AEOD ENGINEERING EVALUATION REPORT*

UNIT: Calvert Cliffs 2
DOCKET NO.: 50-318
LICENSEE: Baltimore Gas & Electric
NSSS/AE: Combustion Engineering/Bechtel
EVENT DATE: October 11, 1983

EE REPORT NO.: AEOD/E415
DATE: June 6,1984
EVALUATOR: E. Imbro

SUBJECT: OVERCOOLING TRANSIENT

Evaluation Summary

Although the safety significance of this event is small, it is an interesting event to document since it contained five independent failures:

(1) The No. 22 main feedwater pump tripped;
(2) The No. 21 feedwater regulating valve failed to close;
(3) The No. 21 main feedwater pump speed controller stuck in the high speed position;
(4) A turbine bypass valve failed in the 50% open position; and
(5) A reactor coolant pump vapor seal failed 1-1/2 hours after the reactor trip.

In addition, the pressurizer pressure behavior during the pressurizer refill demonstrated an interesting phenomenon that can result when the liquid and vapor phases are not in thermodynamic equilibrium. This phenomenon, that can occur when recovering from a transient in which the pressurizer is nearly drained, can result in a temporary decrease in pressure after the level has been returned to normal and all the heaters are on. This pressure response is contrary to what one would normally expect and could be initially puzzling to plant operators.

This event, occurring in 1983, was reported under the old LER system. The attached LER submitted by the licensee, although in accordance with the requirements of the time, lacks much of the detail that would have been reported under the current LER reporting requirements. It does not provide enough information for one to get a complete picture of the event. Reviewing this LER against the account of the actual event clearly demonstrates the benefit of the new reporting requirements.

* This document supports ongoing AEOD and NRC activities and does not represent the position or requirements of the responsible NRC program office.

Discussion:

On October 11, 1983, Unit 2's No. 22 main feedwater pump tripped due to a leak in the pump's hydraulically operated control system. The feedwater pump trip occurred at 11:36:45 AM while the plant was operating at 100% power. In an attempt to avert a plant trip the operators began reducing power by borating and inserting control rods. The feed-flow/steam-flow mismatch was too great and the plant tripped from 83% power at 11:37:35 AM. Following the plant trip, the No. 21 feedwater regulating valve failed to close. This caused the No. 21 feedwater pump's speed to increase, resulting in a rapid rise in the No. 21 steam generator's level. The operator's attempt to decrease feedwater flow by placing the No. 21 main feedwater pump controller in manual and trying to decrease pump speed failed because the pump speed controller had stuck in the high speed position due to an accumulation of dirt in the mechanism. Feedwater flow was isolated to SG 21, approximately three minutes after the reactor trip, at 11:40:45 when the operators tripped the No. 21 main feedwater pump and shut the main feedwater isolation valve. The excessive feeding of SG No. 21 caused the reactor coolant system pressure to drop sharply due to the overcooling which resulted in a shrinkage of the primary coolant. The reactor coolant temperature dropped approximately 50°F in three minutes and the pressure decreased to approximately 1660 psig causing a safety injection actuation. No water was actually injected, since this pressure is above the shutoff head of the high pressure safety injection pumps.

The severity of the overcooling transient was heightened by the effect of a turbine bypass valve that stuck 50% open due to mechanical binding. The turbine bypass valve, which was passing about 4 to 5% of full power steam flow, was manually isolated in the timeframe of 11:41 AM to 11:42 AM. An additional turbine bypass valve was thought to have failed open but this turned out to be an erroneous indication due to an improperly set limit switch.

Following a turbine trip the feedwater supply to the steam generators is designed to ramp down automatically to 5% of full power flow in 20 seconds. This is accomplished by automatic closure of the main feedwater regulating valves on a turbine trip and opening of the feedwater bypass valves to a predetermined position to maintain 5% of normal full power feedwater flow. The speed of the turbine-driven main feedwater pumps is controlled to maintain a constant differential pressure across the main feedwater regulating valves. Therefore, the failure of the No. 21 main feedwater regulating valve to close on a turbine trip and the opening of the bypass feedwater regulating valves caused a decrease in the differential pressure across the main feedwater regulating valves which resulted in the speed increase of the No. 21 main feedwater pump. As previously mentioned, this caused the rapid rise in the No. 21 steam generator's level. The level in SG No. 21, as indicated on the narrow-range level instrumentation, went from its minimum level of -63 inches (Note 1) to offscale high, which is +63.5 inches, in

