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ComEd

July 14, 1999

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

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

Subject: Request for License Amendment for Power Uprate Operation

References:
1) Licensing Topical Report, "Generic Guidelines for General Electric BWR Generic Power Uprate," NEDC-31897P-A, Class III, May 1992.

2) Licensing Topical Report, "Generic Evaluations of General Electric BWR Power Uprate," NEDC-31984P, Class III, July 1991, and Supplements 1 and 2.

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposes changes to Facility Operating Licenses NPF-11 and NPF-18 and Appendix A to the Operating Licenses, the Technical Specifications (TS), for LaSalle County Station, Units 1 and 2. The proposed changes will allow both LaSalle County Units to operate at an uprated power level of 3489 MWt. This represents an increase of 5 percent rated core thermal power.

This proposed change follows the generic guidelines for uprating the power of Boiling Water Reactors (BWRs) described in References 1 and 2. Attachment A contains a detailed description of the specific proposed changes necessary for operation at uprated conditions and the technical bases for these changes. Attachment B provides the proposed markup to the Units' TS. Attachment C provides information supporting a finding of no significant hazards consideration in accordance with 10 CFR 50.92(c). Attachment D provides information supporting an Environmental Assessment. Attachment E contains the detailed plant-specific safety analysis required by the generic guidelines. This enclosure contains proprietary information and we request that it be withheld from public disclosure in accordance with 10 CFR 2.770 (a) (4). The affidavit supporting the request for withholding Attachment E from public disclosure, as required by 10 CFR 2.790 (b)(1), is provided in Attachment F.

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change: NRC PDR

The Plant Operations Review Committee and the Nuclear Safety Review Board, in accordance with the Quality Assurance Topical Report, have reviewed the proposed changes.

ComEd is notifying the State of Illinois of this license amendment request by transmitting a copy of this letter and its attachments to the designated State Official.

Any plant modification that requires the plant to be shutdown to conduct the work necessary to support the implementation of these proposed changes will be made during the next Unit 1 and Unit 2 refueling outages, scheduled for October 1999 and October 2000, respectively. Other modifications will be implemented prior to operating at uprated conditions. ComEd plans to fully implement the uprated power conditions for Unit 1 by May of 2000, and will operate Unit 2 at limited uprated conditions supported by the unmodified hardware configuration. ComEd will fully implement uprated conditions on Unit 2 following the startup from the October 2000 refueling outage. Therefore, ComEd requests that if found acceptable, this proposed change be approved by May 1, 2000.

Should you have any questions related to this request, please contact Mr. Robert R. Brady, Jr. at (630) 663-7205.

Respectfully,

R. M. Krich
Vice President - Regulatory Services

Attachments

Affidavit
Attachment A: Description and Summary Safety Analysis for Proposed Changes
Attachment B: Marked-Up TS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: GE Report NEDC-32701P, "Power Uprate Safety Analysis Report for LaSalle County Station Units 1 and 2," July 1999 (proprietary)
Attachment F: GE Affidavit for NEDC-32701P

cc: Regional Administrator -- NRC Region III
NRC Senior Resident Inspector -- LaSalle County Station
Office of Nuclear Facility Safety -- IDNS

STATE OF ILLINOIS)

COUNTY OF DUPAGE)

IN THE MATTER OF:)

COMMONWEALTH EDISON (COMED) COMPANY) Docket Numbers

LASALLE COUNTY STATION - UNITS 1 and 2) 50-373 and 50-374

SUBJECT: Request for License Amendment for Power Uprate Operation

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

RMKrich

R. M. Krich
Vice President – Regulatory Services

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 14th day of

July, 1999

Linda S. Sanders
Notary Public

OFFICIAL SEAL
LINDA S. SANDERS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 12-29-2002

DESCRIPTION AND SUMMARY SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

The requested amendment consists of a number of proposed changes that will permit uprated power operation of the LaSalle County Station (LCS), Units 1 and 2. During the course of the power uprate review, a non-conservative value was noted in Technical Specifications (TS) Section 3.6.1.6 "Drywell and Suppression Chamber Internal Pressure." The upper limit for drywell and suppression chamber internal pressure will be changed to reflect the input assumptions of the accident analysis.

