ATTACHMENT 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS AND BASES

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated June 20, 1996, Commonwealth Edison Company (ComEd), the licensee, requested changes to the Dresden Nuclear Power Station, Units 2 and 3, Technical Specifications (TS). Dresden Units 2 and 3 currently use Siemens Power Corporation (SPC) fuel and licensing methodologies. ComEd renewed the SPC contract to supply fuel and support services for Dresden Unit 3 Cycle 15 and Dresden Unit 2 Cycle 16. Two significant changes under the new SPC contract include introduction of an advanced fuel design, ATRIUM-9B, and the use of a new revision to Siemens' LOCA methodology. Thus, the proposed changes to the Dresden Units 2 and 3 TS incorporate NRC approved thermal limit licensing methodology in the list of approved methodologies used in establishing the cycle specific thermal limits. Other minor editorial changes are also proposed.

By letters dated December 30, 1996, and March 5, 1997, ComEd submitted revisions that were required for the approval of Technical Specification changes for SPC fuel transition for LaSalle County Nuclear Power Station Units 1 and 2. The revision lists the specific NRC approval date and the revision/supplement for each of the new topical reports, and revises section 5.3.A description of fuel assemblies.

2.0 EVALUATION

2.1 Safety Limits and Limiting Safety System Settings

2.1.1 Section 2.1.D Reactor Vessel Water Level

The licensee has proposed to remove the word "the" prior to "active," and add a footnote following the word "fuel" in section 2.1.D, Reactor Vessel Water Level. The proposed change would be as follows:

> 2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of active irradiated fuel[a].

The footnote would state:

> a. The top of active irradiated fuel is defined to be 360 inches above vessel zero.

The licensee stated that current fuel designs, including ATRIUM-9B, incorporate slight design variations in the length of the active fuel and therefore, the true location of the top of active (irradiated) fuel may be difficult to distinguish. However, this fixed reactor vessel reference point, i.e., 360 inches above vessel zero, for top of active fuel is used for the automatic initiations associated with both accident and transient analyses. This reference point can also be found in other TS such as the Emergency Core Cooling System and Isolation Instrumentation Tables. Based on this information, the staff finds the inclusion of the footnote and the editorial change acceptable.

2.1.2 Table 2.2.A-1 Reactor Protection System Instrumentation Setpoints

The licensee also proposed to add the above footnote to Table 2.2.A-1, Reactor Protection System Instrumentation Setpoints. The proposed change would be as follows:

> 4. Reactor Vessel Water Level - Low $\geq$ 144 inches above top of active fuel[(b)]

This change causes footnote b on Table 2.2.A-1 to become footnote c. Since footnote b is the same as footnote a in section 2.1.D, the staff finds the incorporation of the footnote and the editorial change acceptable.

2.2 Safety Limits Bases

The licensee proposed an editorial change to the section 2.1, third paragraph, of the Bases. The current wording states "the fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an AOO." The proposed change would consist of the following:

> The fuel cladding integrity limit is set such that no fuel damage is calculated to occur as a result of an AOO.

The staff concludes that the change clarifies the meaning of the sentence and is acceptable.

In section 2.1.B, Thermal Power, High Pressure and High Flow, the licensee proposed editorial changes to paragraph one. The current wording of the last sentence states that "the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties." The editorial change would have the last two sentences of paragraph one consist of the following:

> Therefore, the fuel cladding integrity Safety Limit is defined such that, with the limiting fuel assembly at the MCPR Safety Limit, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. This includes consideration of the power distribution within the core and all uncertainties.

The staff notes that this wording is consistent with the bases in Improved

Standard Technical Specifications, NUREG-1433, Rev. 1, and therefore, it is acceptable.

In section 2.1.D, Reactor Vessel Water Level, the licensee proposed the incorporation of a few words to clarify the last sentence of the paragraph. The current wording is "the top of active fuel is 360 inches above vessel zero." The licensee has proposed that the sentence be as follows:

> The top of active irradiated fuel is defined to be 360 inches above vessel zero.

The staff finds this consistent with the above footnotes and acceptable.

2.3 <u>Limiting Safety System Settings Bases</u>

In section 2.2.A.1, Reactor Protection System Instrumentation Setpoints - Intermediate Range Monitor, Neutron Flux - High, the licensee proposed an editorial change to the third paragraph. The sentence with the proposed change states that "the results of this analysis show that the reactor is scrammed and peak power is limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit." The licensee proposed to change 1% to 7.7% to reflect the correct value in the UFSAR and SAR analysis. Section 7.6.1.4.3 of the UFSAR cites, in graphical form, 7.7% as the power level at which the IRMs terminate the low power RWE transient. Based on this information, the staff finds this editorial change acceptable.

