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U. S. Nuclear Regulatory Commission
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Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Additional Information Supporting the Request for Amendment to Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" Scram Discharge Volume Level Instrumentation Surveillance Requirements

References:

1. Letter from Mr. Thomas S. O'Neill (AmerGen Energy Company, LLC) to U. S. NRC, "Request for Amendment to Technical Specification 3.3.1.1, 'Reactor Protection System (RPS) Instrumentation' Scram Discharge Volume Level Instrumentation Surveillance Requirements," dated January 26, 2007

2. Letter from Mr. Darin M. Benyak (AmerGen Energy Company, LLC) to U. S. NRC, "Supplement to Request for Amendment to Technical Specification 3.3.1.1, 'Reactor Protection System (RPS) Instrumentation,' Scram Discharge Volume Level Instrumentation Surveillance Requirements," dated June 6, 2007

3. Letter from U. S. NRC to Mr. Christopher M. Crane (AmerGen Energy Company, LLC), "Clinton Power Station, Unit No. 1 – Request for Additional Information Related to Reactor Protection System Instrumentation Scram Discharge Volume Level Instrumentation Scram Discharge Volume Level Instrumentation Surveillance Requirements for the Clinton Power Station, Unit No. 1 (TAC No. MD4111)," dated August 9, 2007

In Reference 1, AmerGen Energy Company, LLC (AmerGen) requested an amendment to the facility operating license for Clinton Power Station (CPS), Unit 1. The proposed change is requested to revise the surveillance frequency for the scram discharge volume (SDV) level float switch from every 92 days to every 24 months. Reference 2 provided additional information requested by the NRC to support their review of Reference 1.

In Reference 3, the NRC requested that AmerGen provide additional information in support of their review of Reference 1. The attachment to this letter provides the requested information.

AmerGen has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. No new regulatory commitments are established by this submittal.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 11th day of October 2007.

Respectfully,

Darin M Benyak

Darin M. Benyak
Director – Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachment: Additional Information Supporting the Request for Amendment to Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" Scram Discharge Volume Level Instrumentation Surveillance Requirements

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Additional Information Supporting the Request for Amendment to Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" Scram Discharge Volume Level Instrumentation Surveillance Requirements

Request 1

On Attachment 1(page 4) of the submittal it was stated that the Clinton probabilistic risk assessment (PRA) model and documentation has been maintained current and is routinely updated to reflect the current plant configuration. NUREG-1560 shows Clinton's Individual Plant Examination (IPE) core damage frequency (CDF) to be ~2.7E-5 and large early release frequency (LERF) ~0.8E-6, while page 5 of the submittal shows CDF and LERF to be as low as 6.47E-6 and 1.65E-7 respectively. Discuss the major factors that led to this reduction in internal events risk.

Response 1

The reduction in industry calculated core damage frequency since the original IPE submittals is well documented. A report from NEI to the NRC (Reference 1) documents this trend. Since the industry IPEs were completed in 1992, the industry average CDF has dropped by nearly a factor of five. This risk reduction has been spurred by risk-informed initiatives (e.g., Maintenance Rule, Reactor Oversight Process) and by the following:

- continued improvement in plant performance,
- continued improvement in equipment performance,
- continued plant enhancements, and
- continued PRA Model Improvements.

In the 1992 Clinton Power Station (CPS) IPE, the CPS core damage frequency was calculated at approximately 2.7E-5/yr. As stated in this request, the CDF from the CPS CL06B PRA (i.e., the PRA model used in support of this amendment request) is calculated as 6.47E-6/yr. The CL06B CDF is approximately a factor of four reduction from the CPS IPE. This is consistent with the industry trend.

Two of the dominant contributors to this reduction are improvements in plant performance (i.e., lower initiating event frequencies in the PRA) and equipment performance (i.e., lower component failure and unavailability probabilities in the PRA). These aspects are discussed below with examples specific to the CPS IPE and CL06B PRA.

