June 30, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES RELATED TO SURVEILLANCE FREQUENCY FOR SCRAM DISCHARGE VOLUME WATER LEVEL FLOAT SWITCH (TAC NO. MD4111)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 179 to Facility Operating License No. NPF-62 for the Clinton Power Station (CPS), Unit No. 1. The amendment is in response to your application dated January 26, 2007, as supplemented by letters dated June 6, October 11, 2007, and April 10, 2008.

The amendment revises the surveillance frequency for CPS technical specifications for the float switches of the Scram Discharge Volume Water Level – High Function, from every 92 days to every 24 months.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

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

Docket No. 50-461

Enclosures:
1. Amendment No. 179 to NPF-62
2. Safety Evaluation

cc w/encls: See next page

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Docket No. 50-461

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Package: ML081410234
Amendment: ML081410213
Tech Spec Pages: ML081410234

* safety evaluation dated

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. NPF-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by AmerGen Energy Company, LLC (the licensee), dated January 26, 2007, as supplemented by letters dated June 6, October 11, 2007, and April 10, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-62 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 179 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(2) This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: June 30, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Facility Operating License and Appendix “A” Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-62 Page 3	License NPF-62 Page 3
TSs 3.3-9	TSs 3.3-9

(4) AmerGen Energy Company, LLC, pursuant to the Act and to 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) AmerGen Energy Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) AmerGen Energy Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

AmerGen Energy Company, LLC is authorized to operate the facility at reactor core power levels not in excess of 3473 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 179 , are hereby incorporated into this license. AmerGen Energy Company, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-62

AMERGEN ENERGY COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated January 26, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070330254), as supplemented by letters dated June 6 (ADAMS Accession No. ML071580953), October 11, 2007 (ADAMS Accession No. ML072840565), and April 10, 2008 (ADAMS Accession No. ML081060254), AmerGen Energy Company, LLC (the licensee) requested changes to the technical specifications (TSs), facility operating license, and surveillance requirements (SRs) for Clinton Power Station, Unit No. 1 (CPS). The proposed changes would revise TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 8, "Scram Discharge Volume Water Level – High," item b, "Float Switch," by replacing SR 3.3.1.1.9 with SR 3.3.1.1.12. This change will effectively revise the surveillance frequency for the scram discharge volume (SDV) level float switch from every 92 days to every 24 months.

The June 6, October 11, 2007, and April 10, 2008, supplements contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

2.0 REGULATORY EVALUATION

2.1 Description of Structures, Systems, and Components

The RPS initiates a reactor scram when one or more parameters exceed their specified limit to preserve the integrity of the fuel cladding and reactor coolant system.

The SDV function is to receive water which is discharged from the control rod drives (CRD) during a scram. If at the completion of the scram, the level of water in the SDV is greater than the trip setting, the RPS cannot be reset until the discharge volume has been drained. In addition, the trip setting has been selected such that sufficient volume would be available to receive a full discharge of CRD water in the event that the SDV high level trip does not occur and subsequent scram protection is required.

Four non-indicating level switches (one for each channel) provide SDV high water level inputs to the four RPS channels. In addition, a non indicating level transmitter and a trip unit for each channel provide redundant SDV high water level inputs to the RPS. This arrangement provides diversity, as well as redundancy. Sensors are arranged so that no single event will prevent a reactor scram caused by SDV high water level.

2.2 Applicable Regulatory Criteria/Guidelines

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NRC Standard Review Plan (SRP), NUREG-0800, Section 19.2, "Use of Probabilistic Risk Assessment (PRA) in Plant-Specific Risk-Informed Decisionmaking: General Guidance." Section 19.2 states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following five key principles:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), the Commission established its regulatory requirements related to the contents of the TSs. Specifically, 10 CFR 50.36(d)(3), addresses the requirements for establishing the frequency of the surveillance tests. Section 50.36(d)(3) of 10 CFR states, "Surveillance requirements are requirements relating to test calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff reviewed the proposed TS changes against the 10 CFR 50.36(d)(3) requirements to ensure that there is reasonable assurance that the instruments affected by the proposed TS changes will perform their required safety functions in light of the extended surveillance interval.

NUREG-0800, Section 19.1, "Determining the Technical Adequacy of Probabilistic Results for Risk-Informed Activities," addresses the technical adequacy of a baseline PRA used by a licensee to support license amendments for an operating reactor. More specific guidance related to risk-informed TS changes is provided in NUREG-0800, Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications."