In section 2.2.A.4, Reactor Protection System Instrumentation Setpoints - Reactor Vessel Water Level - Low, the licensee proposed to add a clarification of the top of active fuel at the end of the paragraph. The proposed last sentence of the paragraph would read "the top of active fuel is defined to be 360 inches above vessel zero." This statement is consistent with footnotes and other sections of the bases and therefore, is acceptable.

2.4 <u>Instrumentation Bases</u>

The licensee proposed a clarification to section 3/4.2, Instrumentation. The licensee proposed to add the following sentences to the bottom of the paragraph.

> Current fuel designs incorporate slight variations in the length of the active fuel and, thus the actual top of active fuel, when compared to the original fuel designs. Safety Limits, water level instrument setpoints and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiations associated with these events.

The proposed additions provide a clear definition and use of the top of active fuel reference point. The staff finds this addition to the bases acceptable.

In section 3/4.2.A, Isolation Actuation Instrumentation, the licensee proposed the removal of the following sentences from the second paragraph since retrofit 8x8 fuel is no longer used at Dresden Units 2 and 3.

> Retrofit 8x8 fuel has an active length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the loss-of-coolant accident (LOCA) analysis for Dresden Units 2 and 3.

The staff finds the removal of these sentences acceptable.

2.5 <u>Reactivity Control Bases</u>

In TS 3.3.B, Reactivity Anomalies, requires that the reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall not exceed 1% $\Delta k/k$. This limit ensures that plant operation is maintained within the assumptions of the safety analyses. The licensee proposed to include an alternative to monitoring reactivity anomalies in the technical specification bases. The SPC core monitoring code, Powerplex, provides the capability to monitor actual K_{eff} versus predicted K_{eff}. This method is currently used at Dresden to monitor reactivity anomalies. Thus the following will be added to section 3/4.3.B, Reactivity Anomalies Bases:

> Alternatively, monitored K_{eff} can be compared with the predicted K_{eff} as calculated by the 3-D core simulator code.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The staff notes that this proposed change only revises the current method of measuring the difference between predicted and monitored core reactivity and does not change the required limit, therefore, the change is acceptable.

In sections 3/4.3.D, 3/4.3.E, and 3/4.3.F, Control Rod Maximum Scram Insertion Times, Control Rod Average Scram Insertion Times, and Four Control Rod Group Scram Insertion Times, the licensee proposed to remove the following comments:

first paragraph:	"(as adjusted for statistical variation in the observed data);"
second paragraph:	"In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above;"
third paragraph:	"Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly;" and
fourth paragraph:	"If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is

> undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specification is not statistically significant and should not be used in the re-determination of thermal margins."

The licensee stated that the above information is based on past data which is a GE methodology. Current SPC methods used to evaluate the 5%, 20%, 50% and 90% control rod scram insertion times, collected during the performance of the scram timing Surveillance Requirement 4.3.D, will replace the above information as follows:

> Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and Nominal Scram Speed (NSS) insertion times. These analyses result in the establishment of the fuel cycle dependent TSSS MCPR operating limits and NSS MCPR operating limits which are presented in the COLR. Results of the control rod scram timing tests performed during the current fuel cycle are used to determine the operating limit for MCPR. Following the completion of each set of scram time testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times.

The NSS insertion times are typically faster than the TSSS insertion times, thus, the NSS insertion times are used to calculate the NSS MCPR operating limit. If any of the average scram insertion times do not meet the NSS times, the TSSS MCPR operating limit is used. TS 3.11.C, Minimum Critical Power Ratio, requires that the MCPR shall be equal to or greater than the MCPR operating limit specified in the COLR. These changes to the bases clarify the SPC methodology used at Dresden and how it is used to meet TS 3.11.C. Based on this information, the changes to section 3/4.3.D, 3.E, and 3.F bases are acceptable.

2.6 Primary System Boundary Bases

In sections 3/4.6.E and 3/4.6.F, the licensee proposed to add the following sentence to the middle of the first paragraph:

> SPC methodology determines the most limiting pressurization transient each cycle.

The addition of this statement clarifies the SPC methodology for analyzing the overpressurization event and therefore, is acceptable.