The transient initiating event frequencies used in the 1992 CPS IPE were estimated based on the higher expected plant transient frequencies consistent with plant performance in the 1990 time frame. The sum of the transient initiator frequencies in the 1992 CPS IPE was approximately 6.5/yr. In the CL06B PRA, the sum of the transient initiator frequencies is approximately 1.5/yr. The transient initiator frequencies in the CL06B PRA are due to improved performance across the industry and at CPS specifically. The CL06B transient initiator frequencies are based on a Bayesian update of more recent industry initiator frequency information with recent CPS experience. If the transient initiator frequencies from the 1992 IPE were inserted into the CL06B PRA, the CDF would increase by a factor of approximately two to approximately 1.3E-5/yr.

Another key factor to the CDF reduction is the reduction in equipment failure and unavailability probabilities. Many of the component random and common cause failure rates and maintenance related unavailability rates in the CL06B PRA are reduced from those in the IPE submittal. The CPS IPE generally used generic failure data from NUREG/CR-4550, "Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines," which is known to provide conservative failure rates estimates compared to current industry estimates.

The CPS PRA uses current industry data where available. The CPS plant specific component failure rates are based on Bayesian updates of the latest available generic industry data (e.g., NUREG-1715, "Component Performance Study," series of studies). As an example, the diesel generator failure to start probability in the CPS IPE was 3E-2/demand (based on NUREG/CR-4550); whereas the value in the CL06B PRA is approximately a factor of 4 lower (i.e., 7.95E-3/demand) based on a Bayesian update of industry generic data with recent CPS plant experience. The MOV failure to operate rate used in the CPS IPE was 3E-3/demand. The MOV failure to operate rates in the CL06B PRA are calculated separately for key systems using a Bayesian update of recent industry generic data with recent CPS plant experience. As an example, the Residual Heat Removal (RHR) system MOVs in the CL06B CPS PRA have a failure rate of 1.51E-3/demand, a factor of two lower than the value in the CPS IPE.

The common cause failure rates in the CPS PRA are likewise generally reduced from that in the CPS IPE. The CL06B PRA uses the latest common cause failure rate information (i.e., 2006 updated data from the NRC website) from Idaho National Engineering and Environmental Laboratory, now the Idaho National Laboratory (INL). As a result, key components (e.g., diesel generators, RHR pumps, batteries, etc.) have lower common cause failure rates in the CL06B PRA compared to the CPS IPE.

The reduction in LERF between the CPS IPE and the CL06B PRA is due primarily to the CDF reduction from improvements in plant performance and equipment reliability discussed above. The conditional probability of a LERF release (approximately 3%) is the same between the CPS IPE and the CL06B PRA.

Request 2:
With regard to both the IPE and individual plant examination of external events (IPEEE) confirm that plant improvements identified in the IPE and IPEEE have been implemented or do not impact the proposed scram discharge volume (SDV) level switch 24-month surveillance interval evaluation.

Response 2:
This license amendment request (LAR) involves a reactivity control system; its impact is measured through changes in Anticipated Transient Without Scram (ATWS) sequences. The potential changes identified in the IPE and IPEEE do not impact the preferred systems or methods for ATWS response and, therefore, have minimal impact on the results of this LAR. The proposed improvements from the IPE and IPEEE are shown in Table 2-1 along with the completion status and its impact on this LAR.

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Table 2-1
Summary of IPE and IPEEE Identified Potential Plant Improvements

