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March 9, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Draft Beyond Scope Item Safety Evaluations for the Conversion to Improved Standard Technical Specifications

References: (1) Letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Convert to Improved Standard Technical Specifications," dated March 3, 2000

(2) Letter from S. N. Bailey (U. S. NRC) to O. D. Kingsley, "Draft Beyond Scope Item Safety Evaluations for the Conversion to Improved Standard Technical Specifications for Dresden Nuclear Power Station Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2," dated March 6, 2001

Commonwealth Edison (ComEd) Company, currently Exelon Generation Company (EGC), in a letter dated March 3, 2000 (Reference 1) proposed changes to the Technical Specifications (TS) of Facility Operating License Nos. DPR-19, DPR-25, NPF-11, NPF-18, DPR-29, and DPR-30 for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2,

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and Quad Cities Nuclear Power Station, Units 1 and 2. The NRC issued the draft Beyond Scope Item Safety Evaluations (SEs) supporting the conversion to the Improved Technical Specifications (Reference 2) and requested that comments be provided by March 9, 2001.

We have completed our review of the draft Beyond Scope Item SEs and specific comments on the draft SEs are attached.

Should you have any questions concerning this letter, please contact Mr. J. V. Sipek at (630) 663-3741.

Respectfully,

R. M. Krich
Director-Licensing
Mid-West Regional Operating Group

Attachment: Comments on Draft Beyond Scope Item Safety Evaluations

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - LaSalle County Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

ATTACHMENT

Comments on Draft Beyond Scope Item Safety Evaluations

DRAFT SAFETY EVALUATION INPUT

DRESDEN NUCLEAR POWER PLANT, UNITS 2 AND 3

BEYOND SCOPE ITEMS

Six Hour Delay to Perform SR (ITS 3.3.3.1, DOC L.2)

For the post-accident monitoring (PAM) instrumentation, a note has been added to the Surveillance Requirements allowing a 6 hour delay from entering into the associated Conditions and Required Actions for a channel that is placed in an inoperable status solely for performance of SRs. For the PAM instrumentation, this is only allowed provided the other channel in the associated function is operable. The loss of one PAM channel is acceptable in this case since another channel is operable to monitor the required function. The short period of time (6 hours) in this condition will have no appreciable impact on risk. Also, upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable Condition must be entered and Required Actions taken. The staff finds this change to be acceptable.

Rod Worth Minimizer Operability Requirements (ITS 3.3.2.1, DOC L.4)

The rod worth minimizer (RWM) acts to enforce boundaries for control rod withdrawal sequences in order to minimize control rod worths during startup thus, mitigating the consequences of a control rod drop accident (CRDA). The licensee proposes to reduce the power level that the RWM must be operable from less than 20 percent to less than or equal to 10 percent of rated thermal power. (or equal to)

The proposed reduction in RWM power level operability is, in part, based on Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", Revision 8, Amendment 17. The NRC staff approved this topical report for referencing by a safety evaluation dated December 27, 1987. Additionally, the licensee stated that reducing the RWM operable power level is acceptable because a Siemens Power Corporation (SPC) analysis performed for the SPC fuel in Dresden Units 2 and 3, finds that the consequences of a CRDA are mitigated above 10 percent power.

The NRC staff stated in its safety evaluation of December 27, 1987, that the 20 percent limit for RWM operability was originally required because of analytical uncertainties and that current analyses show that a RWM operable power level limit of 10 percent was acceptable. The NRC staff has reviewed the licensee's requested change and finds that reducing the RWM operable power level limit to 10 percent is acceptable based on the staff's safety evaluation of December 27, 1987.

Deletion of Modes 3 and 4 in Reactor Protection System (RPS) Electric Power Monitoring Assembly (EPA) Applicability (ITS 3.3.8.2, DOC L.1)

The operability requirements for the RPS EPAs is changed to delete the requirement for them to be operable in Modes 3 and 4. The EPAs provide a regulated power supply for the RPS instrumentation electrical buses. RPS EPAs are provided to isolate the RPS bus from the motor generator set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency condition. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions and forms an important part of the primary success path of the essential safety circuits. This change is made to establish consistent requirements between RPS instrumentation (LCO 3.3.1.1) and ITS 3.3.8.2 (RPS Electrical Power Monitoring Assemblies). In addition, conforming changes are made to require channel functional testing prior to entry into Mode 2 from Modes 3 or 4.

The only essential equipment required to be operable in Modes 3 and 4 that are powered from RPS buses are the RPS logic and the scram pilot valve solenoids. With the unit in Mode 3 or 4, all control rods are fully inserted and will remain inserted because the Reactor Mode Switch, while in the Shutdown position, enforces a control rod withdrawal block. Thus, it is not necessary for the EPAs to be operable in Modes 3 and 4. However, ITS 3.10.2 (Single Control Rod Withdrawal—Hot Shutdown) and ITS 3.10.3 (Single Control Rod Withdrawal—Cold Shutdown) provide exceptions to the restrictions on control rod withdrawal in Modes 3 and 4. To address these two exceptions, ITS 3.10.2 and ITS 3.10.3 include operability requirements for RPS instrumentation (ITS 3.3.1.1), control rods (ITS 3.9.5), and EPAs (ITS 3.3.8.2). The staff finds this change to be acceptable because the RPS EPAs will be required to be operable when necessary to support RPS operability.

