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PERRY NUCLEAR POWER PLANT


March 16, 1990
PY-CEI/NRR-1140 L

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

Perry Nuclear Power Plant
Docket No. 50-440
Technical Specification Change
Request-Performance of
Containment Isolation Valve
Testing With Primary
Containment Integrity-Shutdown

Gentlemen:

The Cleveland Electric Illuminating Company (CEI) hereby requests amendment to the Perry Nuclear Power Plant - Unit 1 Facility Operating License NPF-58. In accordance with the requirements of 10CFR50.91(b)(1) a copy of this request for amendment has been sent to the State of Ohio as indicated below.

This application is filed to revise the Primary Containment Integrity-Shutdown Technical Specification (LCO 3.6.1.1.2) to provide limited flexibility during the performance of 10CFR50 Appendix J Type C leak rate testing of containment isolation valves. Attachment 1 to this letter provides the Summary, Safety Analysis, Significant Hazards and Environmental Impact Considerations. Attachment 2 provides the proposed changes to the Technical Specifications.

In order to permit potential critical path LLRT's to be performed during periods requiring containment integrity during the upcoming second refueling outage, it is requested that this amendment be processed prior to the start of the outage, presently scheduled for September 7, 1990.

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

Very truly yours,

Al Kaplan
Vice President
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## Summary

On March 31, 1989 the NRC issued Amendment No. 19 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant. The approved amendment revised the Technical Specification for Primary Containment Integrity-Shutdown. It allowed the performance of containment isolation valve Type C local leak rate tests with 3/4 inch vent and drain lines open on certain penetrations that would otherwise not be testable when the specification is applicable (such as during refueling activities). This amendment was approved for the first refueling outage only. The Safety Evaluation accompanying the approved amendment indicated that final acceptance of a permanent change to the Technical Specifications was contingent upon a favorable review by the staff of the original analysis without reliance on a 15 day decay period used by the staff in the interim approval. CEI has performed analyses which demonstrate that as many penetrations as desired could remain open provided there is a seven day delay between plant shutdown and starting the local leak rate tests. This seven day period provides for significant decay of the source term contained in the fuel. Based on this analysis, CEI is requesting that the Technical Specification be changed to permit the performance of Type C local leak rate tests with as many as six (6) open vent/drain line pathways as long as the facility has been subcritical for 7 days or more. The analyses in this submittal supercede the previously submitted analyses in the December 29, 1988 letter.

## Safety Analysis

The Type C local leak rate tests specified in 10CFR50 Appendix J require that the containment isolation valves be tested by pressurization with air or nitrogen at the calculated peak containment internal pressure resulting from the design basis accident (11.31 psig). In the original Technical Specification for Primary Containment Integrity-Shutdown, many of the local leak rate tests could not be conducted during refueling activities since primary containment integrity would not be maintained during depressurizing/draining of the test volume in preparation for testing, and during system restoration. In our letter dated December 29, 1988 we included a typical simplified piping arrangement and a description of the sequence of events for the testing of the containment isolation valves. The December 29, 1988 letter assumed only 24 hours of decay time, and requested that the Technical Specifications be revised to permit opening of up to two vent and/or drain line pathways (3/4 inch diameter each) while Primary Containment Integrity-Shutdown was required. Based on our analysis at that time we also stipulated that primary to secondary containment differential pressure had to be verified within the limits of Technical Specification 3.6.1.6 whenever 1 or 2 vent/drain line pathways were open.

The NRC staff reviewed our request and performed an independent calculation based on our submittal and concluded that it was acceptable to change the Technical Specifications as requested with the following stipulations:

1. The staff assumed a decay period of 15 days versus the 1 day assumed in our submittal since the plant had already been shutdown for 15 days at that point in the staff review.

2. The Amendment was only approved for the first refueling outage. The NRC staff indicated in the Safety Evaluation Report which accompanied the Amendment that they would have to review the issue in more detail to revise the specification permanently.

Based on the above, CEI has completed further analyses in order to support this request to make the change permanent. The new analyses used the following assumptions:

1. A decay period of 7 days (168 hours) subcriticality was assumed. This decay period was selected since it provides for significant decay of the source term contained in the fuel. Realistic scheduling shows that at least six days would be required after entering Mode 3 before primary containment integrity would be needed to support fuel handling activites, and the seventh day was added to ensure that offsite dose consequences from a postulated fuel handling accident would remain within the NRC SRP acceptance critera. This seven day time period is considered acceptable by CEI for support of outage LLRT activities.

2. All the airborne activity existing inside of the containment after the accident is assumed to be immediately discharged to the environment. This highly conservative analysis assumes that there is no containment barrier, no dilution prior to the release and no filtering done by any ventilation system.

3. The radiological analysis utilized the PNPP short term (accident) dilution factors (X/Q) provided in USAR Table 2.3-24 for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) based on seven site years.