Potential Plant Improvement	IPE or IPEEE	Status	Impact on SDV LAR
Operator training to emphasize importance of maintaining offsite power	IPE	Complete	n/a
Operator training to emphasize importance of manual ADS initiation	IPE	Complete	n/a
Modification to HPCS surveillance procedure to demonstrate unobstructed flow path from suppression pool	IPE	Complete	n/a
Installation of bypass line to allow easier use of fire protection system for vessel makeup	IPE	Deferred	No Impact on LAR: Fire protection alternate injection does not impact ATWS scenarios.
Evaluation of possible changes to training program beneficial to recovery of AC power supplies during LOOP	IPE	Complete	n/a
Provide additional procedural confirmation that shutdown service water pumps have started when required for diesel generator operation	IPE	Complete	n/a
Operator training to emphasize importance of maintaining offsite power related to preventing offsite releases	IPE	Complete	n/a
Operator training to emphasize importance of AC power recovery to preventing offsite releases	IPE	Complete	n/a
Operator training to emphasize importance of manually isolating containment bypass path into fuel pool cooling/cleanup line during station blackout	IPE	Complete	n/a
Operator training to emphasize significance of scram system hardware failures to release frequency	IPE	Complete	n/a
Procedures for DC load shedding during station blackout	IPE	Complete	n/a
Procedures for RCIC and HPCS operation during station blackout	IPE	Complete	n/a
Portable fan to cool main control room during station blackout	IPE	Complete	n/a
Installation of concrete barriers around all outside transformers	IPE	Complete	n/a
Reroute Div. 2 Nuclear System Protection System (NSPS) cabling	IPEEE	Complete	n/a

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Request 3:
The current revision of the Clinton PRA is identified as CL06B. Identify any plant changes (i.e., modifications, technical specification changes, procedures, etc.) not yet incorporated. Provide justification that these changes do not impact the proposed 24-month SDV level float switch surveillance interval risk impact.

Response 3:
CPS routinely evaluates procedure and hardware changes for their impact on the current PRA model. The process is described in Exelon Generation Company, LLC (EGC) Training and Reference Material (T&RM) ER-AA-600-1015, "Full Power Internal Events PRA Model Update." A review of the known installed or pending hardware and procedure changes not already incorporated in the CL06B PRA model found none of the potential changes significantly impacting the ATWS modeling in the CL06B PRA model. Therefore, these plant changes do not affect the conclusions of the SDV risk evaluation.

The results of the review of installed or pending hardware and procedure changes are summarized in Table 3-1 and Table 3-2, respectively.

Table 3-1
Hardware Changes Impacting the PRA Model
(Not Incorporated in the Model)

Hardware Change	Impact on Model and Application
DC breaker replaced with switch and fuse combination.	Possible minor impact on inadvertent breaker opening failure event. No significant impact on PRA results or ATWS modeling in particular.
Current offsite power connection is being modified to have three Reserve Auxiliary Transformers where there previously was one.	Will improve ability of offsite power system to deliver reliable offsite power and potentially provides flexibility for maintenance. Impacts AC power related sequences which are not significant contributors to ATWS events.
Alternate power supplies are being pursued for the hydrogen ignitors.	May improve the ability to deal with hydrogen production from power related accident initiators. Does not have an impact on the dominant ATWS sequences.
Low condenser vacuum trip of turbine bypass valves changed from a 1 out of 2 logic to 2 out of 2 for tripping the turbine bypass valves	Potential reduction in the likelihood that the turbine bypass valves lock out on spurious low condenser vacuum signal. This would result in a very minor reduction in CDF and does not have a significant impact on ATWS CDF.

Table 3-2
Procedure Changes Impacting the PRA Model
(Not Incorporated in the Model)

Procedure Change	Impact on Model and Application
Loss of ultimate heat sink procedure has been modified to make it a procedure for dealing with severe damage threats (e.g. terrorist attack). This change provides more means for accomplishing traditional PRA issues such as inventory makeup and RPV pressure control.	These potential response strategies could provide a reduction in core damage frequency if they can be utilized in core damaging sequences. They do not involve useful ATWS response strategies and therefore are not expected to impact this application.
CPS Emergency Operating Procedure (EOP) flow charts have undergone a minor revision.	Impact is expected to be minimal as the basic logic structure of the EOPs remains the same.
Changes in In-Service-Testing (IST) intervals.	Minor impact on failure rate data for associated components, in that it can impact the number of demands used in the Bayesian updating of data. Experience has shown this to have a relatively minor impact on PRA data.

The impact of any TS changes manifests itself in data changes that are captured in periodic PRA updates. As discussed in the response to Request 10, reasonable variability in failure data has no potential to change the conclusion that the risk increase of the proposed LAR is "very small."

Request 4:

External Events: On page 5 of Attachment 1, it is stated that external events are addressed qualitatively. However, no discussion of external events is presented in the submittal. Provide this discussion.

