Ref 2

From: Lawrence P. King
To: CAC1
Date: 4/12/96 10:03am
Subject: Difference of Professional Viewpoint

During an inspection at Oconee from March 25 to April 5, 1996 I reviewed several modifications as part of the core inspection. One of these modifications was NSM-12873, Modify feedwater controls on Main Steamline Break (Unit 1). I originally expressed concerns on this modification during a service water inspection when I discoverd that the main feedwater valves were not closed on a MSLB. I was aware that this had become a requirement at other B&W plants as a result of an NRC concern in IE Bulletin 80-04. I was told and showed correspondence that Duke had discovered this problem and had written an LER 269/96-01 which was followed by a commitment to install a modification. I found it difficult to understand why this concern had not been resolved in the response to the IE Bulletin. I reviewed Duke responses to the NRC in NUDOCS and found that responses to Franklin Institute stated there were no problems. Apparently this response was deemed acceptable by NRR and an SER was issued accepting the response. My limited review showed that other B&W plants complied with the bulletin with the exception of making the feedwater valves safety related. It is my understanding that the actuation system for closing the valves at other B&W plants are safety related including installation of high speed operators on the main feedwater block valves that could secure feedwater on a failure of the control valve to close.
I have referenced correspondence in the recent inspection report that is under review that seems to indicate that NRR has accepted the fact that the modification is not all safety related and is not single failure proof. Duke is relying on operator actions within 120 seconds and 20 seconds to prevent overpressure on a single failure.
I disagree with the acceptance of this mod by NRR in its present installation on Unit 1 as non safety related and non single failure proof. This seems to be a bad precedence to set when we have other licensees who responded to the IE Bulletin in detail and met the requirements. I classify this as "different strokes for different folks." I believe we should deal evenly and fairly with each licensee. Duke seems to want to pin the blame on B&W for the initial response but the other plants had the same information and responded properly to the bulletin.

I am however more concerned about the adequacy of the design of the control system for the emergency feedwater pumps as a whole before installation of the modification. I believe we should review the present control system for the emergency feedwater system to determine if it is single failure proof. John York recently showed me an INPO report that stated Duke had a high number of trips overall and that several of these were as a result of systems or components not being single failur proof. I am particularly concerned because of the initial low water inventory in the OTSGs. It takes less than a minute to dry out the OTSGs and cause the primary temperatures to increase due to loss of heat sink.
I have reviewed management directive 10.159 and ROI 2304, Rev 1 and assume that this DPV will be processed with haste due to the present status of the MOD. It has been installed on Unit 1 and is presently being installed on Unit 2 during the present outage.

I have selected four people as candidates for the review although I have expressed my concerns in the past to the EDO office that the individual identifying the problem has no vote.

The individuals who I consider most knowledgeable I have CCd on this E mail. They are Paul Harmon, Paul Kellogg, Randy Moore and Tom Peebles.

Casto, C.A.

I am attaching previous correspondence which might be useful.

CC: PEH, PJK1, RLM, TAP, AAA

Files: P:\MSLB.2

MEMORANDUM FOR: Paul Kellogg, Chief
Operational Programs Section
Operations Branch

FROM: Larry P. King, Reactor Engineer

SUBJECT: MSLB AT OCONEE

Background

On February 8, 1980, IEB 80-04 was transmitted to Duke Power Company (DPC) for a response within days. On May 7, 1980 reference 11 transmitted DPC's response to Region II. This response relied on operation action to prevent overpressurization and stated a PRA was being done for Oconee. References 9 and 10 requested additional information and stated that DPC's response was not sufficient to allow Franklin Research Company (FRC) to complete the evaluation of the potential for exceeding containment design pressure. Since the ICS is non-safety, the ICS cannot be relied upon to function correctly in an accident. Failure of the ICS may cause both MFW and AFW runout flow to the affected steam generator resulting in overpressurization of the containment. DPC did not indicate the time necessary for the operator to take action. The request included the following:

1) an evaluation of the potential for exceeding containment design pressure using MFW and AFW runout flow rates

2) provide the time the start of a MSLB that containment design pressure will be exceeded if no operator action is taken to terminate the accident

3) provide the magnitude of the peak pressure and the time at which the peak occurs

4) provide action required to be preformed by the operator to prevent exceeding containment design pressure and provide justification for the time at which credit is taken for operator action

Reference 10 also required a response to ANSI N660, Time Response Criteria for Safety-Related Operator Actions, dated March 1981 if operator action is required to terminate the action and provide justification for the time at which credit is taken for operator action. On April 9, 1982 reference 8 forwarded the request for additional information contained in reference 9. On May 7, 1980 DPC responded in reference 11 and did not address the stated requests but claimed runout could not occur because of the design of the MFW and AFW systems. The NRC and FRC accepted this incomplete response and issued reference 5, the SER on October 14, 1982.

Discussion

During the service water inspection at Oconee while reviewing the FSAR for accident analysis it was noted that no analysis was done which required the motor operated valves to terminate feedwater on a MSLB. As part of the service water inspection and questions arising as to operability of the RB coolers, Mr. Rapp and I went to Duke offices in Charlotte. We questioned DPC engineering on the MSLB and were told that recent analysis showed the design pressure of the containment would be exceeded. They stated that a letter and a topical report had been sent to NRR concerning this issue. I mentioned the concern to Mr. Gibson and at his request forwarded reference 16 to him with my knowledge of the issue at that time. During the service water inspection at Surry, I received a call from you to proceed to Charlotte on the following week. I than received a call to my home on Friday that canceled the trip because Mr. Gibson wanted to talk to me first.

I am still concerned that NRR has done an inadequate review and has accepted the licensees word. In the following paragraphs I will attempt to outline what I know to be the recent facts. I will state here however, that I find it incredulous that DPC has just discovered this problem.

On May 27. 1993 DPC transmitted reference 4 stating that they have been reanalyzing Chapter 15 MSLB transient to determine the limits that the accident might impose on plans for extended fuel designs and reload design optimization. Reanalysis of the containment response identified that containment design pressure is exceeded without operator action to isolate MFW. It states that it meets the three acceptance criteria in Chapter 15. but does not mention that Chapter 14 under Reactor Protection Criteria states in e) that the reactor building pressure due to mass and energy release within the containment boundary during the accident shall not exceed the reactor building design limits. It also states that the equipment required to mitigate the consequences of the MSLB is qualified and would perform its safety function. There is no analysis to support this and reference 11 shows that even when the design pressure is not exceeded the temperatures exceeds the EQ requirement of 310 F in the short term and barely stays under the 290 F in the long term. Credit is taken for the RB cooling and RB spray to maintain temperature below the EQ requirements in the long term. The service water inspection identified problems with insufficient flow to the RB coolers in an accident scenario which brings into doubt it's ability to perform during an MSLB. A review of the topical report showed no analysis for the situation on page 3 of reference 4 which stated "Without credit for Automation Main Feedwater Control and Main Feedwater Control Valve Sticks Open." It claims that this analysis requires the operator to take action at 120 seconds to limit the containment pressure to approximately 140 psig. All the other scenarios require action in 170 seconds. Page 4 of the reference states that yielding will take place at 144 psieg. In reference 2 dated August 19. 1993 DPC committed to a design change and sited the justification for the delayed times in implementing these changes that 1) the equipment is EQ qualified and 2) the accident is bounded in the FSAR based on the off-site dose consequences. DPC also used PRA justification for the delay in implementing the design changes.

I disagree with NRRs acceptance to reference 2 for the following reasons.

No analysis was included for the case "Without credit for Automation Main Feedwater Control and Main Feedwater Control Valve Sticks Open" was provided in referenced 12

No proof is shown that the instrumentation will meet the higher temperature as a result of the 140 psig containment pressure. Saturation temperature for 140 psig is 360 F.