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(1) This is referenced to zero level which is the normal operating level and is 47 inches above the centerline of the feedring.

about 75 seconds. Steam generator No.21's level remained offscale high for approximately seven minutes. The licensee's estimate is that the level increased only about 10 inches above the highest indicated level or to about +73.5 inches. The level was estimated by recording the blowdown flowrate with the generator isolated and noting how long it took for the level to come onscale. The level in the No. 22 SG dropped to a low of -178 inches, as indicated on the wide-range level instrumentation (Note 2), which automatically started the motor-driven auxiliary feedwater (AFW) pump. The level remained at -178 inches for less than 20 seconds before increasing. This was not long enough to start the turbine-driven AFW pumps. Level on the No. 22 SG recovered to an indicated -112 inches two minutes after the initiation of AFW. For purposes of reference, the steam generator tubes begin to uncover at the -59 inch level.

Twenty minutes after the reactor trip, reactor coolant pump 21A was tripped to reduce the pressurizer spray flow in response to what was initially thought to be a stuck open spray valve. This was done when plant operators observed the pressurizer pressure decrease from 1985 psig to 1854 psig over the span of 15 minutes. Further investigation revealed the spray valves were not stuck open. The pressurizer pressure decrease was subsequently attributed to an inflow of relatively cool water from the hot leg as the pressurizer level was being restored. This pressurizer pressure decrease which led the operators to believe initially a spray valve was stuck open was an interesting demonstration of a facet of non-equilibrium pressurizer thermodynamic behavior, i.e., the liquid and vapor phases are at different temperatures.

Since the water is entering the pressurizer from the surge line at the hot leg temperature, approximately 530°F in this case, the liquid and vapor phase in the pressurizer are no longer in equilibrium, i.e., the water is subcooled. (Saturation temperature at 1660 psig is 609°F.) During the overcooling transient the pressurizer level had dropped from the full power level of approximately 200 inches to 18.9 inches, well below the low level heater cutoff point of 101 inches. Since the lower pressurizer level instrumentation tap is located near the bottom of the lower head, the 18.9 inch pressurizer level attained during the cooldown transient indicates the pressurizer was nearly drained. Level was restored to the 101 inch level at 11:50 AM and the heaters were reenergized. At 11:59 AM the pressurizer level was stabilized at 145 inches, some 30 seconds later the operators noticed the pressurizer pressure begin to decrease from its value of 1985 psig.

The unavailability of the heaters in the initial phase of the pressurizer refill indicates that the initial phase of the pressure recovery, which was observed to be normal, was due to the compression of the steam bubble in the pressurizer. At the time the pressurizer reached the 145 inch level where it was stabilized by the operators, the heaters had only been energized for 9-1/2 minutes, not a long enough time to get the liquid temperature in equilibrium

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(2) The lower level tap of this instrument is located in a region of high flow velocity, resulting in a localized depression of the static pressure at the tap. Therefore, at normal operating conditions the indicated reading is about 50 inches less than the actual level, i.e., an indicated -178 inches corresponds to an actual level of -128 inches.

with the vapor temperature. If we assume the vapor is saturated (Note 3) this would have been at 636°F, the saturation temperature corresponding to 2000 psia. A quick calculation indicates that during this 9-1/2 minute interval of increasing level, the temperature of the liquid would increase about 40°F or to about 570°F, assuming all 1500 KW of pressurizer heaters in operation. Therefore, after the pressurizer level had stabilized at 145 inches the vapor could lose heat both to the liquid and to the pressurizer walls. Again, assuming a saturated vapor phase, the observed pressure decrease from 1985 psig to 1854 psig over a 15 minute period corresponds to only a 10°F decrease of the vapor temperature, from 636°F to 626°F. Calculations show it would take on the order of 15 minutes to heat the volume of water in the pressurizer from 570°F to 626°F. This correlates with the 15 minutes observed by the operators for the pressure to stabilize at 1854 psig. At this time the liquid/vapor equilibrium was reestablished. As the liquid was further heated pressure began increasing. The pressure was subsequently returned to normal without further incident. This pressurizer behavior, although not unusual, is not what one would normally expect. The "instinctive" reaction would be: "All the heaters are on, the level is stable, pressure has been increasing continuously during pressurizer refill; pressure should continue increasing not begin decreasing."

