UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 31, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS SECTION 3.3.6.1, "PRIMARY CONTAINMENT AND DRYWELL ISOLATION INSTRUMENTATION," TO ELIMINATE REQUIREMENTS FOR MAIN STEAMLINE ISOLATION ON HIGH TURBINE BUILDING TEMPERATURE (TAC NO. ME1499)

Dear Mr. Pardee:

By letter to the Nuclear Regulatory Commission (NRC) dated June 15, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091660580), Exelon Generation Company (Exelon) LLC, submitted a request to eliminate the requirement for main steamline isolations on high turbine building temperatures from technical specification (TS) 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," for the Clinton Power Station, Unit No. 1.

By letter dated June 20, 2009 (ADAMS Accession No. ML091730148), Exelon submitted a request for processing the above amendment on an emergency basis. The NRC staff reviewed this request to process the license amendment request on an emergency basis, and identified multiple technical and regulatory issues that have precluded a prompt determination that the proposed changes are acceptable to make a reasonable assurance safety determination on an emergency basis.

The NRC staff is reviewing your submittal and the supplement, dated June 23, 2009 (ADAMS Accession No. ML091740100) , and has determined that additional information is required to complete the review. The NRC staff also expects that the DRAFT information submitted by letter dated June 24, 2009, would be placed on the docket under oath or affirmation as well. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on Tuesday, July 21, 2009, it was agreed that you would provide a response by August 21, 2009.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-3154.

Sincerely,

Marshall David for

Stephen P. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

CLINTON POWER STATION (CPS), UNIT NO. 1

DOCKET NO. 50-461

In reviewing the Exelon Generation Company s (Exelon s) submittal dated June 15, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091660580), related to your amendment request to eliminate the requirement for main steam-line isolations on high turbine building temperatures from technical specification (TS) 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," for the CPS Unit No. 1 (Clinton), the Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

AADB Question 1

The proposed change removes automatic isolation of a steamline leak in the turbine building and a main steam isolation valve (MSIV) isolation. It incorporates manual operator actions to mitigate the consequences of a main steamline break in the turbine building. Per the updated final safety analysis report (UFSAR) Section 15.6.4.2.1, "A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the four main steam lines." In addition, Branch Technical Position ASB 3-1, "Protection Against Piping Failures in Fluid Systems Outside Containment," appears to be part of the CPS design basis. ASB 3-1 pertains to moderate and high energy fluid system piping and states that the effects of each postulated piping failure should be shown to result in acceptable offsite and onsite consequences. UFSAR Section 10.3.3 states: "The break of a main steamline or any branch line will not result in radiation exposures in excess of the limits of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67 to persons located offsite because of the safety features designed into the system." Therefore, an analysis of the spectrum of breaks appears to be part of the CPS design basis.[1]

CPS has stated in telephone conference calls that smaller line breaks would be mitigated by operator actions. In addition, in a response to the NRC's request for additional information (RAI) dated, June 23, 2009, CPS stated, "Operator actions would be taken in accordance with emergency operating procedures to isolate primary and secondary pathways not needed for Emergency Operating Procedure (EOP) actions." These actions would be taken based upon alarms for the Turbine Building ambient temperature and a "Stack" normal range monitor.

The June 23, 2009, correspondence also states, "Consequently, the delay and monitoring designed into the VT system and Stack provides appropriate control of any releases such that an analysis of the entire spectrum of steamline breaks below the bounding break is not warranted."

The NRC staff is concerned that CPS's argument that the "VT system and Stack provide appropriate controls" relies on what appears to be non-safety related equipment (i.e. turbine building[2], "Stack" radiation monitors[3], etc). The crediting of the operator action to mitigate the

1 Small and large breaks are considered for the Main Steamline Break Accident as noted in the description for Event 43, UFSAR page 15A-54.

2 Per UFSAR Chapter 10.1, "The majority of the steam and power conversion system is located in the turbine building which is a non-seismic, non-safety-related building."

consequences of the spectrum of steamline breaks appears to conflict with the current licensing bases that states that CPS conforms to Regulatory Position 5.1.2 of Regulatory Guide 1.183 (see the CPS submittal for the alternative source term and UFSAR Section 1.8). Regulatory Position 5.1.2 states:

> "Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, and are powered by emergency power sources, and are either automatically actuated, or in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed."

These guidelines do not appear to have been addressed; rather CPS states that no changes to the existing radiological analyses are required by the proposed change.