Review of other PRAs indicate that a continuing supply of feedwater to the affected steam generator or failure to isolate the non-affected steam generator will lead to decreasing RCS temperature but the maintenance of high pressure as a result of high pressure injection flow. The net effect will be the potential for pressurized thermal shock. The potential for steam generator tube rupture in the affected steam generator also exists. This is effectively a small break LOCA as the steam line break is in containment. If feedwater flow is terminated to the affected steam generator then steam flow into the containment will be terminated. However, if feedwater is NOT isolated, there will be a continuous heat transfer from the RCS to the secondary side and into containment. In this case, it is assumed that containment heat removal will be required as the accident is equivalent to a LOCA in that majoroyt of decay heat is being transferred to the containment building.

LER 94-10-01 showed that both EFW headers exceeded the continuous operation flow limit of 1098 gpm as stated in the EFW design document. The limit is imposed to protect the steam generator tubes from the effects of flow vibration. This condition would occur in a MSLB as the EFW pumps attempted to maintain steam generator level with no back pressure. The net effect could result in tube failure. The LER stated that an evaluation of EFW response indicated that design flow could have been exceeded due to extremely low steam generator pressure.

I do not believe this problem was discovered until 1993. Crystal River and Davis Besse have secured feedwater automatically for years.

In conclusion. I learned form the SRI at Surry that the block valves do not automatically close. Failure to automatically secure feedwater would seem to ba a problem at other plants as well.

References

1) L. Weins, NRC to J. W. Hampton, V. P. Oconee Site
Subject - Containment Pressurization Due to MSLB Inside Containment - Supplemental to IEB 80-04 - Oconee Units 1, 2, and 3 - dated October 6, 1993.

2) J. W. Hampton, V. P. Oconee Site to NRC
Subject - Supplemental Response to IEB 80-04 dated August 19, 1993

3) J. W. Hampton, V. P. Oconee Site to NRC
Subject - LER: Design Deficiency Results In a Condition Outside the Design Basis for Main Steam Line Break dated July 1, 1993

4) J. W. Hampton, V. P. Oconee Site to NRC
Subject - Reanalysis of Main Steam Line Break Inside Containment

5) J. Stolz, Chief Operating Reactors Branch #4 USNRC to H. B. Tucker, Vice President Nuclear Production Department.
Subject - Safety Evaluation Report for IEB 80-04 dated October 14, 1982.

6) Technical Evaluation Report for Oconee Units 1, 2, and 3 dated September 28, 1982.

7) W. O. Parker Jr., Duke Power Co. to H. R. Denton, USNRC
Subject - Additional Response to IEB 80-04 dated July 23, 1982

8) J. Stolz, Chief Operating Reactors Branch #4 USNRC to W. Parker
Subject - Request for additional information dated April 9, 1982

9) S. Purdey, Project Manager Franklin Research Institute to S. Bajwa, USNRC
Subject - Request for additional information dated March 15, 1982

10) S. Carfagno Franklin Research Institute to S. Bajwa, USNRC
Subject - Request for additional information dated January 6, 1982

11) W. O. Parker Jr., Duke Power Co. to J. P. O'Reilly Region II USNRC
Subject - Response to IEB 80-04 dated May 10, 1980

12) Oconee Topical Report DPC-NE-3003-P Mass and Energy Release and Containment Response Methodology transmitted by letter M. S. Tuckman to NRC dated August 11, 1993

13) LER 10 Report No. 04/18/94

14) FSAR Oconee Rev.11 Chapter 14.2 Standby Safeguards Analysis

15) FSAR (13 DEC 1992) Chapter 15.13 Steam Line Break Analysis

16) L. P. King, Reactor Engineer to A. Gibson, Director Division of Reactor Safety Region II dated February 22, 1994
Subject - Excessive Pressure which exceeds design at Oconee