Commonwealth Edison Company
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April 21, 1995

ComEd

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

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Staff Request for Additional Information (RAI)
Regarding the Technical Specification Upgrade Program (TSUP)
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) Meeting between representatives of ComEd (I. Johnson, P. Piet, J. Schrage, et. al.) and the NRC staff (R. Capra, J. Stang, R. Pulsifer, T. Kim), dated March 23, 1995.

(b) P. Piet letter to T. Murley, dated September 15, 1992.

In Reference (a), the NRC staff requested additional information from Commonwealth Edison (ComEd) to support the review and approval of TSUP Sections 2.0, 3/4.11 and 3/4.12. ComEd submitted TSUP Sections 2.0, 3/4.11 and 3/4.12 to the NRC staff on September 15, 1992 (Reference (b)). The purpose of this letter is to respond to the NRC staff's RAI for TSUP Sections 2.0, 3/4.11 and 3/4.12 and supplements the information previously provided in the Reference (b) submittal.

If there are any questions, please contact this office.

Sincerely,

Peter L. Piet
Peter L. Piet
Nuclear Licensing Administrator

Attachment: ComEd Response to NRC Staff Questions on TSUP Sections 2.0, 3/4.11 and 3/4.12

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Signed before me on this 21st day,

of April, 1995.

Mary Jo Yack
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OFFICIAL SEAL
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ATTACHMENT

ComEd Response to NRC Staff Questions on TSUP Sections 2.0, 3/4.11 and 3/4.12

<u>ComEd TSUP RAI Response for Sections 2.0, 3/4.11 and 3/4.12</u>

1. *The information below is being provided in response to NRC staff questions concerning the apparent deviation from current Tech Specs of the proposed TS APLHGR action requirements.*

   Dresden and Quad Cities CTS 3.5.I specify the LCO, Applicability and Action requirement for the Average Planar Linear Heat Generation Rate (APLHGR). These requirements have been incorporated into proposed TS 3.11.A, Power Distribution Limits. The proposed TS 3.11.A, Actions 1 and 2 revise the CTS Action requirement, consistent with BWR-STS requirements.

   When APLHGR exceeds the limiting value (in the Core Operating Limits Report (COLR)) CTS requires the initiation of actions within 15 minutes to restore operation within prescribed limits. This requirement is incorporated into proposed Action 1. This is consistent with BWR-STS.

   If the APLHGR cannot be restored to prescribed limits within 2 hours, CTS requires the reactor to be placed in Cold Shutdown within 36 hours. This requirement is modified in proposed Action 2. Proposed Action 2 requires a reduction in thermal power to less than 25% of rated thermal power within 4 hours.

   The current requirement to place the reactor in cold shutdown would result in unnecessary thermal cycles of the reactor vessel. The proposed Action 2 is consistent with STS requirements, and is a more conservative operating philosophy, and therefore is not a relaxation of CTS.

   As stated in NUREG 1433, the APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are limiting when assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced,the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% rated thermal power when entry into mode 2 occurs. When in mode 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit requirement in mode 2. Therefore, at thermal power levels less than 25% rated thermal power, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO to place the reactor in cold shutdown is not necessary to ensure safety margin to APLHGR limits.

   NUREG 1433 also states that if the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a mode or other specified condition in which the LCO does not apply. To achieve this status, thermal power must be reduced to less than 25% of rated thermal power within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce thermal power to less than 25% rated thermal power in an orderly manner and without challenging plant systems. Reducing power beyond the proposed TS Action requirement (i.e. to cold shutdown within 36 hours) results in unnecessary thermal cycles of the reactor vessel , and unnecessary challenges to safety systems, given that the level of safety provided by the APLHGR limit is minimal below 25% of rated thermal power. Therefore, the proposed TS Action to reduce thermal power to less than 25% within 4 hours, is a conservative operating philosophy, consistent with the level of safety provided by the APLHGR limit.

2. *The information below is being provided in response to NRC staff questions concerning the apparent deviation from current Tech Specs of the proposed TS requirements concerning reactor vessel steam space terminology.*

   Current Quad Cities 1.2.A states "The reactor coolant system pressure as measured by the vessel steam space pressure indicator shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel." The equivalent section for Dresden (Section 1.2) states: "The reactor coolant system pressure shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel."