Please explain how all accident mitigation features, credited in the argument that the current licensing bases remains bounding (i.e. turbine building, "Stack" radiation monitors, turbine building ventilation, etc.), meet Regulatory Position 5.1.2 or explain why they do not need to meet the Regulatory Position.

AADB Question 2

In the CPS RAI response entitled "Supplemental Information Related to Design Basis Control Rod Drop (CRDA)," dated June 24, 2009, CPS stated:

> "At Clinton Power Station (CPS), the condenser activity release pathway is the limiting accident for the CRDA analysis for dose. The condenser activity release credits a pathway of release through the steam tunnel piping. Following a CRDA, radioisotopes postulated to be released will be transported through the main steamline (MSL) directly to the main steam condenser. From there, activity that was accumulated in the condenser is assumed to leak into the Turbine Building at a rate of 1% per day, then to the environment through the HVAC Stack (Stack). The CPS Alternative Source Term submittal assumed that all activity was transported to the condenser. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," gives guidance on considering more mechanistic approaches for the transport and outlines that forced flow paths should be assumed for the flow rate associated with the most limiting case. CPS recognizes that part of the basis for not including a more mechanistic approach to the transport is that substantial steam leaks from the MSLs do not exist as an initial condition for the accident."

The proposed changes in the letter dated June 15, 2009, state that:

> "No credit is taken in the transient or accident analysis for the automatic isolation of the MSIVs by these Turbine Building area temperature switches. Thus, the Turbine Building temperature switches are not assumed to function to mitigate any accident described in Chapters 6 or 15 of the CPS USAR."

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3 Per the draft June 24, 2009, letter, entitled, "Supplemental Information Related to Amendment to Technical Specification 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation" (TAC No. ME1499)," the Stack process radiation monitors are Category 3 monitors.

However, the purpose of the limiting condition for operation (LCO) for primary containment isolation instrumentation is to satisfy 10 CFR Section 50.36, criterion 2 and 3, which establish, in part, an operating restriction that is an initial condition of a design-basis accident (DBA) or a structure which is part of the primary success path and which functions or actuates to mitigate a DBA.

The NRC staff is concerned that with the proposed change that leakage from the steamline is not modeled for the control rod drop accident, nor is it controlled as required by 10 CFR 50.36 with a TS.

According to the June 24, 2009, RAI response, the largest mass flow rate from a MSL leak that would not result in a Group 1 isolation is approximately 2350 gallons per minute (gpm). Currently, credit is assumed for deposition of radioactivity in the steamline and the condenser and holdup within the condenser. The proposed change will allow leaks beyond the current licensing basis of zero gpm, assumed for the control rod drop accident (CRDA). The radioactivity from these leaks should not be reduced by the holdup and deposition currently credited in the current CRDA design-basis, since it bypasses the condenser and potentially portions of the steamline. Since the deposition and holdup credit is large, the increased leakage pathway may yield substantially greater amounts of radioactivity released during a CRDA than in the current CRDA analysis.

It is not clear to the NRC staff that the current licensing limits for the exclusion area boundary, low population zone and control room will continue to be met since 1) the new pathway leakage can potentially be large, and 2) holdup and deposition would potentially not be creditable for this leakage.

a) Please provide a detailed assessment or a calculation of the consequences of a CRDA with the proposed change. In addition to the dose consequences of leakage from the condenser, please include the impact of leakage from the steamline. The assessment should provide the maximum allowable design leakage from the steam piping, the total mass released before termination of the CRDA, and any reductions in radioactivity credited. Please include a justification for each assumption, methodology and the inputs used. Also, please provide a list of any changes from your current design bases CRDA.

b) Per 10 CFR 50.36, Criterion 2 and 3, please include an LCO for the initial leakage from the steamline or provide a justification why an LCO is not needed.

AADB Question 3

UFSAR Section 5.2.5.2.2, "Containment/Drywell Airborne Radioactivity Monitoring," states: "The radioactivity monitors for detecting RCPB [reactor coolant pressure boundary] leakage are subject to substantial limitations of their usefulness as described below. The particulate and iodine monitors are not effective due to the significant amount of plateout. The noble gas monitor is used to alarm for large leaks and pipe breaks. The reliability, sensitivity and response times of radiation monitors to detect 1 gpm in one hour of RCPB leakage will depend on many complex factors. The major limiting factors are discussed below."