LCS is a dual-unit General Electric (GE) Boiling Water Reactor (BWR)/5 with a Mark II containment. The units at LCS, like most BWR plants, have equipment and system capability to accommodate steam flow rates at least five percent above the original rating. Because of the significant economic advantages of operating at a higher power level, ComEd is proposing permanent changes to the operating license for each LCS unit that will enable the units to be operated at power levels up to 105% of current rated thermal power of 3323 MWt. This corresponds to a thermal power level of 3489 MWt.

The analyses and evaluations supporting the proposed changes directly related to power uprate were completed using the guidelines in GE Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (Reference I.1). Certain issues are evaluated generically and have been submitted to the NRC in GE Topical Report NEDC-31984P, "Generic Evaluation of General Electric Boiling Water Reactor Power Uprate" (Reference I.2). Both of these topical reports have been approved by the NRC, by letters dated September 30, 1991 and July 31, 1992, and form the basis for the LCS thermal power uprate evaluation. A LCS plant specific safety analysis, NEDC-32701P, "Power Uprate Safety Analysis Report for LaSalle County Station Units 1 and 2," (Reference I.3) has been performed by GE. This report is provided as Attachment E.

The method for achieving higher reactor power is to increase core thermal power with a more uniform and flattened power distribution to create an increase in steam flow. A corresponding increase in feedwater flow will be required. In addition, the power/flow map will be extended to the Maximum Extended Load Line Limit (MELLL) domain to increase operational flexibility and minimize need for frequent reactor control rod pattern adjustments. The maximum allowable core flow rate does not change as a result of power uprate. Uprated operation will not involve increasing reactor pressure vessel (RPV) dome pressure because the LCS units have sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine. However, to maintain the GE standard turbine flow margin of three (3)

%, modifications will be made to the high pressure turbines. Attachment E highlights other minor modifications to balance of plant components necessary to maintain adequate performance margins.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

The current Operating License and TS limit operation at a core thermal power of 3323 MWt.

TS Section 3.6.1.6, currently specify an upper drywell internal positive pressure of 2.0 psig.

C. BASES FOR THE CURRENT REQUIREMENTS

The current Operating License and the affected TS were based on a rated core thermal power of 3323 MWt. The supporting accident and transient analysis justifying operation were based on this rated power with appropriate margins added in accordance with regulatory guidance.

The TS Section 3.6.1.6 requirement regarding the drywell internal positive pressure of 2.0 psig is based on limiting the accident pressure to 39.6 psig, which is less than the design pressure of the drywell, to be consistent with the accident analysis.

D. NEED FOR REVISION OF THE REQUIREMENT

The proposed change would allow an increase in licensed core thermal power from 3323 MWt to 3489 MWt and allow the flexibility to increase the potential electrical output of the LCS, Units 1 and 2. The drywell and suppression chamber pressure TS limit was determined to be a non-conservative TS limitation, and was entered into the site's corrective action program. The corrective actions have placed administrative controls to limit drywell and suppression chamber pressure. In accordance with the guidance of Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," this submittal provides timely resolution of the issue.

E. DESCRIPTION OF THE PROPOSED CHANGES

Unless otherwise stated, the affected TS sections are the same for Unit 1 and Unit 2.

1. Rated thermal power is increased from 3323 MWt to 3489 MWt. This change will be reflected on page 3, Condition C.1 of the Unit 1 and Unit 2 Operating License Nos. NPF-11 and NPF-18, respectively. In addition, Section 1.1, "(Definitions" of the Unit 1 and Unit 2 TS, Subsection 1.35, "Rated Thermal Power," will be revised to reflect the change in rated thermal power.

2. The trip setpoint and allowable value for the Average Power Range Monitor (APRM) Flow Biased Simulated Thermal Power – High Scram for two loop operation contained in TS Section 2.2, "Limiting Safety System Settings – Reactor Protection System Instrumentation Setpoints," will be revised. These values appear in TS Table 2.2.1-1, Function 2.b.(1)(a). The trip setpoint will be revised from "0.58W + 59%," to "0.62W + 63.7%," and the allowable value will be changed from "0.58W + 62%," to "0.62W + 69.3%."

