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June 16, 1997

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/97-003

Gentlemen:

Attached is Licensee Event Report 413/97-003 concerning finding the Auxiliary Feedwater System Outside Of The Design Basis.

This event is considered to be of no significance with respect to the health and safety of the public.

Cordially,

W. R. McCollum, Jr.

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IE22

Attachment

cc: Mr. L.A. Reyes
Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

Mr. P. S. Tam
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. R. J. Freudenberger
NRC Resident Inspector
Catawba Nuclear Station

INPO Records Center
700 Galleria Place
Atlanta, GA 30339-5957

Marsh & McLennan Nuclear
John Hoffman
301 Tresser Blvd.
Stamford, CT 06904

NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMD NO. 3150-0104
EXPIRES: 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	1 of 7

TITLE (4)
Auxiliary Feedwater System Found Outside of Design Basis

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER(S)
									Catawba Nuclear Station, Unit 2	05000414
05	15	97	97	- 003	- 00	06	16	97		05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	x 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (*Specify in Abstract below and in Text, NRC Form 366A*)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
D. P. Kimball, Safety Review Group Manager	(803)	831-3743

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	x NO				

ABSTRACT *(Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines)* (16)

Event Description: On 5/15/97, with both units in Mode 1, at 100% power, engineering was reviewing the hydraulics of the non-safety suction source to the Auxiliary Feedwater (CA) pumps, when a problem was discovered that would lead to the introduction of air into the suction of the CA pumps. The event scenario is a dual unit Loss Of Offsite Power (LOOP) coincident with a steam line or feedwater line break on one unit occurring prior to transfer of suction for the CA pumps to the safety grade assured source, Nuclear Service Water (RN).

Root Cause: This event is attributed to an unanticipated interaction of systems or components. The limiting design criteria, for ensuring the CA pumps receive a constant water supply, was assumed to be the ability of the CA System to survive a break in the nonseismic condensate piping at a point downstream of where the three condensate sources tie together. Because this was the criteria used, no transient analysis of the individual sources was performed.

Corrective Actions: Following the initial evaluation on 5/15/97, the CACST was isolated pending further evaluation. Engineering, with off-site assistance will continue evaluation of the condensate quality suction sources issue to ensure that all possible configuration concerns have been identified and evaluated, and that all necessary corrective actions have been taken.

BACKGROUND

There are three nonsafety grade, condensate [EIIS:KA] quality water supplies for the Auxiliary Feedwater [EIIS:BA](CA) System. These sources are:

- The Auxiliary Feedwater Condensate Storage Tank [EIIS:TK] (CACST), which is shared by unit 1 and unit 2,
- Each unit's Upper Surge Tank (UST), and
- Each unit's Condenser [EIIS:COND] Hotwell.

The three sources are normally aligned to the common suction header of all three CA Pumps [EIIS:P]. Each of the three sources is capable of being isolated from the common suction header by a motor operated isolation valve [EIIS:20]. Check valves in series with the respective isolation valves prevent volume exchange between the three sources, each of which is providing different head pressures at their check valve's location. It is this difference in head pressures that accounts for the concept of "preferred suction source usage". The order in which the CA pumps are supplied condensate quality water is: the CACST, the UST, and finally, the Condenser Hotwell. When the CACST decreases to a predetermined level, isolation valve 1(2)CA6 closes automatically and the UST begins to supply the CA pumps.

The CA System utilizes the Nuclear Service Water System [EIIS:BI](RN) as the assured, safety grade water supply for the CA Pumps. In order to reduce the potential for adverse chemistry effects, the noncondensate quality RN System is normally isolated from the CA System, and is automatically aligned upon detection of low-low CA pump suction pressure during CA autostart operation.

Because the condensate quality sources are nonsafety grade, the limiting design criteria, for ensuring the CA pumps receive a constant water supply, was assumed to be the ability of the CA System to survive a break in the nonseismic condensate piping at a point downstream of where the three condensate sources tie together. Because this was the criteria used to confirm the CA System to RN System transfer scheme adequacy, no transient analysis of the individual condensate sources was performed.