   Proposed TSUP 2.1.C states: "The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

   APPLICABILITY: Operational Mode(s) 1,2,3 and 4.

   ACTION: With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.4."

   The vessel steam space is equivalent (current Quad Cities wording) to the reactor vessel steam dome (proposed TSUP wording). The proposed wording was chosen to be consistent to current industry standards and does not pose a relaxation of current requirements. There is no deviation from current requirements, and the proposed TSUP wording is consistent to the BWR STS and the Improved Standard Technical Specifications.

   Regarding Safety Limit Applicability, as discussed in Attachment 2, of ComEd's submittal dated September 15, 1992, the present applicability of, "at any time when irradiated fuel is in the reactor vessel," is replaced with Operational Modes 1,2,3 and 4. The proposed operational modes envelope all situations where the reactor vessel could become pressurized. The elimination of Mode 5 is self-evident because Mode 5 (Refueling) is defined to be at a temperature of less than or equal to 140 degrees Fahrenheit with the mode switch in the Shutdown or Refuel Position. In addition, footnote (c) clarifies that Refueling is when there is fuel in the vessel with one or more vessel head closure bolts less than fully tensioned or with the head removed. Therefore, it is not possible in TSUP Mode 5 to achieve pressures significantly greater than atmospheric conditions and the current requirements are not applicable. Therefore, the proposed Safety Limit applicability is acceptable for application to the

Dresden and Quad Cities Nuclear Power Stations.

Regarding Safety Limit Actions, the current requirements are located in Section 6.4 of the Technical Specifications and require the immediate shutdown of the reactor and subsequent internal and external reporting requirements. The proposed requirements allow a period of 2 hours to comply and then subsequently initiate the appropriate reporting requirements. The proposed TSUP requirements allow a period of time to assess, evaluate and choose the safest course of action. The current requirements may in fact be imprudent because no time to pause and assess the situation is provided. Thus, during an event or transient that threatens a plant safety limit, immediate shutdown of the reactor may introduce additional uncertainty into the event. The proposed changes have been shown by industry experience and precedence to provide reasonable assurance that the reactor coolant system pressure boundary integrity can be maintained within the requirements of the Standard Technical Specification and the Improved Standard Technical Specifications. The small time frame (2 hours) is insignificant with respect to overall plant vulnerability, and prudently allows a reasonable time period to assess a situation in which a safety limit may be approached and thus, the proposed changes are appropriate.

3. *The information below is being provided in response to NRC staff questions concerning miscellaneous issues from TSUP Section 3/4.12.*

   A. In TSUP 3/4.12.A, ComEd deviated from STS requirements by specifying a reactor coolant temperature of 212 °F as compared to STS requiring 200 ° F. These requirements are essentially equivalent. 212 °F is the value in the current Dresden and Quad Cities Technical Specifications and the value approved in TSUP Section 1.0 for the Definition of Operational Mode. The difference between 212 ° F and 200 °F results in an insignificant change in water density so the coolant provides the same moderator effect and essentially an equivalent level of safety protection.

   B. In TSUP 3/4.11.B, ComEd deviated from STS guidelines by including reference to TSUP 2.2.A and 3.2.E to avoid duplicating setpoints. The APRM Trip setpoints are already included in TSUP 3.2.E "Control Rod Block Instrumentation - Table 3.2.E, Functional Unit 2" and in TSUP 2.2.A "Reactor Protection System Instrumentation Setpoints - Table 2.2.A-1, Functional Unit 2". Therefore, this proposed deviation from STS requirements is administrative in nature and does not relax current requirements.

   C. Proposed TSUP LCO 3.11.B adopts STS terminology but deviates due to plant/fuel vendor specific terminology/methodology and also provides several enhancements. The enhancements to STS guidelines are made to avoid duplicating setpoints within the proposed technical specifications and to clearly delineate when the actions for the specification are to be implemented.