3. The trip setpoints and allowable values for the APRM Flow Biased Simulated Thermal Power – High Scram and the High Flow Clamped for single loop operation contained in TS Section 2.2, "Limiting Safety System Settings – Reactor Protection System Instrumentation Setpoints," will be revised. These values appear in TS Table 2.2.1-1, Function 2.b.(2)(a) and 2.b.(2)(b). For the APRM Flow Biased Simulated Thermal Power – High Scram the trip setpoint will be revised from "0.58W + 54.3%," to "0.55W + 51.5%," and the allowable value will be changed from "0.58W + 57.3%," to "0.55W + 56.8." For the High Flow Clamped, the trip setpoint will be revised from 113.5%" to "108.1%" and the allowable value will be changed from "115.5%" to "112.3%."

4. The trip setpoint and allowable value for the Automatic Initiation – Primary Containment Isolation – Main Steam Line - Flow – High contained in TS Section 3.3.2, "Isolation Actuation Instrumentation. Main Steam Line - Flow – High setpoint and allowable values appear in TS Table 3.3.2-2, Function A.1.c.(3). The trip setpoint will be changed from 111 psid to 125 psid and the allowable value will be changed from 116 psid to 128 psid.

5. The trip setpoint and allowable value for the APRM Flow Biased Simulated Thermal Power – Upscale Control Rod Withdrawal Block for two loop operation contained in TS Section 3.3.6, "Control Rod Withdrawal Block Instrumentation," will be revised. These values appear in TS Table 3.3.6-2, Function 2.a.(1). The trip setpoint will be revised from "0.58W + 47%," to "0.62W + 52.3%," and the allowable value will be changed from "0.58W + 50%," to "0.62W + 57.9%".

6. The trip setpoint and allowable value for the APRM Flow Biased Simulated Thermal Power – Upscale Control Rod Withdrawal Block for single loop operation contained in TS Section 3.3.6,"Control Rod Withdrawal Block Instrumentation," will be revised. These values appear in TS Table 3.3.6-2, Function 2.a.(2). The trip setpoint will be revised from "0.58W + 42.3%," to "0.55W + 40.0%," and the allowable value will be changed from "0.58W + 45.3%," to "0.55W + 45.4%."

7. The upper limit of Drywell and Suppression Chamber internal pressure contained in TS Section, 3.6.1.6, "Drywell and Suppression Chamber Internal Pressure," will be changed from +2.0 psig to +0.75 psig.

8. The peak post accident containment pressure (Pa) will be changed from 39.6 psig to 39.9 psig in the Primary Containment Leakage Rate Testing Program in TS Section 6.2.F.7.

Bases Changes

9. The peak post accident containment pressure (Pa) is changed from 39.6 psig to 39.9 psig. This value appears in the Bases Section B 3/4.6.1.6. In addition, the reference to 2.0 psig is changed to 0.75 psig to be consistent with Change #7.

10. The bases for TS Section 3.6.2, "Depressurization Systems," will be revised. The 200°F bulk suppression pool limitation to prevent condensation oscillation and other condensation-related loads has been removed. The bases have been revised to reflect the new limitation of 208°F bulk suppression pool temperature.

11. In the bases for TS Section 3.4.6, "Pressure/Temperature Limits," TS Table B3/4.4.6-1 will be revised as follows for Unit 1 and Unit 2 to reflect the results of the fracture toughness review conducted under power uprate conditions.

    The specified Reference Temperatures and Upper Shelve Energy values for Unit 1 Beltline Plate Heats C5978-2 and C6345-2 and Beltline Weld Heat 1P3571/3958 will be changed to reflect values contained in Table 3-1 of Attachment E. In addition, the * and ** footnotes will be removed since they are no longer applicable.

    The specified Reference Temperatures and Upper Shelve Energy values for Unit 2 Beltline Plate C9404-2 and C9425-1, and Beltline Weld Heat 3P4966/1214 will be changed to reflect values contained in Table 3-2 of Attachment E. In addition, the ** footnote will be removed since it is no longer applicable.

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

Change # 1 Change in rated thermal power from 3323 MWt to 3489 MWt

The detailed safety analysis for the proposed changes is contained in Reference I.3 and is provided as Attachment E. The analyses demonstrate that LCS, Unit 1 and Unit 2 can operate safely with the proposed five percent increase in maximum core thermal power, with a corresponding 5.9 percent increase in steam flow from the RPV, and the required increases of the flow, temperature, and pressure in the supporting systems and components. Increasing the licensed maximum thermal power level of LCS, Unit 1 and Unit 2 to 3489 MWt can be accomplished safely. The following summarizes the information provided in Attachment E providing the necessary basis for this conclusion.