Replace Required Actions to Trip a Recirculation Pump with Actions to Declare the Recirculation Loop Not in Operation (ITS 3.4.1, DOC L.2)

The CTS requirement to trip a recirculation pump within 2 hours when the speed between pumps is mismatched (i.e. flows mismatched) is replaced with (1) a requirement (ITS 3.4.1 ACTION B) to declare the loop with the low flow "not in operation" if the flows remain mismatched after 2 hours, and (2) a caution to operators for cases where flow mismatches are large. While a shutdown of the loop may be preferred under some conditions, declaring a pump not in operation will ensure the proper actions are taken in accordance with the single loop analysis.

In most instances flow mismatches can be readily alleviated. However, in cases where large flow mismatches occur, low flow, or reverse flow can occur in the jet pumps of the low flow loop, causing jet pump vibration. If zero or reverse flow is detected, the Bases state the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump. Should a LOCA occur with one recirculation loop not in operation, the core flow coast down and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to declare the low flow loop "not in operation." Once the declaration has been made, the appropriate actions for single loop operation must be taken in accordance with ITS 3.4.1 (CTS 3.6.A.1). It is acceptable to establish the single loop analysis requirements of the LCO as they are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values because this satisfies the initial conditions of the accident analysis; therefore, the staff finds this change acceptable.

Changing the Frequency for Monitoring Primary Containment Sump Flow Rate (ITS 3.4.4, DOC L.1)

CTS 4.6.H.2 requires measurements of primary containment sump flow rate to quantify RCS unidentified leakage, total leakage, and unidentified increase leakage to be made at least once per 8 hours, not to exceed 12 hours. The surveillance frequency has been changed to 12 hours in ITS SR 3.4.4.1. This time interval is consistent with the guidance given in Generic Letter (GL) 88-01, Supplement 1, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," which found that, "monitoring reactor coolant system (RCS) leakage every 4 hours creates an unnecessary administrative hardship for plant operators. Thus, RCS leakage measurements should be taken at least once per shift, not to exceed 12 hours." This change allows the 25% extension specified in ITS 3.0.2 to be applied to the current 12 hour surveillance interval. As such, the maximum interval has been extended from 12 hours to 15 hours. The proposed extension to the surveillance interval is acceptable since the probability of a pipe break occurring in the primary containment during the extension period is small and the vast majority of the surveillances are completed with no indication of excessive RCS operational leakage. Furthermore, the leak detection instrumentation will remain available during the extension period such that excessive RCS leakage will continue to be alarmed in the main control room, and a change in sump flow will continue to be indicated on the ~~drywell sump pump flow integrators~~ control room leak rate recorder. The staff finds a 12 hour surveillance interval to be acceptable and consistent with the guidance in GL 88-01, Supplement 1.

More Restrictive Shutdown Requirements for LPCI Inoperability (ITS 3.5.1, DOC M.1)

CTS 3.5.A.2 defines the low pressure coolant injection (LPCI) subsystem as being comprised of four LPCI pumps and a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel. ITS 3.5.1 will ~~define~~ require two ~~LPCI~~ subsystems to be OPERABLE, each defined in the Bases as consisting of two motor driven pumps, piping, and valves capable of transferring water from the suppression pool to the RPV via the "selected" recirculation loop. CTS 3.5.A Action 2.b, which allows the entire LPCI System to be inoperable for 7 days, has been modified to allow only one LPCI subsystem to be inoperable (ITS 3.5.1 Condition B) or one LPCI pump in each LPCI subsystem to be inoperable (ITS 3.5.1 Condition C) for 7 days, or both LPCI subsystems to be inoperable for 72 hours (ITS 3.5.1 Action D). These changes represent additional restrictions on plant operation. The staff finds these changes to be acceptable.

Change in Number of Automatic Depressurization System Valves (ITS 3.5.1 DOC L.1)

The automatic depressurization system (ADS) is designed to depressurize the reactor to permit the low pressure coolant injection (LPCI) or core spray subsystem to cool the reactor during a small break loss of coolant accident (LOCA) if the high pressure coolant injection (HPCI) system fails or is unable to maintain required water level in the reactor. The Dresden ADS system consists of five valves (four relief valves and one safety/relief valve). Qualification of the accumulator for the safety/relief valve to perform the ADS function has not been demonstrated, therefore, the safety/relief valve is not credited in the safety analyses.