This specification is provided to require the APRM gain or APRM flow biased scram and rod block trip setpoints to be adjusted when operating under conditions of abnormal power peaking so that acceptable margin to the fuel cladding integrity limits are maintained. Abnormal power peaking is represented when the Maximum Fraction of Limiting Power Density (MFLPD) is greater than the Fraction of Rated Thermal Power (FRTP) for Quad Cities. For Dresden, abnormal power peaking is represented when the Fuel Design Limiting Ratio For Centerline Melt (FDLRC) is greater than 1.0. These thermal limits are different between Dresden and Quad Cities due to the different fuel vendor suppliers for the plants (GE currently supplies fuel to Quad Cities and Siemens currently supplies fuel to Dresden). The different fuel vendors utilize different requirements and terminology to describe the thermal limitations placed upon their respective fuel designs. However, the basis of these thermal limits remain equivalent - to maintain the integrity of the fuel cladding. To maintain the appropriate margin under conditions of abnormal power peaking, either the APRM gain must be adjusted upward or the flow biased neutron flux upscale scram trip and rod block setpoints be reduced. This is accomplished by multiplying the APRM gain or setpoints by a factor that is representative of the reduction in margin to the fuel cladding integrity limits. Adjustment to the scram and rod block setpoint are made by multiplying the setpoint by the inverse of the factor for the APRM gains. This factor will be less than one and thus cause the setpoints to be lowered to maintain the margin. When the reactor is operating with normal peaking (i.e. FDLRC < 1.0 or MFLPD < FRTP) it is not necessary to modify the APRM flow biased scram or rod block setpoints. These requirements are consistent to current requirements and do not reduce existing safety margins at Quad Cities or Dresden.

4. *The information below is being provided in response to NRC staff questions concerning the apparent deviation from current Tech Specs of the proposed TS requirements concerning Safety Limits for Reactor Pressure.*

   The applicable pressure is changed to 785 psig in accordance with the Fuel Vendors Critical Power Ratio correlation and the STS. The current specifications conflict using both 800 psig and 800 psia. The 800 psia value is the correct value, but to be consistent to STS nomenclature, ComEd has chosen to express pressures in units of psig. 800 psia is consistent to the current safety analysis requirements as discussed in Section 4.4 of the Quad Cities UFSAR. Excerpts from Quad Cities' UFSAR are provided for completeness [Note: **BOLD** typeface is used to illustrate reference to 800 psia].

"4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Bases

The design basis for the thermal and hydraulic characteristics of the core is to ensure, in conjunction with the fuel system design, the plant equipment characteristics, the nuclear instrumentation, and the reactor protection system, that no fuel damage will occur during normal operation or operational transients caused by any reasonably expected single operator error or single equipment malfunction. Fuel damage is defined in Section 4.2.1.1. The above design basis is used both for the core design and for the determination of operating limits.

4.4.1.1 Fuel Damage Limits

There are two principal mechanisms which could cause fuel damage in reactor transients, each with a corresponding design limit to ensure that fuel damage would not occur. The fuel damage limit to prevent cladding overheating due to inadequate cooling is conservatively defined as the onset of transition boiling, whereas the fuel damage limit to prevent cladding overstraining due to UO2 pellet expansion is defined as 1% plastic strain of the Zircaloy cladding. These are discussed in detail in Section 4.2.1.1.

These fuel damage limits are also employed in the development of operating limits to control reactor operation. Additional evaluations of the fuel rod thermal and mechanical performance during normal operation and anticipated operational occurrences (AOOs) are performed and are described in Section A.4.2.3 of Appendix A of GESTAR II. An AOO is an incident of moderate frequency, i.e., greater than once per 20 years for a particular reactor.

4.4.1.2 Design Criteria, Operating Basis and Operating Limits

4.4.1.2.1 Design Criteria

The design criteria developed to implement the preceding design bases are discussed in the following paragraphs. **The conditions addressed here correspond to reactor pressures above 800 psia and core flows above 10% of rated.** The cases of reactor pressures at or below 800 psia or core flows at or below 10% of rated are addressed in Section 4.4.4.2.2.