Response 4:

The discussion of external events was provided in Section 5 of Attachment 4 to Reference 2. The external events discussion provided in the submittal is concise and indicates that, given the non-significant risk impact of the proposed surveillance extension, explicit consideration of external event risk would not impact the conclusions of the analysis, and thus was not performed. This is consistent with the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

Discussions of the internal fires and seismic risk impacts of the proposed surveillance extension are provided in the responses to Requests 5 and 6 below.

In addition to seismic events and internal fires, the other following external hazard categories exist:

- high winds/tornadoes,
- external floods,
- transportation and nearby facility accidents, and
- other external hazards.

The CPS IPEEE submittal determined that these other external hazard categories are not significant risk contributors. In addition, similar to the discussions below for internal fires and seismic events, the risk impacts from these other external event hazards would not impact the decision making related to the proposed surveillance extension request.

Request 5:
Fires: The IPEEE NUREG-1742 showed Clinton's fire induced CDF to be 3.64 E-6, which is not negligible compared to that reported for Clinton's internal events. Has the Clinton IPEEE fire PRA been updated? What are the current updated results? Potential adverse impacts should be discussed (e.g., impact on relevant cables and instrumentation, compressed air system, and SDV valves).

Response 5:
The CPS IPEEE fire PRA has not been updated since its original issuance. A fire PRA for CPS is in the early stages of development. No current fire PRA results are available at this time.

The risk impact for the requested surveillance interval extension is due to a calculated increase in the scram failure probability due to a postulated increase in the latent failure rate of the SDV level instrumentation. Extension of the surveillance interval results in a theoretical increase (based on use of the standby failure probability statistical model) in the probability that the SDV level is excessively high at the time of the scram demand. As such, fire-induced failures of the scram system in response to a fire initiating event do not change the calculated results or conclusions of this LAR risk assessment. The functionality of the SDV level instrumentation and postulated high SDV water level preceding the scram demand are the focus of this analysis. The proposed LAR does not affect potential fire-induced adverse impacts on the SDV system following a fire-initiating event.

Fire-induced failure-to-scram scenarios include the following three types of cutsets.

- Fire initiating event * fire-induced failure of RPS * failure to achieve safe shutdown

- Fire initiating event * non-fire induced failures of RPS (exclusive of SDV level instrumentation) * failure to achieve safe shutdown

- Fire initiating event * non-fire induced failures of RPS due to latent failures of SDV level instrumentation * failure to achieve safe shutdown

It is only the third type of cutset that is impacted by the proposed LAR.

Based on NRC study RES/OERAB/S02-01, "Fire Events – Update of U.S. Operating Experience, 1986-1999," the industry average frequency for a fire using more current data and methods is approximately 1.5E-1/yr. This initiator frequency is more than an order of magnitude lower than the sum of internal events initiator frequencies. Even conservatively assuming a 1.0 conditional probability for failure to achieve safe shutdown given a fire-induced event with failure to scram, the change in fire risk due to the proposed LAR would be approximately equal to that calculated for the internal events.

Thus, the change in fire risk due to the requested SDV level instrument surveillance interval extension does not change the conclusion that the risk impact is "very small" in accordance with RG 1.174 criteria.

Request 6:
Seismic: Potential adverse impact of seismic events on SDV system (leak sizes, failure of valves, and clogging or failure of piping) was not discussed in the submittal. The rationale for neglecting this aspect should be provided.

Response 6:
The change in seismic risk due to the requested SDV level instrument surveillance interval extension is non-significant.

The risk impact for the requested surveillance interval extension is due to a calculated increase in the scram failure probability due to a postulated increase in the latent failure rate of the SDV level instrumentation. Extension of the surveillance interval results in a theoretical increase (based on use of the standby failure probability statistical model) in the probability that the SDV level is excessively high at the time of the scram demand. As such, seismic-induced failures of the scram system in response to a seismic initiating event do not change the calculated results or conclusions of this LAR risk assessment. The functionality of the SDV level instrumentation and postulated high SDV water level preceding the scram demand are the focus of this analysis. The proposed LAR does not affect potential seismic-induced adverse impacts on the SDV system following a seismic event.