LCS, Unit 1 and Unit 2 are currently licensed for a 100% power level of 3323 MWt. The original safety analysis basis assumed that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level. The uprate power level included in this analysis is a five-percent thermal power uprate (i.e., 3489 MWt). The power uprate safety analyses are based on a power level of at least 1.02 times the uprated power level, except that some analyses are performed at 100% uprated power, because the two percent power factor specified in Regulatory Guide 1.49, "Power Levels of Water-Cooled Nuclear Power Plants," dated December 1973, is already accounted for in the analysis methods.

The analyses presented in Attachment E assure that the margins that are dependent upon power levels are maintained by meeting the appropriate regulatory criteria. NRC-accepted computer codes and calculation techniques were used to perform the calculations that demonstrate meeting the stipulated criteria. Similarly, margin specified by application of the American Society of Mechanical Engineers (ASME) design rules are maintained, as are other margins associated with criteria used to judge the acceptability of the plant. Environmental margins are maintained by retaining the present limits for releases such as ultimate heat sink maximum temperature or plant vent radiological limits, as a consequence of power uprate.

Effects on Plant Systems
Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions. Modifications to balance of plant components necessary to support power uprate, and maintain performance margins, are identified in Attachment E. The review has concluded that operation at power uprate conditions will not affect the reliability of plant equipment, and that current TS surveillance requirements ensure adequate monitoring of system operability.

Fuel Design Considerations
No change is required in the basic fuel design to achieve the uprated power levels or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for power uprate. The established fuel operating limits will still be met at the uprated power level. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC or otherwise approved in the TS. No new fuel design is required for power uprate. Specific core performance is evaluated for each fuel reload. This evaluation will be conducted using approved methodology for the fuel designs implemented at LCS, Units 1 and 2.

Capability of Water Makeup Sources
The BWR design concept includes a variety of ways to pump water into the RPV to deal with all types of events. The BWR has both safety-related and non-safety-related cooling water sources. Many of these diverse water supplies are redundant in equipment and also redundant in systems. Power uprate does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC-approved methods were

used for analyzing the performance of the Emergency Core Cooling Systems (ECCS) during loss-of-coolant-accidents (LOCAs). Power uprate results in a small (five percent) increase in decay heat and, thus, the core cooling time to reach cold shutdown increases slightly. This is not a safety concern, and the existing cooling capacity can bring the plant to cold shutdown within an acceptable time span.

Design Basis Accidents
A review of Design Basis Accidents (DBAs) was conducted. DBAs are very low probability hypothetical events whose characteristics and consequences are used in the design of the plant to demonstrate that the plant can mitigate consequences of DBAs to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, a feedwater lines, and even the most limiting small lines. This break range boun the full spectrum of large and small, high and low energy line breaks and the success of the plant systems in dealing with them while accommodating a single active equipment failure in addition to the postulated LOCA. The following elements are validated by the review of the DBAs.

- Challenges to Fuel or ECCS Performance Analyses. The current licensing basis is provided in Updated Final Safety Analysis Report (UFSAR) Section 6.3, "Emergency Core Cooling Systems," and shows compliance with the rules and criteria of 10 CFR 50.46, "Acceptance Criteria For Emergency Core Cooling Systems For Light Water Reactors," and Appendix K, "ECCS Evaluation Models," by ensuring fuel Peak Cladding Temperature (PCT) limitations are met.
- Challenges to the Containment. The current licensing basis is provided in UFSAR Section 6.2,"Containment Systems." This demonstrates that under DBA conditions, the integrity of the containment can be maintained.
- DBA Radiological Consequences. The current licensing basis is provided in UFSAR Section 15.6, "Decrease in Reactor Coolant Inventory." The calculated consequences are compared to the criteria of 10 CFR 100, "Reactor Site Criteria."