4.4.1.2.1.1 Minimum Critical Power Ratio

The onset of transition boiling results in a decrease in heat transfer from the cladding, and hence an elevated cladding temperature, and the possibility of fuel damage. However, the attainment of critical power, or transition boiling, is not a directly observable event in an operating reactor. Margin to transition boiling is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the fuel assembly power which would produce onset of transition boiling divided by the actual fuel assembly power. The minimum (most limiting) value of this ratio among all fuel assemblies in the core is the minimum critical power ratio (MCPR).

4.4.1.2.1.2 Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from such cracking is gradual and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from thermally caused cladding perforations is just as measurable as that from use-related cracking, the occurrence of such cladding perforations signals a threshold beyond which still greater thermal stresses may cause gross, rather than gradual, cladding deterioration. Therefore, to prevent the possibility of sudden fuel damage, the Fuel Cladding Safety Limit MCPR is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the Fuel Cladding Integrity Safety Limit MCPR is established such that no calculated fuel damage shall result from an AOO. The basis of the values derived for this safety limit is documented in Section A.4.2.3 of Appendix A of GESTAR II.

Because fuel damage by overheating of cladding, defined in Section 4.2.1.1 as onset of transition boiling, is not directly observable, a conservative step-back approach is used to establish this safety limit such that the calculated MCPR for any AOO is no less than the Fuel Cladding Integrity Safety Limit MCPR.

The Fuel Cladding Integrity Safety Limit MCPR has sufficient conservatism to assure that in the event of an AOO initiated from the normal operating condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Fuel Cladding Integrity Safety Limit MCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation. Because the boiling transition correlation is based on a large quantity of full-scale data, there is very high confidence that operation of a fuel assembly at the condition of MCPR equal to the Fuel Cladding Integrity Safety Limit MCPR would not produce boiling transition."

For completeness, excerpts from Dresden's USFAR are provided below:

"4.4.1.3.1 Design Criteria

The design criteria developed to implement the preceding design bases are discussed in Sections 4.4.1.3.1.1 through 4.4.1.3.1.3. **The conditions addressed in these sections correspond to reactor pressures above 800 psia and core flows above 10% of rated.** The cases of reactor pressures at or below 800 psia or core flows at or below 10% of rated are addressed in Section 4.4.4.2.1."

5. *The information below is being provided in response to NRC staff questions concerning miscellaneous issues from TSUP Section 3/4.11.C.*

   The TSUP Specification 3/4.11.C provides limitations and surveillances for MCPR. These are currently contained in CTS 3/4.5.K. The current specification is vague with respect to the surveillance verification of MCPR values. The proposed TSUP 4.11.C reflects the STS and industry approach to establishing surveillance requirements for MCPR. This proposed TS incorporates the need to adjust the MCPR operating limit as a function of average scram time. This need arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. A more detailed summary of this is contained in the Bases for TSUP 3/4.11 and CTS 3.5.K.

   The tau avg term refers to average scram insertion time. At startup, the tau avg is assumed to be 0.86, and the actual tau avg must be determined within 72 hours. The proposed SRs in 4.11.C are more conservative than the current TS, and are consistent with the industry and STS.

6. *The information below is being provided in response to NRC staff questions concerning issues related to Single Loop Operation (SLO) for the Safety Limits.*

   Dresden and Quad Cities conservatively maintain the 0.01 adder for the MCPR limit when in Single Loop Operation to account for increased uncertainties in the core flow and neutron flux monitoring measurements. This deviation from STS guidelines is consistent with current Technical Specification requirements and is consistent with current industry practice (LaSalle County Station). In addition, this is consistent with requirements specified in current Dresden Technical Specification 1.1.A and current Quad Cities Technical Specification 3.6.H.3.a.

7. *The information below is being provided in response to NRC staff questions concerning issues related to Top of Active Fuel (TAF) as it relates to the Safety Limits.*

   Proposed Section 2.1.D, "Reactor Vessel Water Level," incorporates the requirements of the STS Section 2.1.4. Plant specific values for the listed parameters are included to be consistent to the safety analysis for the plant. Proposed Actions and Applicability are changed to match STS guidelines and format.