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002, describes a risk-informed approach, acceptable to NRC, for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines to be employed in the evaluation of proposed permanent licensing basis changes.

RG 1.174 also describes acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analysis used to support the proposed TS changes will remain valid. The monitoring program should include means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule, 10 CFR Part, 50.65, can be used when the monitoring performed under the Maintenance Rule is sufficient for the structures, systems, and components (SSCs) affected by the risk-informed application.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998, describes an acceptable risk-informed approach, including additional guidance geared towards the assessment of proposed permanent TS changes.

NRC Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle," dated April 2, 1991, provides guidance on use of instrument drift parameters to evaluate the adequacy of TS surveillance interval extension. The NRC staff evaluated the proposed SR extension against the criteria in GL 91-04 to ensure that there is reasonable assurance that the instruments affected by the proposed TS changes will perform their required safety functions in light of the extended surveillance interval.

RG 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3, dated December 1999, provides guidance on instrument setpoint calculations. The NRC staff evaluated the proposed SR extension against the criteria in RG 1.105 to ensure that there is reasonable assurance that the instruments affected by the proposed TS changes will perform their required safety functions in light of the extended surveillance interval.

3.0 TECHNICAL EVALUATION – PROBABILISTIC RISK ASSESSMENT

The NRC staff has reviewed the licensee's analysis in support of its proposed license amendment, which are described in the original submittal dated January 26, 2007, as supplemented by the licensee on June 6, October 11, 2007, and April 10, 2008.

3.1 Detailed Description of Proposed Change

The proposed change revises TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," SDV water level high float switch by replacing SR 3.3.1.1.9 with SR 3.3.1.1.12. This change will effectively revise the surveillance test interval (STI) for the SDV level float switch from every 92 days to every 24 months.

3.2 Review Methodology

Per SRP Sections 19.1, 19.2, and 16.1, the NRC staff reviewed the submittal using the five key principles of risk-informed decision-making detailed in RG 1.174 and RG 1.177.

3.3 Key Information Used in the Review

The key information used in the NRC staff's review is contained in the licensee submittal, and licensee response to the NRC staff request for additional information (RAI). The licensee's individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) were also used.

3.4 Comparison Against Regulatory Criteria/Guidelines

The NRC staff evaluation of the licensee's proposed amendment to extend SDV STI interval uses the principles outlined in RGs 1.174 and 1.177 as detailed in the following sections.

3.4.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3 and 5 of the NRC staff's philosophy of risk-informed decisionmaking, which concerns compliance with current regulations, evaluation of defense-in-depth, evaluation of safety margins, and performance measurement strategies.

- Key Principle 1: Compliance with Current Regulations

SDV water level is measured by two diverse methods: (1) by four transmitters with associated analog trip modules (ATMs), Function 8.a in TS 3.3.1.1 and (2) by four float type level switches, Function 8.b, in TS 3.3.1.1. The float switches are mechanical devices. The outputs from these devices are arranged such that there is one signal from a float switch and another from the transmitter with associated ATM to each trip logic division. In accordance with TS Table 3.3.1.1-1, SR 3.3.1.1.9 and SR 3.3.1.1.10 are required to be completed for the transmitters and SR 3.3.1.1.9 is required to be completed for the float switches. SR 3.3.1.1.9 requires that a channel functional test (CFT) is performed on each required channel every 92 days. SR 3.3.1.1.10 requires that each ATM be calibrated every 92 days.

The proposed TS changes will revise the surveillance test interval associated with the CFT of the SDV float switches and will not revise the surveillance testing of the transmitter and the associated ATMs. The transmitters will continue to be functionally tested every 92 days and the ATMs will continue to be calibrated every 92 days. In addition, a channel check is performed for both functions every 12 hours, in accordance with SR 3.3.1.1. One channel of each type of SDV Water Level – High Function associated with each of the four trip logic divisions is required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this function on a valid signal. Thus, the SDV high water level is measured by two diverse and redundant systems, is designed to be single failure proof, and tested on a more frequent basis during the operating cycle by other plant programs.