Only four ADS valves were assumed operable in the Dresden LOCA analyses. One ADS valve of the four valves modeled in the LOCA analyses was assumed to fail for the single failure evaluation resulting in three valve operation credited. The analyses demonstrates that adequate core cooling is provided during small break LOCA and simultaneous battery failure with two of the five ADS valves out-of-service. In order to meet the single failure criteria, the revised TS requires four ADS valves to be operable. It is specified in the revised TS 3.5.1 Bases that the safety/relief valve can not be used to satisfy the ADS valve operability requirements. This ensures that all four relief valves associated with the ADS system will be required to be operable. The analyses in support of the TS change were performed using approved methods, and the licensee has demonstrated that all applicable acceptance criteria continue to be met with the proposed ADS valve operability requirements. Therefore, the staff finds the change to be acceptable.

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Change the Acceptance Criteria for Excess Flow Check Valve Tests (ITS 3.6.1.3.8, DOC L.3)

The requirement in CTS 4.7.D.4 that each excess flow check valve (EFCV) must check flow has been deleted. ITS 3.6.1.3.8 requires, instead, that EFCVs actuate to their isolation position (i.e., closed) on an actual or simulated instrument line break signal. The requirements for the EFCVs are provided in 10 CFR 50 Appendix A, General Design Criteria 55 and 56, and are further detailed in Regulatory Guide (RG) 1.11. These state that there should be a high degree of assurance that the EFCVs will close or be closed if the instrument line outside containment is lost during normal reactor operation, or under accident conditions. The Instrument Line Break Analysis in the Dresden UFSAR, Section 15.6.2, assumes both the EFCV and the manual block valve are unavailable, i.e., fail to close; and the accident is terminated by cooling down the plant and closing the manual valve after the plant is shutdown and depressurized. Since the actual leakage is not an assumption of the accident analysis (the leakage is assumed to be the maximum allowed through the broken line), the leakage limit criteria (i.e., check flow) has been deleted. Further, the proposed change ensures that the RG 1.11 criterion ~~that there is~~ which requires a high assurance that the EFCVs will close will be met. Therefore, the staff finds the change to be acceptable.

Change in Required Spent Fuel Pool Water Level (ITS 3.7.8, DOC M.1)

CTS 3.10.H requires that the spent fuel storage pool water level be maintained at a level of greater than or equal to 33 feet. ~~(In the conversion, this technical specification will be renumbered to ITS 3.7.8.)~~ The licensee proposes to modify the requirement for the ITS to maintain the spent fuel storage pool water level at greater than or equal to 19 feet over the top of the irradiated fuel. The licensee states that this change results in an increase in the water level by approximately 9 inches. No other changes to the spent fuel storage or pool support systems are proposed. The staff finds that this is a more restrictive requirement for fuel movement and therefore, is an acceptable modification to the technical specifications.

Change in Voltage During Diesel Generator Tests (ITS 3.8.1, DOC M.5)

Currently, the allowable emergency diesel generator (EDG) voltage tolerance in the CTS SRs is 4160 ± 420 volts. The licensee has proposed to change the allowable voltage tolerance to 4160 ± 208 volts. The change will provide more restrictive EDG allowable voltage limits (i.e., from ± 10% to ± 5%) during surveillance testing. The licensee stated that the current voltage tolerance may allow EDG operation at the lower end of the voltage limits, which ~~could~~ may not support operation of ECCS loads within design voltages. Reducing the EDG allowable voltage limits to ± 5% will support operation of all required EDG loads within the design voltage ranges for ECCS loads. The staff concludes that the change is conservative and acceptable.

NOTE TO NRC: The deleted sentence is not covered by DOC M.1, but by DOC A.1

DRAFT SAFETY EVALUATION INPUT

LASALLE COUNTY STATION, UNITS 1 AND 2

24 MONTH CONVERSION ITEMS FOR SECTION 3.8

TS 3.8.1 AC Sources - Operating

SR 3.8.1.8

Division 1 and 2

This SR requires transfer of each 4.16 kV emergency bus power supply from the normal offsite circuit to the alternate offsite circuit to demonstrate the operability of the alternate circuit.

SR 3.8.1.9

This SR verifies each required diesel generator (DG) rejects a load greater than or equal to its associated single largest post-accident load and following load rejection, the specified frequency is achieved.

SR 3.8.1.10

This SR verifies each required DG does not trip and the specified voltage is maintained during and following a load rejection of the specified load.

SR 3.8.1.11

This SR verifies on an actual or simulated loss of offsite power signal: a) de-energization of emergency buses, b) load shedding from emergency buses for Division 1 and 2 only, and c) DG auto-starts from standby condition and 1) energizes permanently connected loads in the specified time, 2) energizes auto-connected shutdown loads, 3) maintains the specified steady state voltage, 4) maintains the specified steady state frequency, and 5) supplies permanently connected and auto-connected shutdown loads for greater than the specified time.

SR 3.8.1.12

This SR verifies on actual or simulated Emergency Core Cooling System (ECCS) initiation signal each required DG auto-starts from standby condition and: a) within the specified time after auto-start, achieves the specified voltage and frequency, b) achieves the specified steady state voltage and frequency, and c) operates for the specified minimum time.