CPS has stated in telephone conference calls that line breaks could be mitigated by operator actions. In addition, in a response to a RAI dated June 23, 2009, CPS stated, "Operator actions would be taken in accordance with emergency operating procedures to isolate primary and secondary pathways not needed for Emergency Operating Procedure (EOP) actions." These actions would be taken based upon alarms for the Turbine Building ambient temperature and a "Stack" normal range monitor.

The NRC staff would like to assess the operator actions credited to mitigate the consequences of a steamline break in the turbine building and evaluate continued compliance with 10 CFR 50.67 and conformance to Regulatory Guide 1.183. Like the containment/drywell airborne monitors described in UFSAR Section 5.2.5.2.2, the "Stack" normal range monitors may be "subject to substantial limitations regarding their usefulness" to detect leakage. Please provide an assessment of how the "Stack" normal range monitors are credited. In the assessment, please address the applicable issues identified in UFSAR Section 5.2.5.2.2 and their impact on identifying a leak. In the assessment, provide calculations that show that the monitor can function to mitigate the consequences of all applicable DBAs (i.e. Control Rod Drop Accident, Main Steamline Break Accident, etc) or justify why an assessment is not needed. If an assessment is provided, please provide enough information so that the NRC staff can independently verify the assessment. Please provide the methodology, assumptions, justification for the assumptions, and results used in the assessment. Diagrams with the location of the potential release points and their relationship to the "Stack" detector and the how the radioactivity is transported to the detector should also be provided.

<u>AADB Question 4</u>

In CPS's RAI response entitled "Supplemental Information Related to Design Basis Control Rod Drop (CRDA)," dated June 24, 2009, CPS stated:

"The results of the currently analyzed limiting CRDA event yield dose consequences well below the regulatory limits. The current limiting case of the condenser leakage release to the main control room (MCR) is 0.432 roentgen equivalent man (rem) (i.e., 4.568 rem of margin to 10 CFR 50.67(b)(2)(iii) limits). In addition, the following conservatisms in the CRDA analysis provide additional assurance that this TS change will not result in a CRDA event challenging 10 CFR 50.67 limits:

- Core source term used are those associated with 102% rated thermal power,
- A conservative peaking factor of 1.7 is assumed,
- A conservatively high value of 1% of the fuel in the pins with cladding damage are assumed to melt during a CRDA,
- The activity released from the fuel is assumed to instantaneously mix in the reactor coolant,
- No MCR or standby gas treatment system (SGTS) filtration is credited, and
- An additional 1650 cfm of MCR inleakage is added to the normal 3300 cfm MCR flow as an upper bound of allowable inleakage above that determined in the CPS LOCA calculation."

The NRC staff is concerned that the above argument utilizes margin up to the design bases limits without adjusting the assumptions in the design-basis analyses or providing a new analysis, and that the items cited as conservatism do not appear to be conservatisms. A discussion of the issues cited as conservatisms is given below.

The core source term based upon 102 percent rated thermal power is cited as a conservatism. Typically, the 102 percent power is not a conservatism, but is used because of the uncertainty in the ability to measure thermal power. Regulatory Guide 1.183 states that the uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, which is typically 1.02. If a value for core power is uncertain and may be 102 percent of the core power then the NRC staff does not see this as a conservatism.

Likewise, the peaking factor of 1.7 should be consistent with the peaking factors allowed in the core operating limits report. The reactor core is designed to and may operate with peaking factors up to 1.7. Therefore, if operation is allowed with peaking factors up to and including 1.7, the NRC staff does not understand how this is a conservatism.

While the NRC staff acknowledges that the activity released from the fuel is not instantaneously mixed in the reactor coolant, changing this assumption is expected to have a minimal impact on the results of the CRDA. Without specific data from the licensee for the timing of the releases to the reactor coolant, it is difficult to quantify the impact of this assumption.

The NRC staff acknowledges that no credit for the filtration for the MCR may be a conservatism, but data would need to be provided to justify this. Filtration is only credited when the MCR heating, ventilation, and air conditioning (HVAC) system is in filtered intake mode. NRC staff research of the CRDA input assumptions indicates that neither manual or automatic control room filtration has been justified for the CRDA, so currently no credit can be taken for the filtered intake mode until this has been justified.

