UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-3

## TOLEDO EDISON COMPANY

## AND

## THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

## DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

## DOCKET NO. 50-346

### Introduction

By letter dated September 25, 1981, and in a response for more information from the NRC staff, by letters dated September 26, 1983, and September 11, 1984, the Toledo Edison Company (the licensee) proposed an amendment to the Davis-Besse Unit 1 Facility Operating License No. NPF-3. The proposed amendment provides Technical Specification (TS) changes for converting two existing steam generator (SG) drains to the main condenser presently used at shutdown temperatures and pressures to two SG blowdown systems which will be used at full operating temperatures and pressures. The licensee desires this change to decrease the operating time required to bring the secondary water purity into operating limits during startups and to provide for better control of SG level following possible tube ruptures.

The modifications, as described in the September 26, 1983 submittal, include the removal of the 1-inch motor operated containment isolation valves (CIV) 603A and 611A in the SG drain systems and the modification of the two 4-inch remote motor operated CIVs 603 and 611 to permit automatic closure of these valves on Steam Feedwater Rupture Control System (SFRCS) trips actuated by a loss of feedwater, steam line break, or feedwater line break. The modification will also indirectly provide containment isolation from a High-High containment pressure Safety Features Actuation System (SFAS) trip which isolates the feedwater system thus giving a loss of feedwater SFRCS trip and CIV closure. The modifications also include the addition of two 4-inch remotely operated blowdown control-drag valves which prevent water hammer in the lines near the condenser and two manual isolation valves near the condenser.

Presently, all drain valves and CIVs are closed during startup and power operations. The licensee proposes to operate the new blowdown systems with all valves inside containment open, the MS 603 and MS 611 CIVs may be operated closed or open, and the drag valves closed. Although the blowdown systems are designed to operate at any power level and may be pressurized during operations, they will be normally used for blowdown during startups.

Since the blowdown lines may be open down to the drag valves near the condenser, the lines may experience the normal operating temperatures (594°F) and pressures (925 psig). The present drain line from SG 1-1 leaves containment on the west side through penetration P-58, into Penetration Room PR 208 and through Room 303. These rooms were not designed for the environmental conditions that could result from high pressure line breaks. To avoid high pressure lines in these rooms, a new 4-inch line is installed inside containment routing the blowdown from SG 1-1 on the west side of containment, to spare penetration P-60 on the east side of containment. Penetration P-60 leaves containment into Penetration Room PR-236 which also contains penetration P-57, the existing SG 1-2 blowdown exit. Both blowdown CIVs, MS 603 (relocated) and MS 611 (existing), will be located in PR-236. Penetration P-58 will be converted to a spare penetration and the old SG 1-1 drain lines will be mostly left in place.

A summary of the modifications follows:

| SG 1-1 | SG 1-2 |
|---|---|
| o Installation of 221 ft of new pipe inside containment | o No piping change inside containment |
| o From containment penetration to condenser: 81 feet is existing; 167 feet is new | o From containment penetration to condenser: 104 feet is existing; 140 feet is new |
| o 34 new supports added | o 17 new supports added |
| o 73 existing supports | o 82 existing supports |
| o Added 4 pipe whip restraints (WRs), R1, R2A, R2B, and R3 | o Added 6 WRs, R4 through R9 |
| o Stress Calculation No. 12B, 163, 164 and 641 | o Stress Calculations No. 12A, 64D, and 64G |

Evaluation

We evaluated the modification information supplied in the licensee's submittal of September 26, 1983, against the criteria of Standard Review Plan (SRP), NUREG 0800, Sections 3.6.1, Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment; 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping; 6.2.1, Containment Functional Design; 6.2.3, Secondary Containment Functional Design; 6.2.4, Containment Isolation Systems; Branch Technical Positions BTP ASB 3-1, pertaining to SRP Section 3.6.1; and BTP MEB 3-1 pertaining to SRP Section 6.2.4.

We reviewed selective analyses from the licensee contractor's piping stress analysis No. 163 (D2), dated December 2, 1981. Areas reviewed included:

Global coordinates, pipe O.D. and wall thickness, allowable stress (hot and cold) as specified by specification, and design criteria from Node Points 280 to 320.

Application of stress intensification factor at elbow, Node No. 315.

Loading Combinations:

(1) $S_{th} - S_a$

(2) $S_w + S_p + S_{OBE} - 1.2\ S_h$

(3) $S_w + S_p + S_{SSE} - S_y$ for old pipe

(4) $S_w + S_p + S_{SSE} - 2.4\ S_h$ for new pipe

(5) $S_w + S_p + S_{OBE} + S_{th} - 0.8\ (.2\ S_p + S_a)$

Where

$S_{th}$ = Thermal Stress

$S_w$ = Weight Stress

$S_p$ = Pressure Stress

$S_{OBE}$ = Operating Basis Earthquake Stress

$S_{SSE}$ = Safe Shutdown Earthquake Stress

$S_h$ = 15,000 psi for SA 106 Gr. B Pipe at 536°F

$S_a = f(1.25\ S + 0.25\ S_h)$ = 22,500 psi

$S_y$ = 27,436 psi @ 536°F

No adverse findings were revealed, except an allowable stress of 2.4 $S_h$ was not considered to be technically acceptable for load combination 4, and was not in accordance with the original Safety Analysis Report (SAR) commitment. However, since all pertinent stresses were within $S_y$, safety is not compromised.