4. All other assumptions and input parameters are provided in Table 1 to this letter. These other assumptions are consistent with those used in USAR Section 15.7.6. This USAR analysis had previously been reviewed by the NRC staff in the Perry Safety Evaluation Report (NUREG-0887) dated May 1982 and found consistent with the requirements of Regulatory Guide 1.25 and the procedures specified in the SRP Section 15.7.4. This was restated in the Safety Evaluation which accompanied the Amendment 19 approval letter dated March 31, 1989. Table 2 to this letter provides the values of radiological activity released into the containment and then directly to the environment.

Using the above assumptions, the dose was calculated for both the Exclusion Area Boundary and the Low Population Zone. Table 3 shows the results of these calculations, for both the Design Basis case and the Realistic case values. As shown by the Table, this conservative analysis indicates that the calculated values are all within the guidelines of the SRP Section 15.7.4. The SRP states that the calculated values should be no more than 25 percent of the 10 CFR 100 limits of 300 rem to the thyroid and 25 rem whole body at the EAB and LPZ. This then limits the doses to 75 rem to the thyroid and 6 rem to the whole body. Table 3 shows that all calculated values are within these SRP acceptance criteria even with the highly conservative assumptions made in the calculation.

The above analysis is conservative for several reasons. First the analysis assumes no containment. The analysis demonstrates that it would be acceptable to have no containment integrity for the fuel handling accident after being shutdown for 7 days. However, in the proposed Technical Specification CEI has limited the number of open vent and drain pathways to the six. This was done in response to known NRC concerns over administrative controls on the closing of these vent/drain pathways in the event of an accident, and to ensure that the calculated doses would not be exceeded. When utilizing the proposed change, at no time during the testing process are the containment isolation valves disabled, and therefore, the containment isolation function provided by the valves would remain available if called upon to close. Additionally, administrative controls will be established in order to ensure that the number of 3/4 inch vent and drain pathways opened at any one time will be limited to 6. These controls will include assuring that the control room operators are aware of how many pathways may be opened by the LLRT teams at any one time, and will stipulate that the test engineers make reasonable attempts to isolate vent/drain lines prior to evacuating if evacuation is announced over the PA system. Thus, the containment system will remain intact.

The analysis also does not assume any credit for ventilation filters and no decay of the isotopes following the accident. As described in our December 29, 1988 submittal, the Containment Vessel and Drywell Purge System would typically isolate on high radiation following the accident. In the original submittal, it was conservatively estimated that this would occur 20 seconds after the accident. However, using the new analysis assumptions that all the airborne activity is immediately discharged to the environment, no credit is taken for this isolation or for the charcoal and HEPA filters which this system contains.

The new analysis takes no credit for mixing. This analysis assumed that the air directly over the pool is immediately exhausted to the atmosphere. In reality, substantial mixing would take place with the containment atmosphere which would dilute the discharge to the environment, and greatly reduce the EAB doses.

Based on the above, the analysis is conservative and within the limits described by the SRP and 10 CFR 100 guidelines, and the additional restrictions placed on the number of open vent and drain pathways is conservative.

**Significant Hazards Consideration**

In accordance with the requirements of 10 CFR 50.92, the following discussion is provided in support of the determination that no significant hazards are involved with the changes proposed in this amendment request.

1. No significant increase in the probability or consequences of an accident previously evaluated results from the proposed changes because:

   Having up to 6 vent and drain pathways does not increase the probability of any accident previously evaluated. As discussed above, the accident of concern is the fuel handling accident inside containment. Having vent and drain valves open does not increase the probability of this accident occurring. The proposed change also does not increase the consequences of this accident since as described above, the new analysis performed shows that the doses at both the EAB and LPZ are within the guidelines of SRP 15.7.4 and well within the requirements of 10 CFR 100. Therefore the consequences of the accident are not increased and the proposed action would result in no significant radiological environmental impact.

2. The proposed change will not create the possibility of a new or different kind of accident than previously evaluated because:

   The initiating event and event sequence remain unchanged. The initiating event is the dropping of a channeled fuel assembly onto the core as a result of the failure of the fuel assembly lifting mechanism. The number of fuel rods damaged as a result of this event (124) were conservatively calculated using the methodology provided in USAR Sections 15.7.4 and 15.7.6, and are unchanged by the proposed amendment. The only change resulting from the proposed amendment is the evaluation of the consequences of the postulated event inside primary containment. Additionally, the proposed change does not alter the plant design or functional capability and does not introduce any new operating modes, only new potential pathways. As a result, the proposed amendment does not create the possibility of a new or different kind of accident than previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety because:

The margin of safety is provided by maintaining the offsite dose consequences of a postulated fuel handling accident well within the exposure guidelines of 10 CFR 100. As stated above the Standard Review Plan (SRP) Section 15.7.4 provides additional guidance by defining "well within" as 25 percent or less of the 10 CFR 100 guidelines.

Application of this recommendation would result in a limit of 75 rem for the thyroid and 6 rem for the whole body at the Exclusion Area Boundary (EAB). The margin of safety is provided by the difference between the exposure guidelines in the SRP Section 15.7.4 and the 10 CFR 100 limits of 300 rem to the thyroid and 25 rem to the whole body at the EAB. As shown in Table 3, the calculated doses based on the new analyses is 45.6 rem for the thyroid and 0.114 rem for the whole body at the EAB. Doses for the Low Population Zone (LPZ) were also calculated and determined to be 5.09 rem thyroid and 0.0127 rem whole body. Since these offsite doses are below the exposure guideline values provided in SRP 15.7.4 and 10 CFR 100, the existing margin of safety has been maintained.