The calculated PCT for power uprated conditions is 1296°F for the GE9 (i.e., GE8x8NB) fuel type as provided in Table 4-2 in Attachment E. This PCT includes the effects of the Maximum Extended Load Line Limitation Analysis (MELLLA) and Increased Core Flow (ICF). LCS Unit 2 currently employs a mixed core containing both GE and Siemens Power Corporation (SPC) fuel. The ECCS-LOCA analysis for the SPC fuel was conducted at a power level that bounds the uprated conditions and yielded a resultant PCT of 1807°F. This is documented in Reference I.4. The licensing safety margin is not affected by power uprate. The increased PCT consequences for power uprate are insignificant compared to the large amount by which the results are below the regulatory criteria of less than 2200°F, and all other 10CFR50.46 acceptance criteria continue to be met. Therefore, the ECCS safety margin is not affected by power uprate.

To confirm the continued acceptability of the containment structure, short-term and long-term containment analysis were conducted. The short-term analysis was directed primarily at determining the containment pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis was directed primarily at the suppression pool temperature response, considering the increased decay heat addition to the suppression pool.

The short-term containment analysis concluded that the peak containment drywell pressure would be 39.9 psig, and is below the design value of 45 psig. The long-term containment analysis concluded that in the event of a LOCA the calculated peak bulk suppression pool temperature would be 193°F. This is less than the design temperature of the suppression pool of 275°F, and the criteria used to ensure adequate Net Positive Suction Head (NPSH) to the ECCS pumps which is 212°F. Therefore, power uprate does not challenge the structural integrity of the containment structure and ECCS NPSH is assured.

The radiological consequences due to the LOCA were calculated and were found to be below the applicable regulatory limits. The results are presented in Table 9-3 of Attachment E and are summarized in Table A1.

The LOCA radiological consequences have increased due to power uprate; however, they are still below established regulatory limits.

Radiological evaluations for other non-LOCA DBAs were also performed. Where appropriate, the doses were increased to account for the effects of power uprate. These changes are outlined in Section 9.2 of Attachment E and they demonstrate that the applicable regulatory limits are met for LCS, Units 1 and 2.

<u>Non-DBA Radiological Doses</u>

All of the other radiological releases discussed in UFSAR are either unchanged because they are not power-dependent, or increase approximately in linear proportion to the amount of the uprate. The dose consequences for all of the non-LOCA radiological release accident events are bounded by the "LOCA Radiological Consequences" discussed above and were shown to meet the current dose acceptance criteria. These events are discussed in Section 9.2 of Attachment E.

Table A1
LOCA Radiological Consequences

Location	Pre-Uprate Dose (rem)	Power Uprate Dose (rem)	Limit (rem)
Exclusion Area Boundary			
Whole Body Dose (total)	3.6[a]	3.6	≤ 25
Thyroid Dose	32.3[a]	32.3	≤ 300
Low Population Zone			
Whole Body Dose (total)	0.6[a]	0.6	≤ 25
Thyroid Dose	12.8[a]	12.8	≤ 300
Control Room			
Whole Body Dose (total)	1.4	1.6	≤ 5
Thyroid Dose	19,9[a]	19.9	≤ 30
Beta Dose	1.7[a]	1.7	≤ 30
Auxiliary Electric Equipment Room			
Whole Body Dose (total)	1.4	1.6	≤ 5
Thyroid Dose	28.9[a]	28.9	≤ 30
Beta Dose	2.0[a]	2.0	≤ 30

(a) Prior approved TS amendment (Amendment 125 to NPF-11 and Amendment 110 to NPF-18) incorporated new source terms at uprated conditions

Transient Analyses

The effects of plant transients were evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR) for a representative core. The power uprate transient analyses were performed using the approved methodology specified in the LCS TS. The limiting transient events are slightly more severe when initiated from the uprated conditions. The power uprate transient analyses results show a slightly more limiting event initial Critical Power Ratio (CPR) (i.e., ≤0.02) than when initiated from the current rated thermal power; however for the most limiting transient the evaluation of the representative core showed the Operating Limit Minimum Critical Power Ratio (OLMCPR) to be acceptable. Table 9-2 of the Attachment E provides the evaluation results for the transient analyses. The margin of safety established by the SLMCPR is not affected by the 105 % power uprate to 3489 MWt. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish the cycle specific Minimum Critical Power Ratio (MCPR) safety limit and fuel operating limits.