The review also included a selected seismic restraint analysis for the snubber and rigid restraint at Node No. 283. The contractor's structural and mechanical loads and associated stress calculations were contained in Problem No. 163, "Davis-Besse 1 Steam Generator No. 1-1 Drain/Blowdown System" dated January 14, 1982. The analysis reviewed was considered acceptable.

Due to the deletion of Penetration P-58 and utilization of the spare penetration P-60, new analyses were performed for the flued head containment vessel penetration anchor and affected area vessel reinforcement plate. Design loadings were developed and documented in Plant Design Calculation No. M1 (01), "Pipe Stress Analysis Resolved Penetration Loads P-57 and P-60", dated January 27, 1982. The analyses are reported in "Certification of Addendum 1 to Stress Report (dated February 18, 1977) for Containment Vessel at the Davis-Besse Nuclear Power Station, Unit No. 1, Port Clinton, Ohio", dated September 1, 1983. Measures taken to evaluate the design revisions were considered substantial and acceptable.

We reviewed the criteria used by the licensee to determine pipe break locations and whip restraint (WR) design and the criteria implementation by reviewing one of the highest pipe stress locations shown on Calculation No. 163 (D1), Node Point 80, which required installation of a pipe whip restraint (WR). The direction of pipe whip as well as measures to restrain the movements were checked. The review of the WR design is contained in Civil Calculation No. 6, Vol. F-11, "FCR 78-126: Steam Generator Blowdown System Pipe Whip Restraint R2", Rev. 1, dated February 24, 1982. Design and material strength calculations for the bent plate WR and the stretch bar WR were reviewed and no adverse conditions observed. The design criteria for the base plate and the concrete expansion anchors were based on IE Bulletin 79-02 requirements and were in order.

The use of the above criteria and Regulatory Guide (RG) 1.46 permits the determination of break locations by studying the piping maximum stresses instead of assuming longitudinal and circumferential breaks at all fittings. This is, however, not in accordance with the provisions contained in the present Updated Safety Analysis Report (USAR), Rev. 0, dated July, 1982, Paragraph 3.6.2.2.1, "Pipe Restraint Design Criteria to Prevent Pipe-Whip Impact Within the Containment Vessel." The five systems whose breaks are postulated per RG 1.46 do not include the SG Blowdown System.

The licensee proposes to revise USAR Paragraph 3.6.2.2.1 to include the SG Blowdown System as one of the systems permitted to be analyzed in accordance with RG 1.46. We feel that this method of analysis for the SG Blowdown lines is appropriate and find this revision to be acceptable.

Since the SG Blowdown lines will be at operating conditions of 594°F and 925 psig, safety analyses were also conducted to determine the possible environmental and physical effects due to compartment pressurization and jet impingement resulting from a pipe rupture. The analyses indicated that a more severe accident had already been postulated for all the areas through which the SG Blowdown lines pass, i.e.

The annulus area by a rupture of the 18-inch main feedwater line

Room 236 by a rupture of a 6-inch main steam line

Room 314 by an 18-inch main feedwater line

The Turbine Building by rupture of a 36-inch main steam line

Should one of the blowdown lines rupture outside containment, radioactivity would be released only if there is primary to secondary leakage. Any radioactivity releases of this type are within the allowable limits specified in the Technical Specifications. The rupture, if significant, would be readily detected and isolated by feedwater - steam flow and SG level imbalances. Minor leaks will be detected by routine patrols of Penetration Room, PR-236, Room 314, and the Turbine Building. See USAR Section 3.6.2.7.

The licensee has determined that jet impingement barriers will be required in two locations within containment, one to protect a 2½-inch High Pressure Injection pipe, the other a 1-inch electrical conduit for Decay Heat Valve 12.

The SG Blowdown systems have no other engineered safety function other than containment isolation and are classified as Type III penetrations in the USAR. A Type III penetration is a line that penetrates containment and is neither part of the reactor coolant system pressure boundary nor is it connected directly to the containment atmosphere. General Design Criteria 57 sets down the requirements for containment isolation of this type penetration as requiring at least one CIV which is either closed automatically, is locked closed, or is capable of remote manual operation.

Since the present SG drain lines are used only during shutdown conditions, the existing CIVs, MS-603 and MS-611, are remote manually operated valves, and, as required following a containment isolation signal, they are closed manually by procedure from the control room. The proposed SG Blowdown systems are still classified as Type III penetrations; however, the CIVs, MS-603 and MS-611, have been converted to automatic closure by SFRCS trips. These trips are initiated by low steam line pressure, main feedwater/SG differential pressure, SG low level, and reactor coolant pumps inoperative. CIVs MS-603 and MS-611 will also be closed automatically by High-High pressure (38 psig) in containment since this SFAS trip closes the main feedwater and main steam valves which in turn results in a feedwater/SG differential pressure (SFRCS trip).

The only TS changes required by this modification are in Table 3.6.2, Containment Isolation Valves. Changes include the deletion of the 1-inch CIVs, MS-603A and MS-611A, removal of the asterisk from CIVs MS-603 and MS-611, moving these valves to Section A of the table, changing the name of the valves from SG Drain Lines to SG Blowdown Lines, changing the penetration number for MS-611 from P-58 to P-60 and adding the isolation time of 80 seconds.

Having found that the redesign meets the required criteria, the above TS changes are found to be acceptable.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 11, 1984

The following NRC personnel contributed to this Safety Evaluation:
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