SR 3.8.1.19

This SR verifies on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a) de-energization of emergency buses; b) load shedding from emergency buses for Division 1 and 2 only; and c) DG auto-starts from standby condition and; 1) energizes permanently connected loads in less than the specified time, 2) energizes auto-connected emergency loads, 3) maintains steady state voltages specified, 4) maintains specified frequency, and 5) supplies permanently connected and auto-connected emergency loads for greater than the specified time.

steady state

SR 3.8.1.13

This SR verifies each required DG's automatic trips are bypassed on an actual or simulated ECCS initiation signal except: a) engine overspeed, and b) generator differential current.

SR 3.8.1.14

This SR verifies each required DG operates greater than or equal to 24 hours: a) for 2 hours greater than the specified load, b) for the remaining hours of the test at the specified load.

SR 3.8.1.15

This SR verifies that, starting from a hot condition, each required DG starts and achieves: a) the required voltage and frequency in the specified time, and b) the specified voltage and frequency.

steady state

SR 3.8.1.16

This SR verifies each required DG: a) synchronizes with offsite power while loaded with emergency loads upon a simulated restoration of offsite power, b) transfers loads to offsite power source, c) and returns to ready-to-load operation.

SR 3.8.1.17

This SR verifies with a required DG operating in test mode and connected to its bus: a) For Division 1 and 2 DGs, an actual or simulated ECCS initiation signal overrides the test mode by returning DG to ready-to-load operation; and b) for Division 3 DG, an actual or simulated DG overcurrent trip signal automatically disconnects the offsite power source while the DG continues to supply normal loads.

SR 3.8.1.18

This SR verifies the interval between each sequenced load block for Division 1 and 2 DGs only, is within the specified design interval for each time delay relay.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- SR 3.8.1.2 requires that each DG be tested for operability once every 31 days. This testing, which is not being changed, will provide prompt identification of any substantial DG degradation or failure.

- SR 3.8.1.7 requires that each DG be fast start tested once every 184 days. This test, which is not being changed, will provide prompt identification of any substantial DG degradation or failure.

- DGs are not operated outside of the monthly operability tests in order to minimize wear related degradation.

- DG attributes subject to degradation due to aging, such as fuel oil quality, are subject to its requirements for replenishment and testing.

- An evaluation of known failures did not identify any time-based elements that would invalidate the conclusion that the increased operating cycle will have a small, if any, impact on system reliability.

- The licensee's review of the surveillance test history did not identify any test failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to a 24-month operating cycle.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed changes is small and, therefore, the changes are acceptable.

<u>TS 3.8.4</u>   <u>DC Sources - Operating</u>

<u>SR 3.8.4.3</u>

This SR verifies battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that degrades battery performance.

<u>SR 3.8.4.4</u>

This SR states to remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.

<u>SR 3.8.4.5</u>

This SR verifies battery connection resistance is less than the value specified for inter-cell connections and terminal connections.

<u>SR 3.8.4.6</u>

This SR verifies each required battery charger supplies: a) the specified amps and volts for greater than the required time for Division 1 and 2 125 V battery chargers; and b) the specified amps and volts for greater than the required time for Division 3 125 V battery chargers; and c) the specified amps and volts for greater than the required time for the 250 V battery charger.

SR 3.8.4.7

This SR verifies battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The design, in conjunction with the technical specification requirements which limit the extent and duration of inoperable DC sources, provides substantial redundancy in DC sources.

- Battery parameters such as float voltage, electrolyte level, and specific gravity are monitored during the operating cycle to verify battery operability and will provide prompt identification of any substantial battery or battery charger degradation or failure. As an example, SR 3.8.4.1, which is performed once every 7 days, verifies that battery terminal voltage on float charge is within limits.

- Batteries are not discharged except for the performance of the operating cycle test demonstrations of operability. Therefore, there is minimal risk of age-related degradation.

- SR 3.8.4.2, which is performed once every 92 days, requires monitoring for visible corrosion at battery terminals and connectors. These examinations will provide prompt identification of any substantial battery degradation.

- The licensee's review of surveillance test history did not identify any test failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to a 24-month operating cycle.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed changes is small and, therefore, the changes are acceptable.

BEYOND SCOPE ITEMS

Six Hour Delay to Perform SR (ITS 3.3.3.1, DOC L.2, ITS 3.3.3.2, DOC L.2, and ITS 3.4.7, DOC L.3)

For the post-accident monitoring (PAM) instrumentation, the RCS leakage detection system (LDS) instrumentation, and the remote shutdown system (RSS) functions, a note is added to the Surveillance Requirements that allows a 6 hour delay from entering into the associated Conditions and Required Actions for a channel placed in an inoperable status solely for performance of SRs. For the PAM instrumentation, the 6-hour allowance only applies provided the other channel in the associated function is operable. For LDS instrumentation, the allowance only applies provided the other required LDS instrumentation is operable. The loss of one PAM detection channel or a channel of LDS instrumentation during required testing is acceptable because during these tests another channel is operable to monitor the associated parameter. For the RSS, the short time period (6 hours) does not significantly reduce the probability of properly monitoring the parameters, when necessary. Thus, the short period of time (6 hours) in this condition will have no appreciable impact on risk. Also, upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable Condition must be entered and Required Actions taken. The staff finds the changes to be acceptable.