Dresden and Quad Cities conservatively maintain the deviation from STS 2.1.4 by specifying the Reactor Vessel Water Level Safety Limit as greater than or equal to twelve inches above top of active fuel (TAF). STS 2.1.4 only specifies TAF. The proposed deviation from STS guidelines is consistent to the plant-specific safety analysis which provides a point which can be adequately monitored and provides adequate margin for effective action. This is consistent with current requirements specified in current Dresden Technical Specification 1.1.D and current Quad Cities Technical Specification 1.1.D.

8. *The information below is being provided in response to NRC staff questions concerning issues related to STS Allowable Values as compared to Current Trip Setpoints in the Limiting Safety System Settings.*

   Proposed Section 2.2.A, "Reactor Protection System (RPS) Instrumentation Setpoints," incorporates the requirements of the STS Section 2.2.1. Plant specific values for the listed parameters are included to be consistent to the safety analysis for the plant. Proposed Actions and Applicability are changed to match STS guidelines and format. Proposed Table 2.2.1-1 is equivalent to STS Table 2.2.1-1. In addition, TSUP Table 3.1.A-1 is equivalent to STS Table 3.3.1-1.

   TSUP Section 2.2.A, Action statements, does not adopt STS nomenclature for "Allowable Values." In TSUP, Dresden and Quad Cities have replaced "Allowable Values" with nomenclature that is consistent to the current Technical Specification nomenclature, "Trip Settings". TSUP "Trip Setpoint" is equivalent to the CTS term "Trip Setting." Therefore, TSUP "Trip Setpoint" is equivalent to STS "Allowable Values". The STS requirements for "Trip Setpoint" have not been adopted in TSUP. This deviation from STS guidelines maintains consistency with current Technical Specification requirements and is consistent to the guidance provided in the Improved Technical Specifications. These values/parameters (STS "Trip Setpoint") are more appropriately controlled via the provisions of 10 CFR 50.59. TSUP has included the appropriate OPERABILITY limitation for the functional unit (TSUP "Trip Setpoint") and this value is retained. Details relating to system design, purpose and operation (e.g., STS "Trip Setpoint" - requirements necessary as a result of channel specific drift characteristics) are unnecessary in the LCO and are more appropriate to be controlled administratively by owner-controlled procedures. Changes to the acceptance criteria detailed in procedures will continue to be controlled by the provisions of 10 CFR 50.59.

9. *The information below is being provided in response to NRC staff questions concerning issues related to the splitting of the Flow-biased Neutron Flux settings in the Limiting Safety System Settings.*

   TSUP Table 2.2.A-1, Item 2.a for Setdown Neutron Flux - High is equivalent to the current Technical Specification 2.1.A.2. STS terminology ("Neutron Flux - Upscale, Setdown) has been reworded from STS format to be consistent to existing plant terminology and current Technical Specification requirements. The APRM trip remains at 15% of rated thermal power. The specification for TSUP's Flow Biased Neutron Flux - High, deviates from STS requirements to be consistent to the current Technical Specification requirements. TSUP Table 2.2.A-1, Item 2.b is changed as follows: 1) STS terminology is modified to be consistent to the current Technical Specification requirements; 2) the section is separated into two categories depending on the number of recirculation pumps running. The setpoints remain the same as CTS but deviate in some instances from STS terminology in order to maintain the CTS requirements. STS terminology ("Flow Biased Simulated Thermal Power - Upscale") is equivalent to TSUP ("Flow Biased Neutron Flux - High") and provides protection for the same requirement - the fuel cladding integrity Safety Limit. STS terminology ("High Flow Clamped") is equivalent to TSUP ("High Flow Maximum") and provides protection for the same requirement - the fuel cladding integrity Safety Limit. Splitting the requirements for single (SLO) and dual (DLO) recirculation loop operation enhances STS requirements by providing greater clarity to plant operations personnel, as compared to STS guidance, while maintaining equivalent margins to the current safety analysis and Technical Specifications for Dresden and Quad Cities. The STS requirements for SLO and DLO are not explicitly delineated.