The licensee reviewed the impact of the TS SR change on the CPS licensing basis. The SDV receives water displaced by the motion of the control rod drive pistons during a reactor scram. Because insufficient volume to accept the displaced water will hinder the control rod insertion, a reactor scram is initiated when the remaining free volume is still sufficient to accommodate the water from a full core scram. The licensee stated that no credit is taken for a scram initiated from SDV water level – high function for any design basis accidents or transients analyzed in the CPS updated safety analysis report. Therefore, the NRC staff concurs with the licensee's assessment that the proposed TS change does not invalidate any assumptions in the plant accident analysis.

- Key Principle 2: Evaluation of Defense-in-Depth

There is no hardware or software (component or system design) change and because of that, there is no change in the defense-in-depth design of the plant.

- Key Principle 3: Evaluation of Safety Margins

Because there is no setpoint changes and no change in the system design, and because the affected instruments are practically drift free, the NRC staff concludes there is no change in the plant safety margins.

From the above evaluations the NRC staff concludes that the proposed TS SR change from 92 days to 24 months for SDV High Water Level – High Function meets the deterministic guidance provided in GL 91-04, RG 1.105, RG 1.174, and RG 1.177, and the requirements specified in 10 CFR 50.36(d)(3).

- Key Principle 4: Risk Evaluation

The licensee provided a risk-informed evaluation of the proposed change to extend the SDV level instrumentation STI. The risk metrics for the change in core damage frequency (CDF) and change in large early release frequency (LERF) were used to assess the risk impact of the proposed change, consistent with RG 1.174 and RG 1.177 guidance. Results of the staff review of the licensee evaluation are provided in section 3.4.2 of this safety evaluation (SE).

- Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

RG 1.174 also establishes the need for an implementation and monitoring program to ensure that an extension to STI does not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the proposed change continues to maintain the levels of reliability and availability of the SDV Float Switches. The evaluation of the Licensee's implementation and monitoring program is provided in Section 3.4.3 of this SE.

By the letters dated January 26, 2007, and June 6, 2007, the licensee stated that it has reviewed the CFT results for the previous 12 quarters during which the As-Found values of the four float switches were found to be within the acceptance criteria (½ inch difference between the Acceptable As-found and the Acceptable As-Left value) specified in the surveillance test procedures and required no readjustment.

The licensee also performed a review of the available calibration test results, the condition reports entered into the corrective action program, and the associated maintenance records. In response to the NRC staff RAI, by letter dated April 10, 2008, the licensee provided calibration data on the four SDV High Water Level Float Switches from 6/13/1990 to 9/18/2007 (a total of 46 calibrations) which shows no drift between the As-Left and As-Found values, except for two cases where the drifts were less than ½ inch, indicating that the associated instruments are practically drift free. The maximum calibration interval recorded was for 960 days with zero drift. This surveillance data provided more than 95/95 percent confidence level that the drift will be within the ½ inch acceptance criteria specified in the plant surveillance procedure CPS No. 9531.22. Furthermore, by letter dated April 10, 2008, the licensee confirmed that the affected float switches are monitored under the CPS Maintenance Rule program which assesses the effect of the proposed extension of the functional test interval. The licensee, also provided CPS procedures 9431.22 and 9531.22 on SDV High Water Level Float Switch C11-N013A (B, C, D) Channel Calibration, as well as Exelon procedure ER-AA-520, Instrument Performance Trending.

In response to NRC staff RAI, by letter dated April 10, 2008, the licensee confirmed that its industry operating experience review did not identified any Part 21 notifications for this model switch. The licensee also stated that there have been some failures due to pivot point contamination or wear. As a preventive measure, several years ago, the licensee created a preventive maintenance task that inspects the switch mechanisms for these float switches every 24 months. The switches are inspected for excessive wear on the actuating lever or misalignment of the adjustment screw at the point of contact between the screw and the lever. The licensee stated that thus far, it has not directly experienced any problems with these switches. From the above the NRC staff concludes that the instrument setpoint and drifts comply with the guidance in GL 91-04 and RG 1.105.

3.4.2 Staff Technical Evaluation (PRA)

<u>PRA Capability and Insights</u>

This part of staff review involves two aspects:

(1) Evaluation of the validity of the PRA and its application to the proposed change.

(2) Evaluation of the PRA results and insights based on the licensee's proposed application

PRA Technical Adequacy

The objective of the PRA technical adequacy review is to determine whether the CPS PRA used in evaluating the proposed change is of sufficient scope and level of detail for this application. Plant changes (e.g.; design changes, procedural changes, TS changes, etc) incorporated in the plant, and not yet in the CPS PRA model were evaluated by the licensee, and found to have minor or advantageous impact on PRA results.

