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DUKE POWER

April 27, 992

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

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/90-001, Revision 1

Gentlemen:

Attached is Licensee Event Report 413/90-001, Revision 1 concerning PRESSURIZER SAFETY VALVE BLOWDOWN INCONSISTENT WITH DESIGN ANALYSES AND GREATER THAN MANUFACTURER'S RATING.

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

Very truly yours,

W. R. McCollum, Jr.
Station Manager

Attachment

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NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CATAWBA NUCLEAR STATION, UNIT 1	0 5 0 0 0 4 1 3	1 OF 0 7

TITLE (4) PRESSURIZER SAFETY VALVE BLOWDOWN INCONSISTENT WITH DESIGN ANALYSES AND GREATER THAN MANUFACTURER'S RATING

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 1	1 6	8 9	9 0	0 0 1	0 1	0 4	2 7	9 2	CATAWBA, UNIT 2	0 5 0 0 0 4 1 4
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X 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)	Courtesy LER Part 21
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	
ROBERT FUTRELL, COMPLIANCE MANAGER	8 0 3	8 3 1 - 3 6 6 5

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
B	A B	V	D 2 4 3	Y					

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)

On November 16, 1989, a potential concern was identified with pressurizer safety valve blowdown being greater than assumed in the safety analyses. Unit 1 was in Mode 1, Power Operations, at 100% power and Unit 2 was in Mode 1, Power Operations, at 97% power at the time. Review of FSAR Chapter 15 feedwater line break analyses indicated that the pressurizer safety valves may not reseat as assumed due to high Reactor Coolant System temperatures. The vendor's analyses did not properly include consideration of the safety valve's blowdown. Whereas the manufacturer's rated blowdown was approximately 5%, testing indicated blowdown of 10-12% occurs with current valve settings. Evaluation of the effects of increased blowdown concluded that no degradation of overpressure protection resulted. Further, none of the conclusions of the FSAR Chapter 15 analyses were invalidated and no Technical Specification limits were violated. Reanalysis of the feedwater line break accident was performed with acceptable results. This report is provided as a Courtesy LER with respect to the inconsistencies with the FSAR analyses and pursuant to 10CFR Part 21 with respect to actual valve blowdown in excess of the value rated by the manufacturer.

BACKGROUND

The Reactor Coolant [EIIS:AB] (NC) System consists of four heat transfer loops connected in parallel to the Reactor Vessel [EIIS:VSL]. Each loop contains a Reactor Coolant Pump [EIIS:P] and a Steam Generator [EIIS:HX] (S/G). The B loop also includes a Pressurizer, a Pressurizer Relief Tank (PRT), interconnecting piping and instrumentation necessary for operational control.

NC System pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electric heaters [EIIS:EHTR] and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to reduce pressure variations due to contraction and expansion of the Reactor coolant. Three spring loaded safety valves [EIIS:V] 1(2)NC-1, 2 and 3, are connected to the pressurizer and discharge to the pressurizer relief tank.

The three pressurizer safety valves are of the totally enclosed pop-type. The valves are manufactured by Dresser, Model 6-31749A-2-XNC019, and are spring-loaded, self-activated with back pressure compensation. The combined capacity of the valves is equal to, or greater than, the maximum surge rate resulting from complete loss of load without Reactor Trip or any other control. Temperature indicators [EIIS:XI] in the safety valve discharge manifold alert the Operator to the passage of steam due either to leakage or valves lifting.

The Pressurizer is equipped with three Power Operated Relief Valves (PORVs) which limit system pressure for a large power mismatch and thus prevent actuation of the fixed high-pressure Reactor trip. The PORVs are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated valves are provided to isolate the inlet to each PORV if excessive leakage occurs.

The PRT condenses and cools the discharge from the Pressurizer safety and relief valves. Steam is discharged through a sparger pipe [EIIS:PSP] under the water level. The PRT is equipped with an internal spray and a drain which are used to cool the tank following a discharge. The PRT is protected against a discharge exceeding the design value by two rupture discs which discharge into the Reactor Containment.

The Feedwater [EIIS:SJ] System (CF) and the Auxiliary Feedwater [EIIS:BA] System (CA) function to provide feedwater to the S/Gs. The steam Turbine Driven Feedwater Pumps discharge through two stages of high pressure feedwater heaters (A and B) with equalization headers preceding and following the two parallel heater strings. The feedwater then divides into four main feedwater lines, each feeding one of the four S/Gs. A tempering flow line also originates from the feedwater equalization header. This line splits into four lines, each connecting to one of the S/G auxiliary feedwater nozzles.

Each S/G has a sixteen inch diameter main feedwater nozzle and a six inch diameter auxiliary feedwater nozzle. A line connects each S/G main feedwater line to its corresponding auxiliary feedwater nozzle. Feedwater flow is normally delivered by the main feedwater lines to the main feedwater nozzles. During some modes of operation, feedwater is delivered to the auxiliary feedwater nozzles.