Seismic-induced failure-to-scram scenarios include the following three types of cutsets.

- Seismic initiating event * seismic-induced failure of RPS * failure to achieve safe shutdown
- Seismic initiating event * non-seismic induced failures of RPS (exclusive of SDV level instrumentation) * failure to achieve safe shutdown
- Seismic initiating event * non-seismic induced failures of RPS due to latent failures of SDV level instrumentation * failure to achieve safe shutdown

It is only the third type of cutset that is impacted by the proposed LAR.

The CPS Operating Basis Earthquake (OBE) peak ground acceleration is 0.11g. Based on NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," the exceedance frequency of the CPS OBE is 5.5E-4/yr. The OBE "exceedance" frequency represents the frequency of all earthquake events with magnitudes greater than or equal to the OBE. Use of this low magnitude earthquake as the seismic initiating event is conservative for this discussion.

This initiator frequency is orders of magnitude lower than the sum of internal events initiator frequencies. Even conservatively assuming a 1.0 conditional probability for failure to achieve safe shutdown given a seismic event with failure to scram, the change in seismic risk due to the proposed LAR would be less than that calculated for the internal events.

As such, the change in seismic risk due to the requested SDV level instrument surveillance interval extension does not change the conclusion that the risk impact is "very small" in accordance with RG 1.174 criteria.

Request 7:

PRA Quality: Pages 5 of Attachment 1, and 3 of Attachment 4 have a brief discussion of PRA quality.

- *What was the rationale for concluding that the PRA quality is sufficient for this application?*
- *Is Clinton's PRA in compliance with published standards (e.g.; as referenced in RG 1.200)?*
- *Provide the results of the Clinton PRA independent peer review including the status of the peer review A, B, and C facts and observations (F&Os), and date of certification. Discuss the F&O applicability to the proposed SDV float level switch 24-month completion time.*
- *Reference procedures/documentation for maintaining and updating the PRA including revision history.*

Response 7:

The rationale for concluding that the PRA quality is sufficient for this application is that the change in ATWS frequency due to this application is small enough that the application would be able to meet the RG 1.174 criteria for ΔCDF and ΔLERF even if the plant had no mitigation capability for ATWS (i.e., had a conditional probability of core damage given ATWS of 1.0). While the plant has mitigating capability for ATWS events (e.g., Standby Liquid Control System with RPV inventory makeup systems), the construction of the base PRA model that covers this capability is not critical for concluding the application is acceptable.

The CPS PRA was reviewed by a certification team using the Boiling Water Reactor Owner's Group (BWROG) certification team process in 2000. From this review there were 5 "A" level Facts and Observations (F&Os), 92 "B" level F&Os, and 52 "C" level

F&Os. Since the certification team review, all the "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 these F&Os that are still awaiting resolution involve ATWS mitigation issues and do not affect the proposed SDV instrument interval extension.

The CPS PRA update process includes a self-assessment against the ASME PRA Standard (i.e., RA-Sa-2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications") to identify gaps to be addressed in PRA updates.

The process for performing updates of the Clinton PRA model is controlled under EGC T&RM ER-AA-600-1015. The revision history for the Clinton PRA model is as shown in Table 7-1 below.

Table 7-1
Revision History of CPS PRA Model

PRA Model Version	Issue Date	CDF (/yr)
CPS IPE	September 1992	2.6E-5
Revision 1	April 1994	9.9E-6
Revision 2	January 1995	6.0E-6
Revision 3	June 2000	2.67E-5
Revision 3a	December 2000	1.38E-5
Revision CL03A	August 2003	9.97E-6
Revision CL03C	May 2004	1.00E-5
Revision CL06A	March 2006	1.16E-5
Revision CL06B	November 2006	6.47E-6
Revision CL06C	March 2007	5.57E-6

Although the SDV surveillance interval extension request was specifically based upon the CPS CL06B PRA model, the more recent CL06C model did not have any changes relative to the treatment of ATWS modeling and therefore would have produced similar result for this LAR.

