UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-58

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated June 24, 1992, as supplemented by letter dated September 25, 1992, the Cleveland Electric Illuminating Company, et al. (the licensee) submitted a request to amend the Technical Specifications (TSs) for the Perry Nuclear Power Plant, Unit No. 1. In particular, the licensee requested revision to a number of primary containment pressure and temperature limits and the suppression pool water level limit based on a revised containment response analysis in order to address the problem of recurring high ambient and lake temperatures during the summer months. The supplemental letter did not affect the notice of Opportunity for Hearing published in the Federal Register on July 1, 1992 (57 FR 29337). By letter dated June 21, 1993, the licensee requested partial issuance of the amendment for the summer of 1993; however, the NRC staff did not act on that request.

By letter dated November 16, 1992, the licensee submitted a request to amend the TSs to permit a reduction in the water level of the upper containment pool water level during plant operations, provided that the suppression pool water level was increased in order to compensate. Due to the similarity of the subject matter, the staff decided to review this amendment request in conjunction with the above mentioned amendment request.

2.0 EVALUATION

The licensee's amendment request is based upon a containment response analysis performed by GE Nuclear Energy (GE), as well as structural design and operational impact reviews performed by Gilbert Associates, Inc. (GAI), and the licensee. The containment analysis comprises the bulk of the licensee's technical justification, and is discussed in detail below.

2.1 CONTAINMENT ANALYSIS

Primary containment temperature and pressure response following a postulated loss-of-coolant accident (LOCA) is of great importance when determining the potential for offsite release of radioactive material, in determining Emergency Core Cooling System (ECCS) pump net positive suction head (NPSH) requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment. As part of the generic BWR power uprate program, GE proposed to update the calculational methods used for determining peak containment temperatures and pressures following a postulated LOCA. In particular, GE proposed to utilize the SHEX computer code when calculating the peak pressures and temperatures during the long-term portion of containment post-LOCA response, in place of the previously used M3CPT/HXSIZ combination. The staff agreed with the use of SHEX in support of the generic BWR power uprate program; however, at that time the staff did not support the use of SHEX for other uses. In a July 13, 1993, letter from A. Thadani (NRC) to G. Sozzi (GE), the staff expanded the acceptable uses of SHEX, and stated that although SHEX had not been approved by the NRC staff for generic use, the use of SHEX on a plant-specific basis would be permitted. The use of SHEX in the evaluation of long-term containment response is not currently part of the licensing basis at Perry.

The licensee's June 24, 1992, amendment request contained a GE Topical Report entitled, "Perry Technical Specifications Improvement - Containment Response Analysis" (NEDC-31940). This report contains the results of revised containment pressure and temperature analysis performed to evaluate both short-term and long-term response of the containment to postulated LOCAs. GE used the M3CPT computer model to perform the short-term analysis, and the SHEX computer model to perform the long-term analysis.

2.1.1 Input Assumptions

In performing the containment response analysis, GE proposed a number of revised TS limits which would provide greater operational flexibility for the licensee. A comparison of the current limits to the analyzed limits is included in Table 1. (The licensee subsequently revised several of these proposed limits, based upon structural and equipment qualification evaluations performed by GAI and the licensee.) The staff does not object to the use of these limits in the containment response evaluation.

Table 1

COMPARISON OF CURRENT AND ANALYZED TECHNICAL SPECIFICATION LIMITS

Description	Current Limit	Analyzed Limit
Containment Ambient Temperature	90°F	104°F
Suppression Pool Temperature	90°F	95°F
Suppression Pool Lower Water Level (Depth)	18'0"	17'6"
Minimum Upper Pool Depth (above reactor vessel flange)	22'10"	22'9"

Other input assumptions were revised in order to make the results of the analysis more conservative. These revised input assumptions are listed in Table 2.

Table 2

REVISED INPUT ASSUMPTIONS USED
IN CONTAINMENT RESPONSE ANALYSIS

Description	FSAR Value	Revised Value
COMMON ASSUMPTIONS		
Reactor Power (MWt)	3650	3729
Initial Suppression Pool Temperature	90°F	95°F
SHORT-TERM ANALYSIS		
Initial drywell/containment differential pressure (psid)	0.0	-0.5
Initial primary/secondary containment differential pressure (psid)	0.0	+1.0
LONG-TERM ANALYSIS		
Initial suppression pool water level	18'0"	17'6"
Initial drywell/containment differential pressure (psid)	0.0	+2.0
Upper pool dump to suppression pool	1800 sec	1800 sec or Rx low-low level
Temperature of upper pool water	100°F	110°F
ESW temperature at inlet to RHR heat exchanger	80°F	85°F

2.1.2 Short-Term Response

When evaluating containment post-LOCA response, the M3CPT code is used to calculate short-term containment temperature and pressure response following a postulated LOCA, while either SHEX or a combination of M3CPT and HXSIZ would be used to determine the long-term suppression pool temperature. The M3CPT code uses a mechanistic method to model the highly transient conditions in the containment immediately following a LOCA, and is capable of modelling containment long-term response, up to the initiation of containment cooling. M3CPT has been verified against experimental data and has been previously approved by the NRC staff.