Specific Instrument Listings in the LCO are Relocated (ITS 3.3.3.2, DOC [illegible])

The CTS 3.3.7.4, CTS 3.3.7.4 Action a, CTS 4.3.7.4, CTS Table 3.3.7.4-1, and CTS Table 4.3.7.4-1 listings of specific equipment (instruments) included as RSS instruments ~~give~~ provide details relating to system design and operation that are not necessary in the LCO and are relocated to the TRM. ITS 3.3.3.2 requires the RSS functions to be operable, and Surveillance Requirements ensure the required instruments are properly tested. The Bases discusses the types of functions required for the operability of the RSS and refers to the TRM for the specific instrument function list. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS. The TRM will be incorporated by reference into the LaSalle UFSAR. Changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59. Therefore, the staff finds this change to be acceptable.

Changing the Frequency for Monitoring Primary Containment Sump Flow Rate (ITS 3.4.5, DOC L.1)

CTS 4.4.3.2.1 requires ~~measurements of primary containment sump flow rate to quantify~~ RCS unidentified leakage, total leakage, and unidentified increase leakage to be ~~made at least~~ determined, on average once per 8 hours, not to exceed 12 hours. The surveillance frequency has been changed to 12 hours in ITS SR 3.4.5.[illegible]. This time interval is consistent with the guidance given in Generic Letter (GL) 88-01, Supplement 1, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," which found that, "monitoring reactor coolant system (RCS) leakage every 4 hours creates an unnecessary administrative hardship for plant operators. Thus, RCS leakage measurements should be taken at least once per shift, not to exceed 12 hours." This change essentially allows the 25% extension specified in SR 3.0.2 to be applied to the current 12 hour surveillance interval. As such, the maximum interval has been extended from 12 hours to 15 hours. The proposed extension to the surveillance interval is acceptable since the probability of a pipe break occurring in the primary containment during the extension period is small and the majority of the surveillances are completed with no indication of excessive RCS operational leakage. Furthermore, the leak detection instrumentation will remain available during the extension period such that excessive RCS leakage will continue to be alarmed in the main control room and a change in sump flow will continue to be indicated on the ~~drywell sump pump flow integrators~~ control room leak rate recorder. Thus, the staff finds a 12 hour surveillance interval acceptable and consistent with the guidance in GL 88-01, Supplement 1.

Change to Automatic Depressurization System Minimum Operability Pressure (ITS 3.5.1, DOC L.4)

Not available at this time.

Administrative Means of Verifying Air Lock Door Position (ITS 3.6.1.2, DOC L.4)

ITS 3.6.1.2 includes a Note which allows administrative means to be used to verify the position of a locked, closed air lock door that is in a high radiation area or an area with limited access due to inerting. The air lock door is initially verified to be in the proper position and, access to it is restricted during operation due to the high levels of radiation or due to the containment being inerted. Therefore, the probability of misalignment of the air lock door is small. Eliminating the physical verification of doors in areas of high radiation and inerting removes a risk to personnel safety and is consistent with the As-Low-As-Reasonably-Achievable (ALARA) practices. Further, the staff has approved similar allowances for primary containment isolation valves. Therefore, the staff finds the change to be acceptable.

Change the Acceptance Criteria for Excess Flow Check Valve Operability Tests (ITS 3.6.1.3, DOC L.9)

The requirement in CTS 4.6.3.4 that each excess flow check valve (EFCV) must check flow has been deleted. ITS SR 3.6.1.3.9 requires, instead, that EFCVs actuate to their isolation position (i.e., closed) on an actual or simulated instrument line break signal. The requirements for the EFCVs are provided in 10 CFR 50 Appendix A, General Design Criteria 55 and 56, and are further detailed in Regulatory Guide (RG) 1.11. These state that there should be a high degree of assurance that the EFCVs will close or be closed if the instrument line outside containment is lost during normal reactor operation, or under accident conditions. The Instrument Line Break Analysis in the LaSalle UFSAR, Section 15.6.2, assumes both the EFCV and the manual block valve are unavailable, i.e., fail to close, and the accident is terminated by cooling down the plant. ~~and closing the manual valve after the plant is shutdown and depressurized.~~ Since the actual leakage is not an assumption of the accident analysis (the leakage is assumed to be the maximum allowed through the broken line), the leakage limit criteria (i.e., check flow) has been deleted. Further, the proposed change ensures the RG 1.11 criterion that there is a high assurance that the EFCVs will close will be met. Therefore, the staff finds the change to be acceptable.

