CENTERIOR ENERGY

PERRY NUCLEAR POW...
10 CENTER ROAD
PERRY, OHIO 44081
(216) 259-3737

Mail Address:
P.O. BOX 97
PERRY, OHIO 44081

Michael D. Lyster
VICE PRESIDENT - NUCLEAR

March 20, 1991
PY-CEI/NRR-1338 L

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

Perry Nuclear Power Plant
Docket No. 50-440
Reactor Vessel Overfill Protection
Generic Letter 89-19

Gentlemen:

Our response to Generic Letter 89-19 was provided by letter PY-CEI/NRR-1171L, dated May 4, 1990. In January 1991 we experienced a loss of automatic feedwater trip capability relevant to the concerns expressed in the Generic Letter. Also, in reviewing our May 4, 1990 evaluation, the need for changes to the design description was identified (side bars in the attachment indicate changes). These changes include descriptions of the power supplies for the relay trip circuits for the feedpump and main turbine trips, and a brief description of a related design change implemented as a result of the January 1991 event.

The attached evaluation continues to support our findings that (1) PNPP susceptibility to high water level trip failure compares favorably to the reference BWR analyzed by the NRC in NUREG/CR-4387, (2) PNPP susceptibility to main steam line break also compares favorably, and (3) offsite dose consequences, if main steam line failure and fuel damage were to occur, are reduced from the reference case. Consistent with conclusions reached for the reference BWR in the NRC evaluations, and the BWR Owners Group in their generic evaluation of BWR overfill protection, we still conclude that additional PNPP modifications are not cost-justified.

If you have any questions, please feel free to call.

Sincerely,

M. D. Lyster

Michael D. Lyster

MDL:WJE:njc

Attachment

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III

Operating Companies
Cleveland Electric Illuminating
Toledo Edison

A001
1/1

PNPP Design Features and Procedures for Reactor Vessel Overfill Protection

Safety Evaluation

The Perry Nuclear Power Plant (PNPP) has automatic reactor vessel overfill protection to mitigate main feedwater overfeed events, initiated on a high reactor vessel water level (Level 8) signal based on 2-out-of 3 logic to trip the 2 turbine and 1 motor driven feedwater pumps. System design and setpoints have been selected to minimize inadvertent trips of the main feedwater system during startup, normal operation and surveillance testing. The design employs signals from three water level legs which are monitored for level in excess of Level 8 by separate alarm units. The alarm units are connected in two out of three trip configuration to provide main turbine trip, feedwater pump trip and control room annunciation. The alarm units have independent power sources and fail in the tripped condition on loss of power. These alarm units feed two-out-of-three relay trip circuits which share a common 125 VDC power source. A concern related to these relay trip circuits having fuse protection common to other cirucits off of this DC power source was recently identified during routine feedpump turbine stop valve testing. As reported in LER 91-004, a design change was implemented to prevent a loss of trip function for the main and feedpump turbines caused by a malfunction of unrelated indication and control circuits.

The PNPP design conforms to "Group I" as defined in Enclosure 2 of the Generic Letter, except that control circuits are not physically separated from trip circuits and other requirements including EQ are not met. Most importantly, the PNPP design is less vulnerable to feed pump trip failures than the BWR analyzed (Reference 1). The dominant failure mechanism, feedwater control failure initiated by instrument failures on a common sensing line, or failure of this sensing line, are not applicable to PNPP since three independent sensing lines are used in our design.

If automatic feed pump trip is lost, a coincident feedwater control system malfunction resulting in a feed flow > steam flow mismatch would have to occur before reactor vessel overfill was possible. In that event, adequate procedures and training are in place for the operator to assume manual feed pump control and maintain acceptable reactor vessel level, as further discussed below.

If automatic feed pump trip is lost, coincident with the controller malfunction described above, and operator actions do not prevent vessel overfill, potential main steam line break (MSLB) susceptibility/consequences are less severe than reported in referenced NRC evaluations for the following reasons:

1. MSLB susceptibility is reduced by scram prior to water entering the steam lines. Reactor scram occurs on level 8 (about 4 feet below main steam nozzles) from Class 1E, 1-out-of-2 taken twice logic circuits. As noted in reference 1 (Section 5.0), cooldown and collapse of steam is the major driving force for water hammer. Reduced steaming rates after scram therefore also reduces the potential severity of water hammer. MSLB susceptibility is also reduced in comparison to the reference case as described below regarding main steam line dead loads.

2. Before any PNPP damage could occur in the overfill scenario, the core would be subcritical and cooled for periods estimated greater than 20 seconds into the event (Reference 2).

3. Use of WASH-1400 release categories, to represent overfill damage consequences, overestimates risk. PNPP has a containment spray system to reduce iodine and particulate source terms. WASH 1400 did not include such a reduction for BWRs.