10. *The information below is being provided in response to NRC staff questions concerning issues related to the APRM Inop requirements in the Limiting Safety System Settings.*

The current APRM Inoperative specification is moved to proposed Table 2.2.A-1 in accordance with STS guidelines. Quad Cities current specifications have the APRM Downscale trip in Table 3.1-3, "Reactor Protection System (Scram) Instrumentation Requirements Run Mode." This requirement (APRM Downscale) was not retained in STS. The current action associated with this requirement is that "The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high." Removal of the APRM/IRM companion scram eliminates the APRM downscale scram which occurs in the Run mode with the simultaneous IRM scram that occurs with IRMs "high" or inoperable. This requirement does not enhance safety and has been previously approved by the NRC staff (Amendment Nos. 100/96 by letter dated August 24, 1988) for deletion from the Dresden Technical Specifications. The proposed changes clarify the intent of the current Quad Cities specification by clearly defining the scram functions needed to be operable in each mode of operation and do not involve any modification of the reactor protection system circuitry. Since both the APRM scram and the Control Rod Block actuation functional units are required by other sections in TSUP for operability and surveillance testing, there is reasonable assurance that they will perform their protective functions when needed. Overlap between the IRMs and APRMs is not affected. The IRM Inoperative Scram 'N/A' has been added based upon the plant design and to ensure consistency in function and format to current industry Technical Specification practice (e.g., Dresden, Perry and Hope Creek Technical Specifications). This additional requirement deviates from STS guidance but is merely an administrative enhancement to the Technical

Specification requirements and has been added to maintain consistency in format to the TSUP requirements for APRMs which include an inoperative statement (STS Table 2.2.1-1, Item 2.d).

11. *The information below is being provided in response to NRC staff questions concerning issues related to the Turbine Low EHC Pressure and Turbine Control Valve Fast Closure Scram.*

Current Technical Specification 2.1.I, Turbine Low EHC Pressure scram function is moved to proposed TSUP 2.2, Table 2.2.A-1. The proposed specification is renamed Turbine EHC Control Oil Pressure - Low. STS requirements for this setpoint do not exist. The setpoint of 900 psig remains unchanged from the current Technical Specifications. As such, TSUP provides equivalent protection to existing requirements. Since existing Technical Specification requirements are maintained and ensure existing plant safety margins, there is no reduction in plant safety proposed by these changes.

Current Technical Specification 2.1.F, EHC Fluid Pressure scram function is moved to proposed TSUP 2.2, Table 2.2.A-1. For the Turbine Control Valve Fast Closure, the proposed TSUP terminology deviates from STS guidelines. The current practices at Dresden and Quad Cities do not include the STS terminology for "Turbine Control Valve fast Closure, Trip Oil Pressure - Low." The proposed TSUP RPS setpoint (TSUP Table 2.2.A-1, Item 11, "Turbine Control Valve Fast Closure") maintains the current Technical Specification requirements listed in Section 2.1.F (STS (Table 2.2.1-1, Item 10). This proposed deviation from STS guidelines is administrative in nature and does not affect the safe operation of the plants because the same level of protection, to ensure that the anticipatory scram that occurs during this configuration occurs prior to the rapid increase in pressure and neutron flux resulting from the fast closure of the turbine control valve due to a load rejection and subsequent failure of the bypass valves, is provided.

12. *The information below is being provided in response to NRC staff questions concerning issues related to proposed TSUP Surveillance Requirements in 3/4.11.*

ComEd has proposed new SRs in TSUP 3.4.11.A, 3.4.11.B, 3.4.11.C and 3.4.11.D (also 3.4.11.E for Dresden only) that require additional verification of Thermal Limits following increases in THERMAL POWER of greater than 15% RATED THERMAL POWER and when operating on a LIMITING CONTROL ROD PATTERN. These changes are consistent to STS requirements for Thermal Limits and conservatively ensure that Thermal Limits are calculated on an increased frequency when significant power changes occur because such changes in power may significantly affect power distribution levels in the core.