Change in Voltage Criteria for 250 Volt Battery Test (ITS 3.8.4, DOC M.3)

CTS 4.7.3.d.1 (.d) requires that the overall battery voltage be verified once every 7 days to be greater than or equal to 250 volts, with no requirement of the status of the battery (i.e., on float, open circuit, equalizing, charging, or discharging state). CTS 4.7.3.d.2 requires the voltage of each connected battery to be verified greater than or equal to 250 volts under float charge every 92 days. ITS SR 3.8.4.1 will require a verification every 7 days that the battery voltage is greater than or equal to 256 volts on a float charge. ~~The addition of a float charge requirement to the 7 day verification is considered an administrative change, and is the same as the float charge requirement of the 92 days interval. Verifying the battery terminal voltage on float charge ensures the effectiveness of the charging system, and is acceptable.~~ The change in the required battery voltage to greater than or equal to 256 volts is based on 2.20 volts/cell. This change is an additional restriction on plant operation. The staff finds this change to be acceptable.

NOTE TO NRC: These two sentances are not related to DOC M.3, but to DOC A.7.

Allowance for Performance Discharge Test (ITS 3.8.4, DOCs L.4 and L.5)

NOTE TO NRC: This sentance has been moved to the beginning of the paragraph and modified as shown.

In Note 1 of the STS Surveillance Requirement (SR) 3.8.4.7, it is stated that the modified performance discharge test of a battery may be performed in lieu of service test once per 60 months. The licensee proposes to revise the CTS note to allow the modified performance discharge test be performed at a frequency of 24 months. Further CTS 4.8.2.3.2.e states, "... Once per 60 month interval this performance discharge test may be performed in lieu of the battery service test". The note to ITS SR 3.8.4.7 will allow a modified performance discharge test to be substituted for the service test at anytime, instead of once every 60 months. This change will allow the licensee to perform the modified performance test of LaSalle batteries at every refueling cycle.

IEEE 450 - 1995, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid batteries for Stationary Applications," states that, "it is permissible to perform a modified performance test if the test's discharge rate envelopes the duty cycle of the service test". The modified performance test normally consists of a simulated duty cycle with two rates: the 1 minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test. To ensure that the modified performance discharge test completely envelopes the service test duty cycle, additional loads and durations of the duty cycle may be added to the modified performance test prior to going to a constant current rate. In addition, the note to ITS SR 3.8.4.7 states that the substitution is only allowed as long as the modified performance test completely envelopes the service test. Thus the modified performance test is a more severe test than the performance test, and is acceptable.

Performing the modified performance test at each refueling outage instead of every 60 months allows better trending of the battery capacity with more data points. Over a 20 year battery service life, LaSalle will have 10 trend points if the test is performed every 2 years, instead of 4 trend points if it is performed every 60 months. At the same time, the service test of the battery continues to be verified every 2 years. This has the advantage of having a more accurate identification if the battery is approaching degradation. The additional deep discharges will not significantly affect the batteries. The batteries are designed for 30 deep discharges; the performance of the modified performance test every 24 months only increases the number of deep discharges from 4 to 10. Thus there are 20 deep discharges remaining to support actual challenges to the battery. If these challenges are used, the battery can be replaced at an earlier date. The staff finds the changes to be acceptable.

Decreased Duration of Battery Charger Test (ITS 3.8.4, DOC L.7)

, which relates to the 125 V battery chargers

STS SR 3.8.4.6 states that the battery charger is verified to supply ≥[400] amps at > [250/125] V for ≥ [8] hours. CTS 4.8.2.3.2.c 4 requires, at least once per 18 months, verification that "The battery charger will supply a load equal to the manufacturer's rating for at least 8 hours". ITS SR 3.8.4.6 will require a verification that each required battery charger supplies: a) ≥ 200 amps at ≥ 130 V for ≥ 4 hours for the Division 1 and 2 125 V battery chargers; b) ≥ 50 amps at ≥ 130 V for ≥ 4 hours for the Division 3 125 V battery charger; and c) ≥ 200 amps at ≥ 260 V for ≥ 4 hours for the 250 ~~V battery charger.~~ ITS SR 3.8.4.6 reduces the duration of the test from 8 hours to 4 hours for the 125 V battery chargers.

The 4 hour test is long enough for the battery charger temperature to have stabilized (heat up time usually less than 1 hour) and demonstrate its required capability. The change in the battery charger test duration will not increase the probability of any accident previously evaluated, does not introduce a new mode of plant operation and does not involve physical modification to LaSalle. Therefore, the staff finds this change to be acceptable.

DRAFT SAFETY EVALUATION INPUT

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

BEYOND SCOPE ITEMS

Six Hour Delay to Perform SR (ITS 3.3.3.1, DOC L.2 and ITS 3.4.5 DOC L.3)

For the post-accident monitoring (PAM) instrumentation and the RCS leakage detection system (LDS) instrumentation, a note has been added to the Surveillance Requirements allowing a 6 hour delay from entering into the associated Conditions and Required Actions for a channel that is placed in an inoperable status solely for performance of SRs. For the PAM instrumentation, this is only allowed provided the other channel in the associated function is operable. For LDS instrumentation, the allowance only applies provided the other required LDS instrumentation is operable. The loss of one PAM detection channel or a channel of LDS instrumentation during required testing is acceptable because during these tests another channel is operable to monitor the associated parameter. The short period of time (6 hours) in this condition will have no appreciable impact on risk. Also, upon completion of the Surveillance or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable Condition must be entered and Required Actions taken. The staff finds this change to be acceptable.