EVENT DESCRIPTION

Recognition of this issue arose as part of a Design Engineering (DE) evaluation of the performance of the Pressurizer Code Safety Relief Valves (PSVs), which was undertaken due to recent industry problems with the calibration of valve lift setpoints. These problems occur due to loop seal conditions differing between calibration and operation. In order to obtain stable PSV performance (minimize the potential for valve chatter) the desired ring settings may result in blowdowns in excess of original specification values. Blowdown is defined as the difference in PSV lift and reseat pressures, divided by the lift pressure, expressed as a percentage. Whereas the expected blowdown (rated by the manufacturer) was approximately 5%, the results of Electric Power Research Institute (EPRI) tests and Duke Power tests showed that blowdown up to 10-12% occurs with current valve settings; the current valve settings are sufficient to prevent valve chatter. Based on this information, an evaluation of what blowdown is acceptable was requested of DE.

Based on knowledge of FSAR Chapter 15, the feedwater line break (FWLB) is recognized as the transient that presents the greatest challenge to the PSVs, with respect to NC temperatures, (Section 15.2.8). A FWLB evolves into an overheating event, which, along with safety injection, causes the NC to go water solid. NC heats up until auxiliary feedwater flow matches and then exceeds decay heat. The vendor, Westinghouse, has adopted an acceptance criterion which states that the transient response is acceptable as long as NC remains subcooled. The FWLB is analyzed with assumptions and modeling which conservatively minimize the available heat sink. Neither the PSVs nor their blowdown is explicitly modeled in the analysis. PSV lifting and reseating is modeled as coolant inventory loss initiated when primary pressure exceeds the PSV lift setpoint (+1 percent) and terminated when primary pressure falls below the PSV lift setpoint. NC overpressure protection is the primary safety concern. Core cooling is demonstrated by ensuring that NC remains subcooled. The PSVs provide overpressure protection since credit is not taken for the PORVs.

A preliminary evaluation of what blowdown was acceptable concluded that the reseat pressure should be greater than the saturation pressure at the time of the hottest NC temperature during the FWLB. This would preserve the Westinghouse criterion of NC remaining subcooled. A review of the most recent

FWLB analyses for Catawba (to be submitted in the next FSAR update) indicates that NC temperatures result in saturation pressures greater than the reseat pressure for both the expected 10-12% blowdown, and for the originally specified 5% blowdown. Recognition of this fact prompted the generation of PIR 0-C89-0352 on November 16, 1989. Unit 1 was in Mode 1, Power Operation, at 100% power and Unit 2 was in Mode 1, Power Operation, at 97% power, at the time the PIR was generated. This issue is applicable to each Unit from initial startup; both Units have operated in all modes and at all power levels since that time.

An operability evaluation completed on November 17, 1989, concluded that the PSVs were operable as an extension of the time required for them to reseat would not degrade their ability to preclude primary system overpressurization. Further, it was determined that none of the conclusions of the FSAR Chapter 15 analyses were invalidated and no Technical Specification limits were violated.

As NC pressure increases following a FWLB, the PSVs are challenged initially by steam and then later by liquid. The valve will lift at approximately 2500 psig and cause NC pressure to decrease. Provided that the reseat pressure is greater than the NC saturation pressure, the valve will reseat. NC will then repressurize and a cyclic valve response will occur until the cause of continued pressurization is terminated. If, however, high NC temperatures exist, the depressurization will stop when the NC saturation pressure is reached. If the saturation pressure is above the PSV reseat pressure, then the valve will not reseat and sustained relief will occur. Eventually NC temperatures will decrease, NC saturation pressure will decrease, and the valve will reseat. If excessive NC inventory addition persists, due to continued safety injection, additional valve cycling will occur, but without continuous relief.

CONCLUSION

This incident is attributed to a functional design deficiency in that the vendor's analysis did not properly account for PSV blowdown. A contributing functional design deficiency is attributed to the fact that blowdown of 10-12% occurs with current valve settings, exceeding the valve manufacturer's rated blowdown of 5%. It was concluded that the only potential adverse consequences of extended PSV blowdown are the effects of an increase in NC inventory loss during the reference feedwater line break accident.

The increased blowdown and potential increase in NC inventory loss are only a problem if NC temperatures are high following the FWLB. A review of the FSAR analyses (Section 15.2.8) has identified conservative modeling and assumptions which result in excessively high NC temperatures. A major assumption is that the auxiliary feedwater flow to the S/G with the feedwater line break is lost.

In reality, the auxiliary feedwater is delivered until manually isolated by the Operator and provides a substantial heat sink. Upon isolating the affected S/G, auxiliary feedwater flow increases to the intact S/Gs. Another method of reducing NC temperatures is to stop the Reactor Coolant Pumps. There is a significant benefit in reducing pump heat input to the NC, as is evident when the results of the FSAR FWLB analyses with and without offsite power are compared (Section 15.2.8.2). Operator action to increase auxiliary feedwater flow is also available.