Short-term containment response is primarily affected by initial drywell and wetwell pressures and the suppression pool water level and temperature. By assuming a negative initial drywell to wetwell differential pressure, the analysis took into account the reduction of drywell free-air space due to an increase in the amount of water in the weir area of the drywell. Decreasing the free-air volume tends to increase the peak drywell pressure calculated by the short-term analysis. An increase in the water level in the weir area can be caused by either a negative drywell to wetwell differential pressure or an increase in overall suppression pool level above normal. GE performed a sensitivity study to ensure that the -0.5 psid drywell to wetwell differential pressure bounded all other normal suppression pool level variations.

The short-term containment response analysis yielded results, listed in Table 3, which are similar to those obtained from the Final Safety Analysis Report (FSAR). The main steam line break within the drywell dominated the short-term analysis, predicting higher peak pressures than the recirculation suction line break. The short-term containment hydrodynamic loads due to LOCA bubble and pool swell loadings did not change significantly from the FSAR. The peak pressures calculated by the short-term analysis fall well below the containment design limits, and are therefore acceptable.

Table 3

SHORT-TERM CONTAINMENT RESPONSE

	Revised	FSAR	Design Limit
Peak drywell pressure (psig)	21.8	22.1	30
Peak drywell/containment differential pressure (psid)	20.4	21.2	30
Peak wetwell pressure (psig)	11.2	9.2	15

2.1.3 Long-Term Response

During the 1970's and early 1980's, GE used the M3CPT/HXSIZ combination to model the long-term response of the containment to a spectrum of LOCAs. The M3CPT code was used to model both the short-term and long-term response to the LOCA from the time of the breakup to the time of initiation of containment cooling. After initiation of containment cooling, the HXSIZ code was used to model the containment heat exchangers, using input values obtained from M3CPT. By modelling the containment heat exchangers, the suppression pool temperature could be calculated as a function of time.

The SHEX code utilizes more refined models than those used by M3CPT/HXSIZ to determine suppression pool temperature, and is capable of modelling containment responses to more accident scenarios than the HXSIZ code. Many of the models used in SHEX are the same as, or very similar to, those used in M3CPT. SHEX is also capable of modelling all containment auxiliary systems, permitting a more accurate analysis of actual containment conditions following a postulated LOCA.

Containment long-term response is affected by suppression pool volume, initial temperature, and heat-exchanger efficiency, all of which affect the ability of the containment to absorb the heat loads caused by the proposed LOCA. In the long-term containment response analysis performed by GE, the suppression pool volume was decreased (due to an assumed drywell to wetwell differential pressure and lower overall suppression pool and upper containment pool water levels), the initial pool temperature was increased, and the efficiency of the heat exchangers was decreased (by increasing the temperature of the service water at the inlet to the heat exchangers). Decreasing the ability to remove heat from the containment increases the peak pressures and temperatures calculated by the analyses. GE performed a sensitivity analysis of the effect of suppression pool levels on the long-term containment response analysis in order to determine revised high and low water level limits for the suppression pool.

GE, on behalf of the licensee, analyzed a spectrum of LOCA break sizes and events to determine the impact on the long-term containment response. Results of the long-term containment response analysis are summarized in Table 4. Several different accident scenarios were used to define the limiting peak pressures and temperatures; however, results of all analyses remained within the design basis of the containment structure, and are, therefore, acceptable.

2.2 PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

As a result of the revised containment response analysis, the peak pressure expected to be experienced by the containment has been reduced from 11.31 psig to 7.8 psig. The licensee has proposed to change all references to this peak pressure to the new value, permitting leakage testing of the containment to be performed from a lower initial pressure. Specifically, the licensee has changed references to the peak pressure contained in TSs 4.6.1.1.1, 3.6.1.2, 4.6.1.2, 3.6.1.3, and 4.6.1.3, and the Bases for TS 3/4.6.1.2, 3/4.6.1.6, and 3/4.6.2.5. These changes are consistent with the results of the containment analyses and are therefore acceptable.

The licensee also proposed several changes to the minimum suppression pool and upper containment pool water levels. Specifically, the licensee proposed to change the minimum suppression pool water level described in TS 3.6.3.1 from the present value of 18'0" to 17'9.5" plus a "level adjustment factor." All

Table 4

LONG-TERM CONTAINMENT RESPONSE

	Event	Value	FSAR	Limit
Peak drywell temperature	MSLB	328.7°F	330.0°F	330.0°F
Peak suppression pool temperature	ASD-A	184.7°F	184.6°F	185.0°F
Peak containment pressure (psig)	MSLB	7.8	11.3	15.0
Peak containment temperature	LHS	160.5°F	184.6°F	185.0°F

MSLB -- Main steam line break inside drywell
ASD-A -- Alternate shutdown event A
LHS -- LOCA from hot standby conditions

other explicit references to minimum suppression pool water level have been removed and replaced with a reference to TS 3.6.3.1. The level adjustment factor was developed by the licensee to account for changes in suppression pool volume caused by drywell to wetwell differential pressure. Using the level adjustment factor (which is always zero or positive), the suppression pool water level (and overall suppression pool water volume) would always be maintained above that assumed in the containment long-term response analysis. The licensee stated that although the graph defining the suppression pool adjustment factor is not contained in the TSs, it would be maintained in accordance with other plant procedures, and any changes would be subject to Plant Operations Review Committee (PORC) review. The staff finds this proposed change to be acceptable.

