RECORDS CENTER — Process and file.

NOTES ARE ON CONTINUATION PAGE.

NOTES:

1. All correspondence related to safety concerns shall reference file point 205/T4.4 plus the PSC number.

2. A safety concern that is found to duplicate the subject material of a previously submitted concern shall be returned to the originator with explanation and copy of the previously submitted safety concern.

3. DISTRIBUTION:

   Originator
   Records Center
   Manager, Quality Assurance
   Manager, Licensing
   Manager, Field Engineering and Services
   Manager, Generic Projects
   Manager, Integration
   Manager, Engineering
   Manager, Plant Design
   Manager, Safety Analysis
   Affected Project Manager(s)
   Other Affected Personnel, as applicable

4. Evaluation Report shall contain, as a minimum, the following:

   a. Description of concern
   b. How concern was discovered
   c. Analysis of safety considerations
   d. Equipment and plants affected
   e. Reportability under 10CFR50.55(e) and/or 10CFR21
   f. Corrective actions, as applicable, taken or to be taken

5. QA Manager's concurrence indicates that the applicable NPGD organizations have participated in the evaluation and that an assessment has been made to determine if changes are needed to the QA Program requirements (e.g., increased number of QC Surveillance inspections, increased number of vendor audits, etc.).

   Engineering Manager's concurrence indicates that the evaluation report has been reviewed for accuracy with respect to:

   a. Components, systems, services and plants affected
   b. Nature of the defect or failure to comply and evaluation of the safety concern
   c. Corrective action

NOTES: (cont'd.)

6. Follow-up action is defined, for purposes of this procedure, as follows:

   a. To bring deficient items into conformity with requirements
   b. To identify causes for deficiency
   c. To prevent recurrences of deficiency
   d. To make such other investigations or analyses or take such other follow-up actions as are deemed necessary because of the repeated concern.

7. The follow-up action plan shall include as applicable:

   a. Actions to be taken
   b. Individuals or organizations responsible
   c. Schedule for completion including milestones
   d. Decision points and alternate actions
   e. Funding source

DRAFT

BWNP-20208-2 (11-79)

BABCOCK & WILCOX
PRELIMINARY REPORT OF SAFETY CONCERNS

1 TO: MANAGER, LICENSING, NPGD
(CC: QUALITY ASSURANCE)

FROM: ____________

ORGANIZATION: ____________

CONSULT NPGD LICENSING FOR ASSISTANCE IN COMPLETING THIS FORM

LOG NO. ____________

FILE NO. T4.4

CONTRACT NO.

PAGE 1 OF ______

ATTACH AND IDENTIFY, BY PAGE NUMBER, ANY SUPPORTING INFORMATION/DOCUMENTS

2 WHEN, HOW AND ON WHICH PLANT WAS THE SAFETY CONCERN IDENTIFIED?

3 TO YOUR KNOWLEDGE IS CUSTOMER AWARE? ☐YES ☐NO

WHEN & HOW ____________

4 TO YOUR KNOWLEDGE IS NRC AWARE? ☐YES ☐NO

WHEN & HOW ____________

5 OTHER AFFECTED CONTRACTS (CUSTOMER NAME AND LOCATION)

6 DESCRIPTION OF SAFETY CONCERN-IDENTIFY AFFECTED COMPONENT(S), SYSTEM(S) OR ACTIVITY/SUPPLIER, AND IMPACT ON SAFETY OF PLANT OPERATIONS

7 DESCRIBE CORRECTIVE ACTION COMPLETED/TO BE INITIATED

RESPONSIBLE UNIT

8 SIGNATURE AND DATE

ORIGINATOR ____ DATE ____ MANAGER ____ DATE ____