Environmental Qualification

Safety related electrical equipment within the scope of 10CFR50.49,"Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," was evaluated to assure qualification for the normal and accident conditions associated with proposed power uprate conditions. Environmental qualification (EQ) for safety-related electrical equipment located inside the primary containment includes the environments expected to exist during normal plant operation and is also based on main steam line break and/or LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences. The current accident and normal plant conditions for temperature, pressure, and humidity are nearly unchanged for the uprated power conditions. The current design basis radiation levels under normal plant conditions are unchanged at uprated conditons, and accident conditions are conservatively evaluated to increase by 16 %. Accident temperature, pressure, and humidity environments used for qualification of equipment outside primary containment result from a main steam line break in the pipe tunnel, or other high energy line breaks, whichever is limiting for each plant area. The accident temperature, pressure, and humidity conditions resulting from a LOCA do not change with power level, but some of the high energy line break environmental profiles do increase by a small amount. The normal temperature, pressure, and humidity conditions do not change as a result of power uprate. The current radiation levels under normal plant conditions are unchanged, and accident conditions outside primary containment are conservatively evaluated to increase by 16% for all zones except zones in the reactor building (H4A through H4H), excluding the ECCS equipment cubicles and High Energy Line Break (HELB) local areas and the two floors of the reactor building. For zones H4A through H4H, the accident conditions are conservatively evaluated to increase by 10%. These increases do not impact the bounding environmental conditions currently in the UFSAR. Therefore equipment qualification remains bounded by the current basis.

Instrumentation

The - instrumentation and control signal ranges and analytical limits for setpoints were evaluated to establish the effects of changes in various process parameters. As required, analyses were performed to determine the need for setpoint changes for various functions. The revised setpoints have been established using ComEd setpoint methodology. Each setpoint was selected with sufficient difference between the system setting and the actual value in the safety analysis (i.e., analytical limit) to preclude inadvertent initiation of the protective action while assuring adequate allowances for instrument accuracy, calibration, drift, and applicable design basis events relative to the analytical limit.

Change #2 and Change #5 - APRM Flow Biased Simulated Thermal Power – High Scram and APRM Flow Biased Simulated Thermal Power – Upscale Control Rod Withdrawal Block for two loop operation

The changes to these parameters are made in response to operation at uprated conditions. The justification for operation at uprated conditions is provided in the safety analysis for Change #1.

For two recirculation loop operation (i.e., TLO), the analytical limits for the APRM Flow Biased Simulated Thermal Power (STP) Scram and Rod Withdrawal Block equations are changed to reflect the revised load line due to MELLL and then reduced by the ratio of 100/105 for power uprate.

The Rod Block Monitor (RBM) instrumentation setpoints in terms of percent power and flow are unaffected by MELLL and power uprate conditions as these are determined in the cycle-specific Rod Withdrawal Error (RWE) transient analysis.

Change #3 and Change #6 - APRM Flow Biased Simulated Thermal Power – High Scram and APRM Flow Biased Simulated Thermal Power – Upscale Control Rod Withdrawal Block for single loop operation

The changes to these parameters are made in response to operation at uprated conditions. The justification for operation at uprated conditions is provided in the safety analysis for Change #1.

For single recirculation loop operation (i.e., SLO), the APRM Flow Biased Scram and Rod Withdrawal Block equations are not affected by MELLL, but are reduced by the ratio of 100/105 for power uprate. The analytical limit for the High Flow Clamped setpoint is conservatively reduced by 100/105 for power uprate.

The RBM instrumentation setpoints in terms of percent power and flow are unaffected by MELLL and power uprate conditions as these are determined in the cycle-specific Rod Withdrawal Error (RWE) transient analysis.

Change #4 - Automatic Initiation – Primary Containment Isolation – Main Steam Line - Flow – High

The change to this parameter is made in response to operation at uprated conditions. The justification for operation at uprated conditions is provided in the safety analysis for Change #1.

The analytical limit for uprate remains at 140% of the rated steam flow. The instrumentation will be recalibrated for the higher steam flow condition, and thus, the TS Allowable Value and nominal Trip Setpoint differential pressure units, in psid, are changed accordingly. These changes ensure that sufficient difference to the trip setpoint exists to allow for normal testing of the Main Steam Isolation Valves (MSIVs) and turbine stop valves.