The NRC staff also rechecked the insights and information previously gained from the CPS IPE and IPEEE studies. The CPS IPE and IPEEE were developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerability." The NRC staff concluded that the CPS IPE and IPEEE met the intent of GL 88-20. The licensee confirmed that all but one plant improvement identified in the IPE and IPEEE have been completed. All items identified were shown not to impact the proposed SDV STI evaluation conclusions.

Peer Review

The CPS PRA model was reviewed using the Boiling Water Reactor Owners Group certification team process in 2000. This review resulted in 5 “A” level Facts and Observations (F&Os), 92 “B” level F&Os, and 2 “C” level F&Os. All “A” level F&Os, all but 3 of the “B” level F&Os, and all but 2 of the “C” level F&Os have been resolved. None of the unresolved F&Os impact the proposed SDV STI evaluation or conclusions.

PRA Updates/ Procedures

The licensee stated in its submittal, that the CPS PRA model and documentation has been updated to reflect the current plant configuration, and to reflect the accumulation of additional plant operating history and component failure data. The internal events levels 1 and 2 CPS PRA model was used to perform the risk assessment part of the licensee submittal. The most recent CPS PRA version (CL 06C, March, 2007) was not used in the evaluation of the requested SDV STI extension, the previous version (CL 06B, November 2006) was used instead. CPS confirmed that the newer model has no changes that would affect the results for the proposed SDV STI. In addition, the licensee identified hardware and procedure changes not currently reflected in the model. None of the potential changes identified significantly impacted the proposed SDV level float switch STI risk evaluation including anticipated transient without scram (ATWS) event frequency. The licensee referenced plant procedures that provide the process for evaluating hardware and procedure changes and updating the CPS PRA.

The CPS PRA does not include detailed modeling of the RPS. Generally, the RPS is not modeled by PRAs in significant detail based on the redundancy, diversity, and demonstrated high reliability of the RPS. The SDV instrumentation has been shown to be an insignificant contributor to RPS unavailability and not a primary means of initiating an RPS trip. For CPS, high level system electrical and mechanical failures of the RPS are represented in the PRA. To further evaluate the contribution of SDV events to RPS failure, the licensee developed a comprehensive fault tree model of the SDV contribution to reactor protection system failure that incorporated the proposed 24-month STI. The results of this analysis were used to modify the RPS mechanical failure probability in the CPS PRA. The analysis did not credit any reduction in test caused transients based on the STI extension.

In the case of external events, the CPS IPEEE study was not updated. It was used by CPS as the basis for bounding the associated risk impact as detailed below.

Surveillance History

The licensee in the January 26, 2007, submittal, also states that a review of recent surveillance test history and condition reports entered into the corrective action program was completed to confirm that the affected SDV level float switch instrumentation have been conservative to the value specified in the TS for this instrumentation. The licensee’s surveillance test results from the past 12 quarters have been reviewed and it was determined that the as-found conditions of the four float switches have been within the acceptance criteria specified in the surveillance test procedure.

PRA Results and Insights

Based on the licensee's submittal dated January 26, 2007, the results of the requested change in the SDV level float switch STI shows an estimated change in CDF of 6.0 E-9/year, and an estimated change in LERF of 1.2 E-9/year. These values are based only on the contribution of the internal events portion of CPS PRA, but are within the RG 1.174 acceptance guidelines of 1E-6/year and of 1E-7/year for changes in CDF and LERF respectively. The risk impact associated with the proposed change was also evaluated via an NRC staff confirmatory high level risk assessment that also showed the risk impact to be very small.

External Events

In the supplemental information provided on October 11, 2007, CPS provided rationale for concluding the seismic risk impact of the proposed extension of the SDV STI as negligibly small. In the case of internal fires, CPS conservatively estimated that the fire induced risk would be approximately equal to that calculated for internal events, which makes the overall impact small compared to RG 1.174 criteria. In case of other external events, the CPS IPEEE indicated that high winds, flood, and other external events have negligible risk contribution to the plant and would not impact the proposed SDV STI.