13. *The information below is being provide in response to NRC staff questions concerning proposed TSUP 3.12.A regarding performing the low power PHYSICS TESTS at 1% RATED THERMAL POWER as compared to CTS 3.7.A.2 requirements specifying 5 MWth.*

As discussed in Reference (b), the proposed specification implements the current provisions contained in CTS 3.7.A.2 for both Dresden and Quad Cities. The specification allows the primary containment integrity requirements to be suspended for the purpose of performing low power physics tests with thermal power less than 1% of rated thermal power and reactor coolant temperature is less than 212 °F. CTS requirements specify the reactor power to be less than 5 MWth (~0.2% power). The proposed TSUP requirements are based on STS 3/4.10.1. 1% of RATED THERMAL POWER is a more measurable quantity as compared to the CTS 5 MWth requirements and can be more accurately differentiated from core decay heat. The proposed requirements have been shown based on industry precedence and experience to provide an adequate level of protection in assuring that primary containment integrity is maintained during low power PHYSICS TESTS. Proposed TSUP 3/4.12.A provides enhanced guidance to site operations personnel by including specific actions, applicability and surveillances not included in the CTS to ensure potential degraded conditions associated with primary containment integrity associated with low power PHYSICS TESTS are appropriately addressed. The proposed requirements (1% of rated thermal power) are also consistent to the LaSalle County, Hope Creek, Perry Station and Grand Gulf Technical Specifications.

14. *The information below is being provide in response to NRC staff questions concerning the cross-reference between current Technical Specifications and the proposed TSUP for TSUP Sections 2.0, 3/4.11 and 3/4.12.*

The current Technical Specification (CTS) requirements in Section 1.1/2.1, 1.2/2.2 for both Dresden and Quad Cities, Dresden Table 3.1.1, and Quad Cities Tables 3.1-1, 3.1-2 and 3.1-3 are encompassed within TSUP Section 2.0. The action statements of Dresden CTS Table 3.1.1 and Quad Cities Tables 3.1-1, 3.1-2 and 3.1-3 are addressed in the ComEd response for TSUP 3/4.1. TSUP Section 2.0 is based on STS 2.0, "Safety Limits and Limiting Safety System Settings." The changes to CTS 1.1/2.1 and 1.2/2.2 include clarifying Applicability and Actions based on STS Section 2.0.

The current Technical Specification (CTS) Safety Limits are currently divided into 1.1, "Fuel Cladding Integrity," and 1.2, "Reactor Coolant System." These requirements are encompassed within proposed TSUP 2.1, "Safety Limits," which are based on STS 2.1. As previously discussed, the CTS requirements (1.1.C) related to the Power Transient limits have not been retained within proposed TSUP 2.1.

The CTS Limiting Safety System Settings (LSSS) are also currently divided into 2.1, "Fuel Cladding Integrity," and 2.2, "Reactor Coolant System." These requirements are encompassed where applicable, within proposed TSUP 2.2, "Limiting Safety

System Settings," which are based on STS 2.2. Incorporated into TSUP 2.2 is Table 2.2.A-1, "Reactor Protection System Instrumentation Setpoints," which is based on STS Table 2.2.1-1.

To be consistent to the format of STS Table 2.2.1-1, ComEd has included information from Dresden Table 3.1.1 and Quad Cities Tables 3.1-1, 3.1-2 and 3.1-3 into proposed TSUP Table 2.2.A-1. These changes are administrative in nature as the appropriate Trip Setpoints are maintained and the relocation into Table 2.2.A-1 ensures consistent guidance is provided to site operations personnel for the Reactor Protection System Instrumentation. The RPS Trip Setpoints incorporated within the LSSS more appropriately ensures that plant Safety Limits are protected. Proposed TSUP Table 3.1.A-1 and 4.1.A-1, based on STS Table 3.3.1-1 and 4.3.1-1, respectively, encompass the remaining applicable portions of the RPS system instrumentation requirements.

CTS LSSS requirements regarding thermal limits (FDLRC for Dresden fuel-vendor specific and FRP/MFLPD for Quad Cities), ECCS low water level, Main Steamline Pressure and Safety Valve settings have been relocated to TSUP Section 3/4.11.B and 3/4.11.E [FDLRC], TSUP 3/4.2.B [ECCS Low Water Level], TSUP 3/4.2.A [Main Steamline Pressure], and TSUP 3/4.6.E [Safety Valves]. These values are not reactor protection system instrumentation and therefore, are inappropriate for inclusion within TSUP Table 2.2.A-1.