EVENT DESCRIPTION

On 5/8/97, Catawba Nuclear Station's (CNS) engineering received information from McGuire Nuclear Station (MNS), another Westinghouse PWR, also operated by Duke Power Company, concerning potential vortex formation at the outlet of the CACST. This vortex could form during accident conditions that result in high CA System flow demands on the CACST, potentially from both units simultaneously. The vortex would form at the shared tank outlet, upstream of where this pipe branches into individual lines for each unit. Vortex formation could lead to air bubbles being entrained in the water flow to the CA pumps. Air bubbles reaching the CA pumps could disable the pumps.

Following an initial review of the system design, CNS engineering determined that a similar susceptibility could exist at Catawba, not only for the CACST, but also for the UST. Upon further evaluation, CNS engineering discovered that the complete depletion of the CACST could potentially cause air introduction into the CA pump suction with the UST supplying CA flow if the CACST is not isolated. This is independent of any vortexing issues. Therefore the following three issues were evaluated for potential operability concerns:

- CACST vortexing
- UST vortexing
- CACST depletion

On 5/15/97, engineering apprised site management of the analyses performed and the computer flow model runs conducted over the previous several days. The results of the CACST and UST vortexing issues indicated that neither were an operability concern. CNS Engineering realized that evaluation of the potential introduction of air into the CA pump suction piping from a depleted CACST with the UST supplying the CA flow would require off-site assistance. The potential problem associated with CACST depletion was eliminated by closing 1CA-6 and 2CA-6.

Later in the day, this event was reported in accordance with the criteria of 10 CFR50.72(b)(2)(iii)(D), "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident."

CONCLUSION

The root cause of this event is attributed to an unanticipated interaction of systems or components. Because the condensate quality sources are nonsafety grade, the limiting design criteria, for ensuring the CA pumps receive a constant water supply, was assumed to be the ability of the CA System to survive a break in the nonseismic condensate piping at a point downstream of where the three condensate sources tie together. Because this was the criteria used to confirm the CA System to RN System transfer scheme adequacy, no transient analysis of the individual condensate sources was performed.

The results of the CNS engineering calculations proved that vortexing in the outlet of either the CACST or UST is not an operability concern. The only operability concern identified is the potential for air induction into the suction of the CA pumps as a result of complete depletion of the CACST, with failure of the tank to be isolated. CNS Engineering realized that evaluation of the potential introduction of air into the CA pump suction piping from a depleted CACST with the UST supplying the CA flow would require off-site assistance.

Following subsequent analyses, it was concluded that air could be drawn into the piping junction from the CACST and UST sources. Due to the relatively low suction head provided by the UST under vacuum and the length of piping between the UST and the junction of the UST and CACST piping, there are certain CA flow rates for which a negative pressure will exist at this junction causing air entrainment. At low flow rates, this negative pressure does not occur because the line losses from the UST are not sufficiently high enough. At high flow rates, the autoswap to RN will occur because the line losses to the pumps are high enough to reduce the suction pressure to the RN autoswap setpoint. However, for CA flow rates between 1300 and 2200 gpm, the junction pressure is negative, but RN autoswap does not occur.

A review of reportable events for the 24 months prior to this event indicates that there have been 2 events involving design analysis issues. LER 413/96-004 reported an event whereby a wrong assumption was used to determine the volume of water available for the Turbine Driven CA Pump during security events. LER 413/96-012 reported an event whereby the failure to open of one of the two parallel assured source supply valves could result in inadequate net positive suction head for the CA Pumps. Design analysis issues for the CA System have already been identified as recurring. A Planned Corrective Action in LER 413/96-012 Rev.01 as well as the planned

corrective action in this LER are expected to prevent recurrence. Each of the previous reportable events was self identified and did not involve an actual operational occurrence.

CORRECTIVE ACTIONS

IMMEDIATE

NONE

SUBSEQUENT

1) Closed 1CA6 and 2CA6 as a conservative measure to ensure operability pending the results of the subsequent engineering evaluation into the issue of CACST depletion.