Based on the above considerations, the proposed change does not significantly increase the probability or the consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident from any previously evaluated, and does not involve a significant reduction in the margin of safety. Therefore, the Cleveland Electric Illuminating Company proposes that no significant hazards are involved.

**Environmental Impact Consideration**

Cleveland Electric Illuminating Company has reviewed the proposed Technical Specification change against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed changed does not involve a significant hazards consideration, nor significantly increase the types and amounts of effluents released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, CEI concludes that the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement. The NRC staff arrived at the same conclusion regarding an environmental impact statement for the interim approval given in Amendment 19.

NJC/CODED/V3182

**TABLE 1**
**FUEL HANDLING ACCIDENT INSIDE CONTAINMENT**
**PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS**

| | | Design Basis Assumptions | Realistic Assumptions |
|---|---|---|---|
| I. | Data and assumptions used to estimate radioactive sources from postulated accident. | | |
| A. | Power level | 3758 MWt | Same |
| B. | Burnup | 1,000 days | Same |
| C. | Radial peaking factor | 1.5 | Same |
| D. | Fuel damage | 124 rods | Same |
| E. | Release of activity by nuclide | 10% iodine<br>30% Kr-85<br>10% other noble gases | 0.32% iodine<br>1.80% Kr-85<br>1.80% other noble gases |
| F. | Radionuclide decay time | 168 hrs. (7 days) | Same |
| G. | Iodine gap activity species | | |
| | 1. Organic | 0.25% | Same |
| | 2. Inorganic | 99.75% | Same |
| H. | Minimum water coverage above damaged fuel rods | 23 ft. | Same |
| I. | Pool decontamination factors: | | |
| | 1. Organic iodine | 1 | Same |
| | 2. Inorganic iodine | 133 | Same |
| | 3. Noble gases | 1 | Same |
| J. | Activity Airborne in Containment | Table 2 | Table 2 |
| II. | Data and assumptions used to estimate activity released | | |
| A. | Release pathway | Instantaneous unfiltered exhaust direct to the environment | Same |
| B. | All other pertinent data and assumptions | Reg. Guide 1.25 | Same |
| III. | Dispersion Data (USAR Table 2.3-24) | | |
| A. | Exclusion Area Boundary (863 meters) | $4.3 \times 10^{-4}$ sec/m$^3$ | Same |
| B. | Low Population Zone (4002 meters) | $4.8 \times 10^{-5}$ sec/m$^3$ | Same |
| IV. | Dose Data | | |
| A. | Method of dose calculation | Reg. Guide 1.25 | Same |
| B. | Dose conversion assumptions | Reg. Guide 1.25 | Same |
| C. | Doses | Table 3 | Table 3 |

**TABLE 2**

**FUEL HANDLING ACCIDENT INSIDE CONTAINMENT**

**ACTIVITY AIRBORNE IN THE CONTAINMENT BUILDING AND RELEASED TO THE ENVIRONMENT (CURIES)**

| Isotope | Activity | |
|---|---|---|
| | Design | Realistic |
| I-131 | 2.05E+2 | 1.68E+1 |
| 132 | * | * |
| 133 | 1.89E+0 | 4.95E-2 |
| 134 | * | * |
| 135 | 1.90E-5 | 2.82E-7 |
| Kr-83M | * | * |
| 85M | 7.54E-8 | 4.16E-9 |
| 85 | 8.95E+2 | 6.00E+2 |
| 87 | * | * |
| 88 | * | * |
| 89 | * | * |
| Xe-131M | 1.82E+2 | 4.67E+1 |
| 133M | 1.35E+3 | 1.35E+2 |
| 133 | 2.52E+4 | 4.20E+3 |
| 135M | * | * |
| 135 | 2.69E-1 | 5.59E-2 |
| 137 | * | * |
| 138 | * | * |

* Indicates isotope activity is less than E-10 curies. (i.e. dose contribution is insignificant when compared to the other isotopes).

**TABLE 3**

**FUEL HANDLING ACCIDENT INSIDE CONTAINMENT**

**RADIOLOGICAL EFFECTS**

| | Design Basis Values | | Realistic Values | |
|---|---|---|---|---|
| | Whole Body Dose (rem) | Inhalation Dose (rem) | Whole Body Dose (rem) | Inhalation Dose (rem)* |
| Exclusion Area (863 meters) | 1.14E-1 | 4.56 E+1 | 1.81E-2 | 3.73E+0 |
| Low Population Zone (4,002 meters) | 1.27E-2 | 5.09E+0 | 2.03E-3 | 4.16E-1 |

* These values are over-estimated by a factor of from 10 to $10^6$ since an iodine partition factor of 100 was utilized for conservatism rather than a value of from $10^3$ to $10^8$ as have been experimentally determined.

NJC/CODED/3208