The licensee has proposed to reduce the minimum water level in the upper containment pool from the present limit of 22'10" above the reactor vessel flange to 22'5" above the flange, provided that the suppression pool level is increased by 2.20" to compensate. Since the net volume of the suppression pool (after makeup from the upper pool) will remain the same, the containment long-term analysis will remain valid. Therefore, this change is acceptable.

In addition, the licensee proposed to revise the suppression pool and containment air temperatures described in TSs 3.6.1.7, 3.6.3.1, and 4.6.3.1 from 90°F to 95°F. The staff finds these changes to be acceptable.

The staff has concluded that the containment temperature and pressure response following a postulated LOCA will remain acceptable after implementation of the proposed changes. The staff also concludes that the containment will continue to meet the requirements for sufficient margin from temperature and pressure limits as described in 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment design basis." The staff, therefore, considers the proposed changes to the TSs of the Perry Nuclear Power Plant, Unit No. 1, as proposed by the licensee, to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 PUBLIC COMMENTS

By letter dated August 17, 1992, comments regarding the licensee's June 24, 1992, amendment request were received from the Ohio Citizens for Responsible Energy, Inc. (OCRE). The comments focused on two principal issues regarding the licensee's amendment request: (1) the authors assert that the impact of accidents beyond those in the plant design basis have not been addressed by the licensee in the amendment request, and (2) the change in methodology used in the containment analysis completely masks the effect of the proposed TS changes. The staff does not consider these comments to have technical merit for the following reasons:

With respect to OCRE's first concern regarding the licensee's failure to perform an analysis of the impact of the amendment request on those accidents which are beyond the design basis, 10 CFR Part 50 does not require that requests to amend plant TSs address accidents which go beyond the design basis of the plant. The purpose of the Individual Plant Examination (IPE) program was to specifically address the effect of accidents beyond the design basis on plant equipment, and to highlight risk-significant areas for improvement. The NRC intends that the IPE will be maintained by the licensee as a living document, and that plant modifications, including those done under 10 CFR Part 50.59, will be periodically incorporated into the PRA models developed for the plant. However, the concept of risk-significance is specifically not used in the technical review of plant license amendments. Instead, the staff relies on existing NRC rules, as contained in Title 10 of the Code of Federal Regulations, Regulatory Guides, and NRC generic correspondence, as well as guidance provided in the Standard Review Plan (NUREG-0800) when evaluating the acceptability of proposed license amendments. The analyses presented by the licensee in the abovementioned submittals is technically sound, and the proposed TSs do not violate these existing NRC requirements; therefore, the staff has found them to be acceptable.

OCRE's second concern was that the use of a different methodology for the containment analyses effectively masks the effects of the various changes to input assumptions. Direct comparison of the results of the new analyses against the FSAR analyses, even using the same input assumptions, would not have yielded any useful information. One would expect that the "more realistic" methodology used by SHEX would result in lower peak temperatures and pressures than the preceding analysis. Additionally, "taking credit" for previously unrecognized margin is acceptable to the staff, if done in a cautious manner. The staff has reviewed the input assumptions used in the analyses and has compared these to the input assumptions used in the original FSAR analysis. The licensee revised several input assumptions from the original FSAR analysis to reflect actual plant conditions; since these assumptions tended to produce less favorable results from the analyses, the staff considers this to be an added conservatism from the original analysis. Other assumptions were changed to allow for more flexible operation of the plant; these changes (and the associated changes to the TSs) could prevent unnecessary plant shutdowns and challenges to plant safety equipment. Thus, the staff has no reason to find the proposed changes unacceptable.

The staff compared the results of the new analyses to those obtained using the FSAR methodology for consistency. Unlike calculations performed to show compliance with 10 CFR Part 50, Appendix K, containment response analyses do not need to be performed to a prescribed NRC methodology, using an NRC approved computer code. Although the NRC has not explicitly reviewed the SHEX computer code, the staff did undertake detailed review of the preceding codes M3CPT and HXSIZ, and ultimately granted NRC approval of these codes. The staff has reviewed all aspects of the containment analyses, including the input assumptions, methodologies, and results, compared these against those contained in the FSAR and in applicable NRC regulations, and has found the containment analyses submitted by the licensee to be acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 15, 1994 (59 FR 12013). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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