Change #7 and Change#9 – Change to the upper limit of Drywell and Suppression Chamber internal pressure

The current short-term containment pressure response analysis presented in Section 6.2 of the LCS UFSAR assumed a drywell pressure, as an initial condition, of 0.75 psig. The resulting peak pressure was 39.6 psig. The basis provided the corresponding TS limit states that maintaining internal drywell pressure less than 2.0 psig would ensure that containment peak pressure would remain at the assumed value of 39.6 psig.

A review of plant operational practices found that action levels are taken when containment pressure reaches 0.3 psig, and that units have not operated at a pressure greater than 0.75 psig in the containment.

The analysis conducted for power uprate was also done assuming an initial pressure of 0.75 for determining containment peak pressure, which is 39.9 psig assuming uprated conditions.

Therefore, this proposed change is to make the TS upper containment pressure limitation consistent with the assumptions used in the analysis. In addition the bases for this TS requirement will be revised to reflect the assumptions of the analysis, and to reflect the change to calculated peak pressure under power uprate conditions.

Change #8 – Change to containment peak pressure used in the Appendix J Leak Rate Test Program

This value has changed as a result of the short-term containment response analysis conducted at power uprated conditions. The safety analysis for power uprate is provided in Attachment E and is summarized in the discussion for Change #1.

Change # 10 – Change to Bulk Suppression Pool temperature limitation

The suppression pool temperature is currently limited to a 200°F bulk pool temperature. This limit was meant to ensure that the minimum local subcooling at the quencher location would remain within the range of the available test data used to demonstrate stable condensation during RPV blowdown conditions. After NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," was issued, GE compiled additional quencher test data, which was used to confirm that with quenchers, stable condensation is ensured even with local pool temperatures approaching saturation conditions. GE documented these findings and presented them to the NRC for review and approval. The findings were approved by the NRC by letter dated August 29, 1994, and provide justification for elimination of the local pool temperature limit for plants with T-quenchers.

However, in the August 29, 1994 NRC Safety Evaluation, an additional concern was raised regarding the potential transfer of non-condensed Safety-Relief Valve (SRV) steam to the ECCS suction strainer, if local saturated conditions existed at the quencher and the ECCS suction strainer is at a higher elevation than the SRV quencher. The NRC stated that the local pool temperature limits could be eliminated if the plant's ECCS suction is located below the elevation of the quencher elevation. The ECCS suction line elevations for LaSalle are above the elevation of the T-quenchers. However research results show that long plumes occurred at subcooling levels less than 9°F. The LaSalle T-quenchers are at a submergence of 24 feet and provide approximately 20°F subcooling with a bulk suppression pool temperature of 208°F and the wetwell at atmospheric pressure. This provides sufficient margin to ensure that exiting steam is condensed before posing a steam ingestion potential to the ECCS suction.

Therefore, the new limitation placed on the suppression pool is 208°F bulk pool temperature.

Change #11 - Changes to the Fracture Toughness Values in Tables B 3/4.4.6-1 "Reactor Vessel Toughness"

The fracture toughness values are updated, as a result of the higher uprated neutron fluence values used in the uprate analysis.

G. IMPACT ON PREVIOUS SUBMITTALS

All submittals currently under review by the NRC were evaluated to determine the impact of this submittal. No submittals currently under review are impacted by the information presented in the license amendment request.

H. SCHEDULE REQUIREMENTS

ComEd plans to fully implement the uprated power conditions for Unit 1 by May of 2000, and will operate Unit 2 at limited uprated conditions supported by the unmodified hardware configuration. ComEd will fully implement uprated conditions on Unit 2 following the startup from the October 2000 refueling outage. Therefore, ComEd requests that if found acceptable this proposed change be approved by May 1, 2000. Any plant modification that requires the plant to be shutdown to conduct the work necessary to support this amendment will be made during the next refueling outages for Unit 1 and Unit 2 scheduled for October 1999 and October 2000, respectively. Other modifications will be implemented prior to operating at uprated conditions.

I. REFERENCES

1. Licensing Topical Report, "Generic Guidelines for General Electric BWR Generic Power Uprate," NEDC-31897P-A, Class III, May 1992.

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