2.7 Power Distribution Limits Bases

The licensee proposed changes to sections 3/4.11.A, 3/4.11.B, and 3/4.11.C, Average Planar Linear Heat Generation Rate, Transient Linear Heat Generation Rate, and Minimum Critical Power Ratio, in order to provide clarification of the SPC methodology for the application of thermal limits. TS 3.11.A requires that all APLHGR for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the COLR. The licensee proposed to replace the first two paragraphs in section 3/4.11.A with insert D to describe the SPC methodology:

> This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the Average Planar Linear Heat Generation Rate (APLHGR) operating limits is based on a loss-of-coolant accident analysis. This analysis is performed using calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50.
>
> The PCT following a postulated loss-of-coolant accident is primarily a function of the initial condition's average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod-to-rod power distribution within the assembly.
>
> The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

The staff finds the replacement of the first two paragraphs in section 3/4.11.A with the above paragraphs acceptable.

TS 3.11.B requires that the Transient Linear Heat Generation Rate shall be maintained such that the Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is less than or equal to 1.0. The licensee proposed to replace the last sentence in the first paragraph of section 3/4.11.B with the following sentences:

> The APRM scram settings must be adjusted to ensure that the LHGR transient limit (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The APRM scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is grater than 1.0.
>
> The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

The second paragraph provides clarification of LCO Action Statements 3.11.B.2 and 3.11.B.3. Therefore the above replacement paragraphs clarifies the SPC methodology and is acceptable.

In section 3/4.11.C, the licensee proposed minor editorial changes to second paragraph. These changes affect the first two sentences and are as follows:

> To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients are analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated are change of flow, increase in pressure and power, positive

reactivity insertion, and coolant temperature decrease.

Furthermore, the fourth paragraph is replaced with insert F which again clarifies the SPC methodology which uses four scram insertion points to calculate MCPR Operating Limit and MCPR Safety Limit:

> MCPR Operating Limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits takes advantage of improved scram insertion rates, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.3.E. The measured scram insertion times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.
>
> For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greatest value of MCPR as given by the rated conditions MCPR limit or the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

The proposed change appropriately reflects the NRC-approved SPC methodology and does not change the current requirement that MCPR meet the limits specified in the COLR. Therefore, the proposed change is acceptable.

2.8 Reactor Core

TS 5.3.A, Fuel Assemblies, provides a description of the fuel assemblies. The licensee proposed to expand this description to be consistent with Improved Standard Technical Specifications, NUREG-1433 Rev. 1, and to better reflect the ATRIUM-9B design. The revised description includes a discussion of the use of water rods or water boxes which is consistent with the SPC fuel design, and replaces "zirconium alloy" with "Zircaloy or Zirlo." The proposed change accurately describes the SPC fuel design, is consistent with NUREG-1433, Rev. 1, and does not affect any current TS requirements. Therefore, the proposed change is acceptable.

2.9 Reactor Coolant System

TS 5.4 describes the design pressure, temperature, and volume of the reactor coolant system. The licensee proposed to relocate the contents of specification 5.4 to the UFSAR. Page 5-6 and Table of Contents page XVI are modified to read, "[INTENTIONALLY BLANK]." This proposed change is consistent with Improved Standard Technical Specifications, NUREG-1433, Rev. 1, and is acceptable.

2.10 Reporting Requirements

TS 6.9 requires that in addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted. TS 6.9.A.6.a(4) describes the MCPR limit in the COLR. The licensee proposed to delete the 20% in the statement "including 20% scram insertion time" to reflect the SPC methodology. The proposed change will state "including scram insertion times." This reflects the current SPC methodology and is acceptable.

TS 6.9.A.6.b lists the analytical methods used to determine the operating limits that are previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports. The licensee proposed to include references to the following topical reports which are used to determine the core operating limits:

(7) XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel, Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

(8) ANF-89-014(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

(9) ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs, Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(10) ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, Advanced Nuclear Fuels Corporation, January 1993.

The licensee also proposed to delete current reference 6, and change reference 7 to reference 6 and current reference 8 to reference 11. The additional topical reports are those used in SPC methodology and have been approved by the NRC for use at Dresden. The staff finds this change acceptable because the use of identified NRC-approved methodologies will ensure that the values for cycle-specific parameters are determined consistent with applicable design bases and safety limits, and assist safe operation of the facility.

3.0 Conclusion

ComEd requested changes to the Dresden Nuclear Power Station, Units 2 and 3, TS which would incorporate NRC approved thermal limit licensing methodology in the list of approved methodologies used in establishing the cycle specific thermal limits. Other minor editorial changes were also proposed. The staff concluded that these TS revisions are compatible with the STS, and SPC methodology. Based on the above, the staff concluded that operation in the proposed manner will not endanger the health and safety of the public and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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