PLANNED

1) A reconstitution of the electrical and mechanical system design basis for the SSS interfacing systems will be conducted, and a consolidated Design Basis Document specific to the SSS will be generated. This is a Planned Corrective Action committed to in LER 413/96-012 Rev.01.

2) CNS engineering, with off-site assistance will continue evaluation into the condensate quality suction sources issue to ensure that all possible configuration concerns have been identified and evaluated, and that all necessary corrective actions have been taken.

SAFETY ANALYSIS

Events involving the loss of a steam generator [EIIS:SG](S/G) secondary side pressure boundary such as a Steamline Break or Feedwater Line Break resulting in a significant decrease in S/G pressures are the events that result in the highest CA flow rates. Also in a Loss Of Offsite Power (LOOP) event, a significant S/G pressure decrease can occur, due in part to the introduction of the relatively cold CA water into the S/G's. If the CA System flow rates are throttled sufficiently prior to CACST depletion, a negative pressure at the CACST / UST junction will not exist. Therefore limiting events, from a tank depletion time, are those that result in high

CA flow rates. The most limiting event would be a dual unit LOOP with a Steam Line or Feedwater Line Break on one unit.
On a dual unit event, the CACST could be depleted in 7-8 minutes assuming all three CA pumps on each unit are operating at maximum Accident Analysis flow rates. If condenser vacuum is lost prior to CACST depletion, the UST would become the primary supply source for the CA System, thereby delaying CACST depletion until well into an event at which time CA flow rates would be more than sufficiently throttled.

Isolation valve 1(2)CA-6 has an automatic function to isolate the CACST from the CA suction piping on low CACST level. This valve is not powered from a safety-related source, but does receive power from an "A" Train Blackout load center. If the "A" Train diesel generator [EIIS:DG](D/G) does not start, the valve will not be powered. However, in this case the "A" Train CA Pump would not start either. Therefore the depletion time of the CACST would be extended. The circuitry associated with the automatic closure of 1(2)CA-6 is battery-backed with a battery lifetime in excess of two hours, which is well into an event at which time CA flow rates would be throttled. If however the circuitry should lose power, the circuit fails in such a manner as to provide for indication of a low tank level, which would initiate valve closure. The CACST level transmitter is pneumatically controlled and will fail to a low level indication on loss of air supply, thereby providing a signal to close 1(2)CA-6. Valves 1(2)CA-6 are cycled once a month for CA pump testing. A check of work history revealed no problems with the valve on either unit in the past 10 years. Therefore the valves are proven reliable.

The Severe Accident Analysis Group reviewed the CACST air entrainment problem to determine its impact on plant core damage risk. The frequency of accident sequences involving failure of the CA system due to air entrainment from the CACST was estimated. This value was compared to the Catawba core damage frequency for determining the risk significance of this event. Of particular interest were two-unit LOOP events and single-unit LOOP events. The estimate conservatively assumed that an air entrainment event will always occur when CA flow is demanded and will always cause all CA pumps to fail. This review found that the problem resulted in an estimated increase in core damage frequency of 7E-07 per year, representing approximately a 1 percent increase in the annual core damage frequency.

Based on the above discussions, it is unlikely that a loss of CA function will occur due to air entrainment from a depleted and unisolated CACST. However, if the CA function were lost, Control Room Operators would enter procedure EP/1(2)/A/5000/FR-H.1 "Response to Loss of Secondary Heat Sink."

If offsite power is available, the procedure establishes non-safety feedwater to provide the secondary heat sink. If feedwater can not be established from any source, a feed & bleed of the Reactor Coolant [EIIS:AB](NC)System would be initiated. The feed and bleed process involves initiation of Safety Injection and opening the pressurizer Power Operated Relief Valves. The process is sufficient to maintain core heat removal and prevent fuel damage.

In consideration of the negligible increase in overall plant risk, the health and safety of the public would not have been affected by this event.