The FSAR FWLB accident was re-analyzed with less conservative assumptions and increased PSV blowdown. The results of this analysis show that saturation conditions are not reached in the hot legs and flashing does not occur. Therefore, the PSV would cycle normally with no failure to reseat. No changes to guidance for operator action were found to be necessary.

This issue is reported as a Courtesy LER with respect to the inconsistency between the FSAR analyses and expected PSV performance during the postulated FWLB accident. Pursuant to 10CFR Part 21, the PSVs are considered a "basic component" as defined therein; blowdown performance in excess of the manufacturer's rating constitutes a "defect"; existence of a "substantial safety hazard" will be dependent upon the licensing basis analyses applicable to each plant.

Inconsistencies between design bases/analyses and actual plant operating characteristics is a previously recognized recurring event/problem. In response, Duke Power Company has initiated a comprehensive, long-term Design Basis Documentation improvement program. The results of this effort will significantly improve understanding of both assumed and actual system/equipment operating characteristics and will help identify and eliminate inconsistencies.

Deficiencies in valve manufacturer analyses of valve operating characteristics is also a previously recognized recurring event/problem. Recent Catawba LER (413/89-029) and internal report (2-C88-0143) have documented incidents in which valve operators have been unable to open/close the valve because actual friction forces and/or valve factors exceeded values assumed in determining operator output requirements. NRC IE Bulletin 85-03 and Generic Letter 89-10 also deal with this issue. Catawba Nuclear Station has an on-going program to ensure valves will function properly under all anticipated conditions. It should be noted that this program is primarily focused on the proper interaction of valves and valve operators. The current event did not involve a valve operator nor was the pressurizer safety valve's overpressure protection function impaired. Thus, these events are not exactly similar.

This incident is NPRDS reportable because the PSVs were shown to operate outside the manufacturer's stated range.

CORRECTIVE ACTION

SUBSEQUENT

1) An operability evaluation for extended blowdown of pressurizer safety valves was performed, concluding that the valves are operable.

2) The FSAR feedwater line break accident was reanalyzed with less conservative assumptions.

SAFETY ANALYSIS

The issue of sustained water relief through the PSVs following a FWLB was evaluated to determine any unacceptable consequences. The reference transient response is the latest FSAR FWLB analysis. The new situation to be evaluated is sustained water relief, for an intermediate period of time, through the PSVs.

The reference analysis does not explicitly model PSV behavior in terms of blowdown. Westinghouse holds NC pressure at the PSV lift setpoint, and whatever water relief results from NC inventory expansion is relieved. This approach is conservative based on the analysis objective of maximizing NC temperature and pressure. With modeling of cyclic PSV behavior, additional safety injection flow and NC cooling would result. If blowdown was explicitly modeled, it is expected that additional water relief will result. This aspect is addressed further below.

The impact of increased blowdown on the overpressure concern related to FWLB is non-existent. The effect of blowdown is to decrease pressure.

The impact of increased blowdown on the integrity of the PSV is judged to be a net benefit. Increased blowdown will result in fewer valve cycles, and valve duty increases with the number of cycles.

It is concluded that the only potential adverse effects due to PSV blowdown are related to the consequences of an increase in NC inventory loss. Core cooling, doses, and containment response effects have been evaluated.

The containment response to any transient which does not release sufficient mass and energy to deplete the ice inventory is bounded by a LOCA. Therefore, there is no impact on the design basis containment response.

The dose consequences for the reference FWLB are considered to be bounded by the steam line break. The FWLB consequences should be only minimally impacted by increased NC inventory loss into containment, since the dominant source term is the pre-existing contamination of the S/G secondary inventory caused by the 1 gpm Tech Spec leakage. This source term is unaffected by the increase in NC inventory loss to containment.

Core cooling is not challenged by the increased NC inventory loss. The three intact S/Gs are being supplied with greater than 450 gpm of auxiliary feedwater. This heat sink is mainly what stabilizes NC temperature in the reference analysis. As decay heat decreases, NC temperatures decrease. This heat sink is unaffected by the blowdown issue. Furthermore, should the NC reach saturation, the latent heat of boiling the NC inventory is a substantial energy relief process that is unavailable during subcooled conditions. A transition to steam relief must occur prior to core uncovering. High-head charging pumps are continuously delivering makeup during the transient. An additional consideration is the expectation that the pressurizer PORVs, not the PSVs, will in reality be the relief path should the scenario of concern actually occur. (The PORVs are not proven to be environmentally qualified for the FWLB conditions, and are therefore not taken credit for in the reference analyses. The PORVs are qualified for low temperature overpressure protection conditions.)

The combination of these mitigating processes ensures that core cooling is not impacted by increased PSV blowdown. None of the conclusions of the FSAR Chapter 15 analyses are invalidated and no Technical Specification Limits are violated. Thus, it is concluded that the health and safety of the public were not affected by this incident.