Request 8:
Given the Clinton SDV design details, how significant is the likelihood of control rod drive severe seal leakage into the SDV, exceeding the SDV drain valves capacity? What is the estimated impact on the time available for action in this case? See Attachment A, page A2 of the license amendment request.

Response 8:
The rate of Control Rod Drive (CRD) in-leakage and operator response have been evaluated for this request by assuming that the operator fails to initiate a manual scram for scenarios assuming high CRD in-leakage. That is, the operator fails to initiate a manual scram for Multiple Rod Drift and Loss of Instrument Air events.

Given these conservative assumptions regarding operator actions, the risk increases are about 50% larger than in the original analysis, but still remain well below regulatory acceptance criteria (i.e., over an order of magnitude smaller).

Request 9:
What information (instrumentation, or alarms) related to SDV level exists in the control room?

Response 9:
Each SDV has five level transmitters. One divisional transmitter supplies the "SDV Not Drained" alarm in the main control room. This annunciator alerts the operator that the SDV has not completely drained following a scram reset, or that leakage into the SDV has begun to accumulate. The same divisional transmitter supplies the SDV high level Rod Block to the Rod Control and Information System. This indication is also annunciated in the Main Control Room as a "Rod Out Block." The remaining four divisional transmitters supply the four Reactor Protection System (RPS) SDV Level High-High trip channels that initiate an anticipatory reactor scram while sufficient volume exists in the SDV to accommodate a full scram. The float type level switches actuate RPS Divisions A and B. Differential pressure cells employed as level transmitters actuate RPS Divisions C and D. The SDV high water trip signal is also annunciated in the Main Control Room.

Request 10:
Data used in the assessment (Reference 3, Attachment 1 and Reference 4, Attachment 4) are more than 10 years old. Were there any efforts to incorporate updated data? Would more current data change the results?

Response 10:
Failure probabilities for the events in the SDV fault tree are assigned based on the data analysis approaches and the plant specific and generic data used in the CPS CL06B PRA. The CPS PRA uses current industry data where available. The CPS plant specific component failure rates are based on Bayesian updates of the latest available generic industry data (e.g., NUREG-1715 series of studies). Generic industry data is used directly without Bayesian updating for those components for which plant-specific data is not readily available. The CPS CL06B PRA uses the latest common cause failure rate information from INL (i.e., the 2006 updated data from the NRC website). The Human Reliability Analysis (HRA) in the CPS CL06B uses standard industry HRA methodologies, supported with plant-specific timing information and interviews with Operations personnel.

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Review of the NRC's latest industry failure data study (NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U. S. Commercial Nuclear Power Plants," January 2007) shows that the probabilities of the events used in the SDV fault tree for this risk analysis are consistent with this latest reference source and in some cases the NUREG/CR-6928 estimates are significantly lower. For example, the hourly failure rate used in the CPS SDV fault tree for a level transmitter is 1.0E-6/hr, whereas the NUREG/CR-6928 industry mean is 1.0E-7/hr. The hourly failure rate for AOV spurious operation used in the CPS SDV fault tree is 1.5E-7/hr, which is consistent with the NUREG/CR-6928 industry mean of 1.8E-7/hr.

The data reference (i.e., NUREG/CR-5500, "Reliability Study: GE Reactor Protection System, 1984 - 1995," Vol. 3, May 1999) cited in this request was used for just two elements in the SDV level instrument surveillance interval extension risk assessment: the mechanical failure to scram base probability (i.e., 2.10E-6) and the electrical failure to scram base probability (i.e., 3.70E-6). The most current reference source for failure to scram data is NUREG/CR-5500, Vol. 3. The failure to scram probability is not a failure rate that is expected to receive continual updating based on new operating experience. As the frequency of failure to scram is very low, there will be no new industry data even over significant periods of time with which to update the analysis. The NRC's latest industry failure data study (i.e., NUREG/CR-6928) recognizes this by stating that the NRC/INL industry failure data studies are now updated annually *"except for those covering the reactor protection system".*

The overall risk impact of the requested SDV level instrument surveillance interval extension is so small (e.g., the delta CDF has 2-3 orders of magnitude of margin in the "very small" risk increase region of RG 1.174) that reasonable variability in failure data or human error probabilities has no potential to change the conclusion that the risk increase is "very small".