Deletion of Modes 3 and 4 in Reactor Protection System (RPS) Electric Power Monitoring Assembly (EPA) Applicability (ITS 3.3.8.2, DOC L.1)

The operability requirements for the RPS EPAs is changed to delete the requirement for them to be operable in Modes 3 and 4. The EPAs provide a regulated power supply for the RPS instrumentation electrical buses. RPS EPAs are provided to isolate the RPS bus from the motor generator set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency conditions. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions and forms an important part of the primary success path of the essential safety circuits. This change is made to establish consistent requirements between RPS instrumentation (LCO 3.3.1.1) and ITS 3.3.8.2 (RPS Electrical Power Monitoring Assemblies). In addition, conforming changes are made to require channel functional testing prior to entry into Mode 2 from Modes 3 or 4.

The only essential equipment required to be operable in Modes 3 and 4 that are powered from RPS buses are the RPS logic and the scram pilot valve solenoids. With the unit in Mode 3 or 4, all control rods are fully inserted and will remain inserted because the Reactor Mode Switch, while in the Shutdown position, enforces a control rod withdrawal block. Thus, it is not necessary for the EPAs to be operable in Modes 3 and 4. However, ITS 3.10.2 (Single Control Rod Withdrawal—Hot Shutdown) and ITS 3.10.3 (Single Control Rod Withdrawal—Cold Shutdown) provide exceptions to the restrictions on control rod withdrawal in Modes 3 and 4. To address these two exceptions, ITS 3.10.2 and ITS 3.10.3 include operability requirements for RPS instrumentation (ITS 3.3.1.1), control rods (ITS 3.9.5), and EPAs (ITS 3.3.8.2). The staff finds this change to be acceptable because the RPS EPAs will be required to be operable when necessary to support RPS operability.

Replace Required Actions to Trip a Recirculation Pump with Actions to Declare the Recirculation Loop Not in Operation (ITS 3.4.1, DOC L.2)

The CTS requirement to trip a recirculation pump within 2 hours when the speed between pumps is mismatched (i.e. flows mismatched) is replaced with (1) a requirement (ITS 3.4.1 ACTION B) to declare the loop with the low flow "not in operation" if the flows remain mismatched after 2 hours, and (2) a caution to operators for cases where flow mismatches are large. While a shutdown of the loop may be preferred under some conditions, declaring a pump not in operation will ensure the proper actions are taken in accordance with the single loop analysis.

In most instances, flow mismatches can be readily alleviated. However, in cases where large flow mismatches occur, low flow, or reverse flow can occur in the jet pumps of the low flow loop, causing jet pump vibration. If zero or reverse flow is detected, the Bases state the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump. Should a LOCA occur with one recirculation loop not in operation, the core flow coast down and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to declare the low flow loop "not in operation." Once the declaration has been made, the appropriate actions for single loop operation must be taken in accordance with ITS 3.4.1 (CTS 3.6.A.1). It is acceptable to establish the single loop analysis requirements of the LCO as they are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values because this satisfies the initial conditions of the accident analysis; therefore, the staff finds this change acceptable.

Changing the Frequency for Monitoring Primary Containment Sump Flow Rate (ITS 3.4.4, DOC L.1)

CTS 4.6.H.2 requires measurements of primary containment sump flow rate to quantify RCS unidentified leakage, total leakage, and unidentified increase leakage to be made at least once per 8 hours, not to exceed 12 hours. The surveillance frequency has been changed to 12 hours in ITS SR 3.4.4.1. This time interval is consistent with the guidance given in Generic Letter (GL) 88-01, Supplement 1, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," which found that, "monitoring reactor coolant system (RCS) leakage every 4 hours creates an unnecessary administrative hardship for plant operators. Thus, RCS leakage measurements should be taken at least once per shift, not to exceed 12 hours." This change allows the 25% extension specified in ITS 3.0.2 to be applied to the current 12 hour surveillance interval. As such, the maximum interval has been extended from 12 hours to 15 hours. The proposed extension to the surveillance interval is acceptable since the probability of a pipe break occurring in the primary containment during the extension period is small and the vast majority of the surveillances are completed with no indication of excessive RCS operational leakage. Furthermore, the leak detection instrumentation will remain available during the extension period such that excessive RCS leakage will continue to be alarmed in the main control room, and a change in sump flow will continue to be indicated on the drywell sump pump flow integrators. The staff finds a 12 hour surveillance interval to be acceptable and consistent with the guidance in GL 88-01, Supplement 1.