Generic Letter 89-19 raises the following BWR-relevant concerns regarding consequences of water-filled steam lines:

> **"Reactor vessel ... overfill can affect the safety of the plant in several ways. The more severe scenarios could potentially lead to a steamline break ... The basis for this concern is the following: (1) the increased dead weight and potential seismic loads placed on the main steamline and its supports should the main steamline be flooded; (2) the loads placed on the main steamlines as a result of the potential for rapid collapse of steam voids resulting in water hammer; (3) the potential for secondary safety valves sticking open following discharge of water or two-phase flow; (4) the potential inoperability of the main steamline isolation valves (MSIVs), main turbine stop or bypass valves, feedwater turbine valves... from the effects of water or two-phase flow ...,"**

The PNPP response to these concerns follows:

(1) Dead loads have been analyzed with acceptable results,* but main steam lines filled with water were not analyzed for seismic loads. The cost of reanalysis, and redesign/support modifications if needed, is not justified by the incremental safety benefits derived (cost considerations are further discussed below). We concur with the conclusion in Reference 1 (p. 8.6) that long-term core cooling is not impacted by MSLB and conclude that PNPP is even less vulnerable to that event than represented in Reference 1.

---

* Perry Safety Evaluation Report Section 5.4.2 describes the alternate shutdown cooling mode which fills main steam lines solid to the SRV's which are opened to establish a recirculating coolant path between suppression pool and vessel.

(2) Damage consequences are bounded by discussions above. Long term core cooling is not affected.

(3) This is not a PNPP concern because SRV's are opened intentionally for alternate shutdown cooling.

(4) MSIVs would remain open (without damage) unless an MSLB-related signal was initiated, i.e. following water hammer damage (MSIV's would also close in RUN mode if steam line pressure drops to 807 psig). Reference 1 (Page 11.2) concludes that MSIV reliability would not be affected, and that MSIV's may seat better if closed following pipe break due to hydraulic forces. Other valve damage consequences described by this concern are bounded by discussions above.

We conclude that (1) the worst consequences of an overfill event described in the Generic Letter and its references does not degrade shutdown cooling capability previously described, analyzed and licensed for PNPP, and (2) PNPP dose consequences from overfill are less than reported in the Generic Letter references. Because of overriding cost/benefit arguments (below) that conclude plant changes are not cost justified, reduced dose consequences at Perry have not been quantified.

Operating Procedures

The Generic Letter has requested that plant procedures and technical specifications include provisions to verify periodically the operability of the overfill protection system and to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. In addition the letter requested that all BWR's reassess and modify, if needed, operating procedures and operator training to assure that the operators can mitigate reactor vessel overfill events that may occur via the condensate booster pumps at reduced pressure.

Regarding surveillance testing to verify operability of automatic overfill protection, previously approved PNPP Technical Specification 4.3.9.2 and Table 4.3.9.1-1.2(a) require periodic channel check, channel calibration, and channel functional testing. Corresponding surveillance instructions include setpoint verification. The Limiting Condition for Operation is 3 channels operable in Mode 1.

Regarding operator mitigation of overfill events while running condensate booster pumps at reduced pressure, PNPP operating instructions utilized at low pressure (discharge pressure of the booster pumps is approximately 350 psig) establish a reactor vessel level band. Instructions that utilize booster pumps also direct use of the Low Flow Controller to automatically or manually control level during plant startup or shutdown operations at power levels below 2 to 3%. Licensed operators are trained to these procedures on a plant specific simulator. In automatic, the controller utilizes a level error signal derived from feedwater flow and a tape set value determined by the operator. In manual the operator directly manipulates the control valve position using OPEN-CLOSE pushbuttons mounted on the controller face, and booster pumps are controlled separately to maintain flow to the reactor vessel. With the controller in manual there is adequate time to respond to undesired increases in reactor vessel level.

Cost/Benefit

NUREG 1218 (Reference 3) evaluates the cost and safety benefit of design upgrades for automatic overfill protection. The only upgrade identified as cost-justified was installation of a single channel feedwater pump trip system at a plant with no existing automatic trip. Table 10.3, Reference 3 shows that other evaluated design changes for the reference BWR are not cost-justified at the $1000/averted man-rem level.

NUREG 1218 further concludes that "although some safety benefit could be gained by providing additional reactor vessel water-level redundancy and independence to the existing designs for BWR overfill protection systems that are less reliable than the reference plant design, the benefits are not considered significant for plants that have some sort of automatic reactor vessel high-water-level feedwater trip system." The companion document to this report, NUREG 1217, further notes that "the estimated reduction in frequency of overfill events between plants that have some sort of automatic reactor vessel high-water-level feedwater trip system was not significant."

A GE topical report (Reference 4) was recently commissioned by the BWR Owners Group to verify NUREG 1217 and 1218 assumptions on BWR design, and to review estimates of licensee costs to install the trip logic meeting GL 89-19 requirements for separation and independent power supplies. This report also concludes that plant modifications are not cost beneficial in the estimated range of $192,000 to $1,074,000. PNPP cost estimates for an upgrade to independent/separate trip circuits are at the high end of that range.

CEI therefore concludes that PNPP modifications are not cost-justified.

References for Generic Letter 89-19 Evaluation

1. NUREG/CR-4387, "Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a General Electric Boiling Water Reactor," 12/85.

2. NUREG 1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plant - Technical Findings related to USI A-47," 6/89.

3. NUREG 1218, "Regulatory analysis for Resolution of USI A-47," 11/89.

4. EDE-07-0390, "BWROG Response to NRC GL89-19, Enclosure 2, Hardware Change Recommendations," submitted by letter BWROG-9048 (Floyd to Partlow), 4/2/90.

NJC/CODED/4482