Request 11:
Section 4.1, "Surveillance History," discusses the maintenance history for the scram discharge float switches and states that surveillance test results for the last 12 quarters have met the surveillance test acceptance criteria. However, the discussion does not provide information on the impact the extended surveillance interval may have on level switch performance and reliability. Specifically, the staff is concerned that the proposed 24-month interval may introduce additional failure mechanisms, since the mechanical level switches will no longer be exercised quarterly and no active output is available from the level switches (channel check) over the proposed 24-month level float switch surveillance interval. Discuss how the level float switch failure probabilities used in the submittal reflect this concern, and how the uncertainty in this area is addressed in the submittal.

Response 11:
The risk analysis performed for the LAR submittal addresses the impact on level switch failure probabilities of the proposed surveillance test interval extension by use of the standby failure probability model to calculate the level switch failure probabilities. The standby failure probability model is a constant failure rate model where the failure

probability of a component is defined by the exponential equation $1 - e^{-\lambda t/2}$, where λ is the constant failure rate per hour and t is the test interval. The failure probability rises from a value of 0.0 immediately after a surveillance test up to a value of λt just before the next surveillance test. The probability used in fault tree modeling, which assumes that the component demand may occur randomly at any time between surveillance tests, is $\lambda t/2$ (i.e., the average failure probability between tests). This approach is consistent with the guidance in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications".

This request questions whether the requested surveillance test interval extension introduces additional failure mechanisms such that the assumed constant standby failure rate could be postulated to increase during the 24-month test interval. Regulatory Guide 1.177 recognizes this issue but also states that *"[e]xperience data are not available to assess the STI values beyond which the component failure rate, λ, increases."* Regulatory Guide 1.177 indicates that the effect of the constant failure rate assumption be treated with sensitivity studies.

Such a sensitivity study was not documented in the original LAR submittal given the very low risk impact of the requested surveillance interval extension. Discussion of the issues and a quantitative sensitivity study are provided here. The following issues were considered:

- time-related and demand-related failure contributions,
- effect of STI on constant failure rate assumption, and
- quantitative sensitivity assuming conservative increase in level switch failure rate.

NUREG/CR-6141, "Handbook of Methods for Risk-Based Analyses of Technical Specifications," states that a component failure rate is comprised of the following two contributions:

- standby time-related failure contribution, and
- cyclic demand-related failure contribution.

The former contribution is associated with failure mechanisms that can be postulated to occur while the component is in standby between tests. The latter contribution is associated with the failure mechanisms caused by the shock or stress of demanding the component. Decomposing the level switch failure probability into separate time-related and demand-related contributions would result in a lower calculated risk impact. Assuming the failure rate to be 100% time-related results in the maximum calculated risk impact. The Reference 2 risk assessment took a conservative approach (i.e., the affected SDV components are assumed to be all time-related failures).

The effect of standby time on the constant failure rate used in the analysis is assumed to be inherent in the generic industry data used. The level switch constant failure rate used in the LAR risk assessment is taken from generic industry data that includes component failure events that are associated with various environmental conditions, maintenance histories, life cycles, surveillance intervals, etc. No unique issues are identified for the proposed surveillance interval extension that would significantly change the assumed constant failure rate such that the conclusion (i.e., that the risk impact is "very small" in accordance with RG 1.174 criteria) of the risk analysis would be changed.

A conservative quantitative sensitivity is documented here to illustrate that even assuming increasing standby failure rates and conservative failure probability assumptions for the level switches does not change the conclusions of the analysis. This sensitivity quantification is summarized as follows.

- The conservative assumption used in the base analysis of 100% time-related failure rate is maintained in this sensitivity study.