More-Restrictive Shutdown Requirements for LPCI Inoperability (ITS 3.5.1, DOC M.1) (to be OPERABLE)

CTS 3.5.A.2 defines the low pressure coolant injection (LPCI) subsystem as being comprised of four LPCI pumps and a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel. ITS 3.5.1 will define two LPCI subsystems, each consisting of two motor driven pumps, piping, and valves capable of transferring water from the suppression pool to the RPV via the "selected" recirculation loop. CTS 3.5.A Action 2.b, which allows the entire LPCI System to be inoperable for 7 days, has been modified to allow only one LPCI subsystem to be inoperable (ITS 3.5.1, Condition B), or one LPCI pump in each LPCI subsystem to be inoperable (ITS 3.5.1 Condition C) for 7 days, or both LPCI subsystems to be inoperable for 72 hours (ITS 3.5.1 Action D). These changes represent additional restrictions on plant operation. The staff finds these changes to be acceptable.

(require)

(each defined in the Bases as)

Change in Number of Automatic Depressurization System Valves (ITS 3.5.1 DOC L.1)

The automatic depressurization system (ADS) is designed to depressurize the reactor to permit the low pressure coolant injection (LPCI) or core spray subsystem to cool the reactor during a small break loss of coolant accident (LOCA) if the high pressure coolant injection (HPCI) system fails or is unable to maintain required water level in the reactor. The Quad Cities ADS system consists of five valves (four relief valves and one safety/relief valve). Qualification of the accumulator for the safety/relief valve to perform the ADS function has not been demonstrated, therefore, the safety/relief valve is not credited in the safety analyses.

Only four ADS valves were assumed operable in the Quad Cities LOCA analyses. One ADS valve of the four valves modeled in the LOCA analyses was assumed to fail for the single failure evaluation resulting in three valve operation credited. The analyses demonstrates that adequate core cooling is provided during small break LOCA and simultaneous battery failure with two of the five ADS valves out-of-service. In order to meet the single failure criteria, the revised TS requires four ADS valves to be operable. It is specified in the revised TS 3.5.1 Bases that the safety/relief valve can not be used to satisfy the ADS valve operability requirements. This ensures that all four relief valves associated with the ADS system will be required to be operable. The analyses in support of the TS change were performed using approved methods, and the licensee has demonstrated that all applicable acceptance criteria continue to be met with the proposed ADS valve operability requirements. Therefore, the staff finds the change to be acceptable.

Change the Acceptance Criteria for Excess Flow Check Valve Tests (ITS 3.6.1.3, DOC L.7)

The requirement in CTS 4.7.D.4 that each excess flow check valve (EFCV) must check flow has been deleted. ITS 3.6.1.3.8 requires, instead, that EFCVs actuate to their isolation position (i.e., closed) on an actual or simulated instrument line break signal. The requirements for the EFCVs are provided in 10 CFR 50 Appendix A, General Design Criteria 55 and 56, and are further detailed in Regulatory Guide (RG) 1.11. These state that there should be a high degree of assurance that the EFCVs will close or be closed if the instrument line outside containment is lost during normal reactor operation, or under accident conditions. The Instrument Line Break Analysis in the Quad Cities UFSAR, Section 15.6.2, assumes both the EFCV and the manual block valve are unavailable, i.e., fail to close; and the accident is terminated by cooling down the plant and closing the manual valve after the plant is shutdown and depressurized. Since the actual leakage is not an assumption of the accident analysis (the leakage is assumed to be the maximum allowed through the broken line), the leakage limit criteria (i.e., check flow) has been deleted. Further, the proposed change ensures that the RG 1.11 criterion ~~that there is~~ , which requires a high assurance that the EFCVs will close will be met. Therefore, the staff finds the change to be acceptable.

Change in Required Spent Fuel Pool Water Level (ITS 3.7.8, DOC M.1)

CTS 3.10.H requires that the spent fuel storage pool water level be maintained at a level of greater than or equal to 33 feet. ~~In the conversion, this technical specification will be renumbered to ITS 3.7.8.~~ The licensee proposes to modify the requirement for the ITS to maintain the spent fuel storage pool water level at greater than or equal to 19 feet over the top of the irradiated fuel. The licensee states that this change results in an increase in the water level by approximately 9 inches. No other changes to the spent fuel storage or pool support systems are proposed. The staff finds that this is a more restrictive requirement for fuel movement and therefore, is an acceptable modification to the technical specifications.

NOTE TO NRC: The deleted sentance is not covered by DOC M.1, but by DOC A.1.

Change in Voltage During Diesel Generator Tests (ITS 3.8.1, DOC M.5)

Currently, the allowable emergency diesel generator (EDG) voltage tolerance in the CTS SRs is 4160 ± 420 volts. The licensee has proposed to change the allowable voltage tolerance to 4160 ± 208 volts. The change will provide more restrictive EDG allowable voltage limits (i.e., from ± 10% to ± 5%) during surveillance testing. The licensee stated that the current voltage tolerance may allow EDG operation at the lower end of the voltage limits, which ~~could~~ may not support operation of ECCS loads within design voltages. Reducing the EDG allowable voltage limits to ± 5% will support operation of all required EDG loads within the design voltage ranges for ECCS loads. The staff concludes that the change is conservative and acceptable.