Sensitivity Study

In the supplemental information provided in the June 6, 2007, letter, the licensee provided the results of a sensitivity study related to the impact of the extended STI on float switch performance and reliability. This analysis was performed using a failure rate over an order of magnitude higher than the rate used in the base case. Results of this study showed an increase in delta risk of a factor of three as compared to that of the base case. This result is still within the RG 1.174 acceptance guidelines. In addition, since 1.25 times the interval specified in TS 3.0, "Surveillance Requirements (SR) Applicability" SR 3.0.2, is allowed, the licensee also evaluated the risk impact of 25 percent increase in the requested 24 month extension, and found the incremental risk impact to be negligibly small compared to the base case result. Further, the licensee stated that no unique issues were identified for the proposed STI that would significantly change the assumed SDV level float switch constant failure rate such that the risk analysis results would be changed.

Based on the above, the staff finds that the licensee has satisfied the intent of RG 1.174, RG 1.177 and SRP Section 19.1, and that the quality of the CPS PRA is sufficient for the proposed change in the SDV STI. The sum of internal events and external events contribution to change in risk as a result of the requested STI extension is considered a very small change when compared to RG 1.174 acceptance guidelines for changes in CDF and LERF.

3.4.3 Implementation and Monitoring Program

RG 1.174 also establishes the need for an implementation and monitoring program to ensure that an SDV STI extension does not degrade operational safety over time and that no adverse degradation occurs due to an unanticipated mechanism or common cause factor. An implementation and monitoring program is intended to ensure that the impact of the proposed change continues to reflect the reliability and availability of SSCs impacted by the change.

The SDV water level float switches are monitored to demonstrate that their performance is adequate. The reliability of the float switches is monitored under the CPS maintenance rule program per 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.” The SDV water level float switch monitoring will assess the effect of the proposed SDV STI on level float switch instrumentation performance. Should the performance or condition of the float switches not meet the established performance criteria, appropriate corrective action will be taken per the requirements of the maintenance rule. Based on the above, the licensee has met the intent of RG 1.174 (Section 2.3).

3.5 Comparison with Regulatory Guidance

Limitations and conditions identified in RG 1.174 and RG1.177 specify acceptable changes in CDF and LERF. Acceptability of the licensee’s proposed change is discussed in the next section.

3.6 Staff Findings

The NRC staff finds that the licensee has demonstrated the acceptability of the proposed change in the SDV level switch STI from 92 days to 24 months. The PRA models used by the licensee were found to be acceptable for this application. The overall risk impact for the change in CDF and LERF as detailed in the licensee submittal Ref. [1,2 and 3] are found to be within the acceptance guidelines of RG 1.174 and RG 1.177 for a very small change. The SDV water level float switches are monitored to demonstrate that their performance remains adequate. The NRC staff, as a check, performed a confirmatory evaluation using a simplified model of the CPS RPS. Based on these results and the licensee’s analysis, the NRC staff concludes that the licensee has met the intent of RG 1.174 (Section 2.2.4 and 2.2.5), RG 1.177 (Section 2.4), SRP Section 19.2 and therefore the proposed SDV STI increase should result in only a very small increase in risk for CPS.

The licensee used PRA models acceptable for this application that included a comprehensive fault tree of the SDV level function. The overall incremental impact on risk, as detailed in the licensee submittal is found to be acceptable according to RG 1.174 acceptance guidelines for changes in CDF and LERF for a very small change. The SDV water level float switches are monitored to demonstrate that their performance remains adequate. Therefore, based on the above evaluation, the NRC staff concludes that the proposed amendment to extend the SDV STI from 92 days to 24 months is acceptable.

The proposed TS change will reduce personnel radiation exposure resulting from frequent containment entry at power. The staff evaluated the drift data and the plant procedures that the licensee provided for the SDV Water Level – High Function and finds that the float switches are practically drift free and because of that the proposed TS changes specified in Section 1.0 meet the deterministic guidance provided in GL 91-04, RG 1.105, RG 1.174 and RG 1.177, and the code requirements specified in 10 CFR 50.36(d)(3) and is therefore, acceptable.

The NRC staff has reviewed the licensee’s regulatory and technical analyses in support of its proposed license amendment which are described in Section 5.2 of the licensee’s submittal. The detailed evaluation above will support the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 28719; dated May 22, 2007). The NRC issued a proposed no significant hazards consideration on May 22, 2007 and, due to an administrative error, a second proposed no significant hazards consideration on May 20, 2008 (73 FR 29160). There were no changes to the NRC staff's initial proposed finding of no significant hazards consideration. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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