The NRC staff acknowledges that the addition of 1650 cubic feet per minute (cfm) for the control room habitability calculation may be a conservatism, but additional data would need to be provided to justify this. Data from tracer gas testing during the normal mode of operation would be one potential method of justifying any conservatism associated with the assumption of 1650 cfm of additional inleakage during the normal mode of MCR HVAC operation.

Therefore, the NRC staff requests additional information regarding the argument that sufficient margin exists to mitigate the consequences of a CRDA. Since the above assumptions are part of the current licensing bases and the results of the CRDA are considered to be a significant input to the evaluations required by 10 CFR 50.59, credit for any conservatisms should be included in a revised CRDA analysis containing new results (doses). Please provide a revised assessment of the CRDA crediting any new input values and assumptions or justify why an assessment is not needed. Please provide a justification for any changes to the current CRDA analysis (i.e. inputs, assumptions or methodologies).

IOLB Question 1

All questions pertain to the spectrum of steam leak or break sizes that are ≥ 25 gpm but NOT large enough to cause MSIV isolation on high steamline flow (at approximately 2350 gpm). This spectrum of leaks or breaks would currently be prevented at CPS by MSIV isolation on high turbine building temperature, but would require operator action if the MSIV isolation on high turbine building temperature were eliminated.

Assuming the high turbine building temperature MSIV isolation is eliminated, for a small steam break or leak (at 25 gpm), what alarms and indications will be received by the control room operator, such that the break or leak can be properly diagnosed (e.g., high turbine building temperature alarms, common stack HVAC radiation alarms, increasing turbine building sump and equipment drain flows to radwaste)? At what time after a small break or leak starts will these alarms and indications be available? Please provide calculations which support these alarms, indications, and the times they occur. Also, please provide the same analyses for the spectrum of steam breaks between 25 gpm and 2350 gpm, and verify that operators will receive the necessary alarms and have enough time to manually close the MSIVs.

IOLB Questions 2

The NRC needs to fully understand the procedure(s) that will direct an operator to manually close the MSIVs (and drains) should a steam break or leak as described above occur. Please describe or provide to the NRC the associated current in-use CPS annunciator response procedures for any control room alarms that will be used in diagnosing a steam break or leak for the spectrum of break sizes from 25 gpm to 2350 gpm, and provide any follow-on off-normal procedures (e.g., offsite release off-normal, steam break/leak off-normal) that will direct MSIV closure. In addition, describe or provide to the NRC any proposed revisions to these procedures that would be put in-place if this amendment were approved (e.g., you propose to revise the turbine building high temperature annunciator response procedure if the amendment were approved).

IOLB Question 3

On June 24, 2009, the results of three simulator scenarios at CPS were forwarded to the NRC, regarding plant and operator performance during a large steam break (2350 gpm). Please provide this information again and on the docket. In addition, conduct the same scenario (large steam break, no automatic MSIV closure) again on a different crew of licensed operators, and evaluate plant and crew response, including response times.

IOLB Question 4

Please verify that the control room simulator, as used in the scenarios above, properly models expected plant performance during a large steamline break (2350 gpm). In particular, evaluate that the simulator provides the expected alarms and indications used by the operators to diagnose the steam break, and the timing of those alarms and indications, versus expected plant response.

EICB Question 1

By the letter dated June 24, 2009, the licensee stated that the Stack Process Radiation Monitors are qualified as Regulatory Guide (RG) 1.97, Revision 3, Category 3 variable. Under the proposed modification (change from automatic to manual action for main steam line isolation) is the licensee going to declare them as a RG 1.97, Revision 3, Type A (Accident Monitoring) variable? Is Function 7.f, Main Steamline Turbine Building Temperature – High, in TS Table 3.3.6.1-1, Primary Containment and Drywell Isolation, a safety function as described below? If so, the stack radiation monitors should be declared a Type A variable and meet the RG 1.97, Revision 3, Category 1 criteria.

As stated in RG 1.97, Type A variables are plant-specific variables that provide "primary information" required to permit the control room operator to take specific manually-controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBA events. "Primary information" is the information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

EICB Question 2

Provide plant procedures that specify the operator actions in the event of the actuation of alarms in the instrumentation mentioned above. If the licensee intends to modify them, provide the proposed modifications or describe the scope of the modifications.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for NRC staff review and contribute toward the NRC's goal of efficient and effective use of NRC staff resources. If circumstances result in the need to revise the
requested response date, please contact me at (301) 415-3154.

Sincerely,

/RA/ Marshall David for

Stephen P. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure:
Request for Additional Information

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