Reactor coolant pump 21A was restarted after being idle for 1 to 1-1/2 hours. Following pump restart, an increase in the drainage frequency of the containment sump was noted. The cause of the leakage into containment was discovered to be a failed vapor seal on the reactor coolant pump that had been stopped. The licensee attributed this to normal end-of-life failure. This seal had been in service for 29 months and the average pump seal life at Calvert Cliffs has been shown by experience to be on the order of 23 months. The licensee did not consider the seal failure to be related to the pumps being idle for 1 to 1-1/2 hours and then being restarted, as this is routinely done during plant startups and shutdowns without adverse effect.

Findings

If one were trying to prove the existence of "Murphy's Law" using a process of inductive reasoning, this event, containing five independent failures, would be a good example to use. In spite of the sequential string of failures, the operators did a good job in handling the event and at no time were any of the T/S safety limits violated. This is an interesting event to document, particularly in view of the licensee-submitted Licensee Event Report, 83-054/03L which is attached for information. The LER which was submitted under the old reporting requirements does not discuss the overcooling transient, the safety injection actuation, the stuck open turbine bypass valve or the reactor coolant pump vapor seal failure, or the non-equilibrium pressurizer behavior.

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(3) If the compression of the initially saturated vapor by the rising liquid surface was isentropic, the vapor would be superheated. Since there are irreversibilities in the process, namely heat transfer from the vapor, the compression is not isentropic. However, depending on the magnitude of the irreversibilities during compression, the vapor still may be slightly superheated.

Conclusions

In reviewing events such as this, the value of the new LER rule, effective as of January 1, 1984, is readily apparent. Had the event occurred after January 1st of this year, the event would have, of necessity, been better documented by the licensee in the LER. In actual fact, the real safety significance of this event is small. The significant point highlighted by this report is how much valuable operating data was not captured under the former LER system.

Since this was a rather interesting event that contained a number of independent failures in addition to an example of non-equilibrium pressurizer behavior, I would recommend it be considered for inclusion in Power Reactor Events

(ATTACHMENT)

NRC FORM 366 (12-81)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

APPROVED BY OMB 3150-0011 EXPIRES 4-30-82

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | M D C C N 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

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01 | REPORT SOURCE L (6) | 0 5 0 0 0 3 1 8 (7) | 1 0 1 1 8 3 (8) | 1 1 1 0 8 3 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 At 1138 following a reactor trip, 21 Main Feedwater Regulating Valve
03 failed to shut and feed flow did not decrease to 5% within 20 seconds
04 as required by Technical Specification 3.3.2.1. The operator tripped
05 the feed pump and shut the Main Feedwater Isolation Valve, terminating
06 the event.
07 Similar events: none.
08

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
09	C H (11)	E (12)	B (13)	R E L A Y X (14)	X (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 3	—	0 5 4	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER (26)
A (18)	A (19)	A (20)	C (21)	0 1 9 4	Y (23)	N (24)	A (25)	F 1 3 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The cause of the failure of the 21 Main Feedwater Regulating Valve to
11 shut was found to be a failed horizontal relay (Fisher Controls,
12 Pneumatic, P/N AJ6206000A) in the valve positioner. The failed relay
13 was replaced and the valve tested satisfactorily. No further action
14 is required.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
15	G (28)	0 0 0 (29)	N/A	A (31)	Operator Observation

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
16	Z (33)	Z (34)	N/A	N/A

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
17	0 0 0 (37)	Z (38)	N/A

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
18	0 0 0 (40)	N/A

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LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
19	Z (42)	N/A

IE22

PUBLICITY

	ISSUED	DESCRIPTION (45)
20	N (44)	N/A

NRC USE ONLY

NAME OF PREPARER M. A. Junge/R. Androsik PHONE: 301-269-4969/4986

LER NO. 83-54/3L
DOCKET NO. 50-318
LICENSE NO. DPR 69
EVENT DATE 10-11-83
REPORT DATE 11-10-83
ATTACHMENT

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (CONT'D)

At 1138 following a reactor trip on low Steam Generator level due to a loss of 22 Steam Generator Feed Pump, 21 Main Feedwater Regulating Valve failed to shut and feed flow did not decrease to 5% within 20 seconds as required by Technical Specification 3.3.2.1. With 21 Steam Generator Feed Pump in automatic and 21 Main Feedwater Regulating Valve open, the pressure drop across 21 Main Feedwater Regulating Valve was low causing 21 Steam Generator Feed Pump to increase speed resulting in a rapid filling of 21 Steam Generator. The operator placed the feed pump in manual and tried to decrease speed. Speed did not decrease and the operator tripped the feed pump and shut the Main Feedwater Isolation Valve, terminating the event. Similar events: none.