- The standard PRA approach when using the standby failure probability model is to acknowledge that the component demand may occur randomly at any time between surveillance tests, such that the component failure probability is estimated using $\lambda t/2$ (i.e., the average failure probability between tests). This is the approach directed by NUREG/CR-6141, and used in the base analysis. This sensitivity study conservatively calculates the level switch failure probability assuming that all scram demands occur at the end of the surveillance test interval, such that the level switch failure probabilities (both for random and common cause failure basic events) are calculated using λt.

- Neither RG 1.177 nor its technical basis reference, NUREG/CR-6141, provide guidance for adjusting the constant failure rate assumption as a function of surveillance test interval. The level switch constant failure of 3.3E-7/hr used in the base analysis is increased here using information from the Department of Defense (DOD) Reliability Analysis Center (RAC) Non-electric Parts Reliability Data (NPRD-95) component failure database. The NPRD-95 database provides component failures for various environments. The failure rate for a level switch in a less than ideal (e.g., potential unheated building) fixed ground location is 4.7E-6/hr. This failure rate is used in this sensitivity to calculate the standby failure probabilities for the SDV level switch random and common cause failure basic events. This failure rate is over an order of magnitude higher than the failure rate used in the base analysis. It is also higher than the 95% percentile (assuming an error factor of 10 and a lognormal distribution) of the failure rate used in the base analysis.

Using the above conservative assumptions results in calculated risk increases about 3 times higher than in the original analysis, but still remain well below regulatory acceptance criteria (i.e., over an order of magnitude smaller). Therefore, the SDV level switch test extension would remain acceptable.

Request 12:
In addition, the proposed surveillance interval as stated in the submittal is 24 months. Discuss how the failure probabilities used in the submittal account for an allowable 1.25 times the interval specified in the TS 3.0, "Surveillance Requirement (SR) Applicability," SR 3.0.2.

Response 12:
SR 3.0.2 permits a 25% extension of the specified surveillance test interval. This extension facilitates scheduling and considers plant operating conditions that may not be suitable for conducting the surveillance (e.g., transient conditions).

Assuming 1.25 times the 24 months proposed surveillance interval would not change the conclusion of the LAR risk assessment that the risk impact of the proposed surveillance test interval extension is "very small" in accordance with RG 1.174 criteria. The calculated risk impact is so small that a 25% extension to the 24-month test interval used in the standby failure probabilities would not change the conclusions. In addition, if the 25% extension were applied in the risk assessment to both the 3-month and the proposed 24-month interval, the calculated risk impact would be very close to the result provided in the base analysis.

Request 13:
Section 1.0 of the submittal refers to the amendment request as "risk based," but references Regulatory Guide (RG) 1.174, which describes a risk-informed approach, acceptable to the U.S Nuclear Regulatory Commission, for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. The implementation of risk-informed decisionmaking is expected to meet a set of five key principles as described in RG 1.174, Section 2, "An Acceptable Approach to Risk-Informed Decisionmaking."

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.*

Specifically address the five key principles including the implementation and monitoring program for the proposed 24-month SDV float level switch.

ATTACHMENT

Additional Information Supporting the Request for Amendment to Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" Scram Discharge Volume Level Instrumentation Surveillance Requirements

Response 13:
AmerGen has previously provided the requested information in a letter dated June 6, 2007 (i.e., Reference 3). Specifically, the attachment to Reference 3 contains an evaluation of the proposed change against the five key principles, including the implementation and monitoring program, of risk-informed decision making from RG 1.174.

References:

1. M. Fertel (NEI) to D.E. Klein (Chairman, USNRC), "Updated Industry Report on the Safety Benefit of Probabilistic Risk Assessment (PRA)", dated July 18, 2007

2. Letter from Mr. Thomas S. O'Neill (AmerGen Energy Company, LLC) to U. S. NRC, "Request for Amendment to Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" Scram Discharge Volume Level Instrumentation Surveillance Requirements," dated January 26, 2007

3. Letter from Mr. Darin M. Benyak (AmerGen Energy Company, LLC) to U. S. NRC, "Supplement to Request for Amendment to Technical Specification 3.3.1.1, 'Reactor Protection System (RPS) Instrumentation,' Scram Discharge Volume Level Instrumentation Surveillance Requirements," dated June 6, 2007