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U.S. Nuclear Regulatory Commission
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Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Response to Request for Additional Information, Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

References:

1. Letter from Pamela B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated March 24, 2010.

2. Letter from Peter Bamford, U.S. Nuclear Regulatory Commission, to Michael J. Pacilio, Exelon Nuclear, "Three Mile Island Nuclear Station - Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Relocation of Surveillance Frequencies to a Licensee Controlled Program (TAC No. ME3587)," dated July 2, 2010.

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1). The proposed amendment would modify TMI Unit 1 TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled program. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. On June 17, 2010, draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On June 23, 2010, a teleconference was held between the NRC and Exelon to further discuss the additional information requested by the NRC. In Reference 2, the NRC formally issued the request for additional information. Attachment 1 to this letter provides a restatement of the questions along with Exelon's responses.

In addition, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed Technical Specifications Task Force] Initiative 5b," dated March 18, 2009, provided an optional insert to existing TS Bases to facilitate adoption of the TSTF traveler. The TSTF-425 TS Bases insert states as follows:

"The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program."

Recently several licensees submitting license amendment requests (LARs) for adoption of TSTF-425 have identified a need to deviate from this statement because it only applies to Surveillance Frequencies that have been changed in accordance with the Surveillance Frequency Control Program (SFCP) and does not apply to Surveillance Frequencies that are relocated to the SFCP but not changed. For Surveillance Frequencies relocated to the SFCP but not changed, the existing TS Bases description provides a valid description of the bases for the unchanged Surveillance Frequencies.

Therefore, upon implementation of the proposed change, where appropriate, the existing TS Bases information describing the bases for the Surveillance Frequencies will be relocated to the SFCP. This will ensure that the information describing the bases for unchanged Surveillance Frequencies is maintained. Also, relative to the Bases insert, Exelon proposes to replace the TSTF-425 Bases insert specified above with a revised insert that reads "The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program," as indicated on revised proposed TS/Bases pages provided in Attachment 2.

Exelon has concluded that the information provided in this response does not impact the conclusions provided in the original submittal (Reference 1).

This response to the request for additional information contains no regulatory commitments.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of July 2010.

Respectfully,

D. P. Helker

David P. Helker
Manager, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information
Attachment 2: Revised Proposed Technical Specifications/Bases Pages

cc:	Regional Administrator - NRC Region I	w/attachments
	NRC Senior Resident Inspector – TMI Unit 1	"
	NRC Project Manager, NRR – TMI Unit 1	"
	Director, Bureau of Radiation Protection – PA Department of Environmental Resources	"
	Chairman, Board of County Commissioners of Dauphin County	"
	Chairman, Board of Supervisors of Londonderry Township	"

ATTACHMENT 1

License Amendment Request

Three Mile Island Nuclear Station, Unit 1
Docket No. 50-289

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

<u>Response to Request for Additional Information</u>

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
APPLICATION FOR TECHNICAL SPECIFICATION CHANGE REGARDING RISK-INFORMED JUSTIFICATION FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE CONTROLLED PROGRAM
(ADOPTION OF TSTF-425, REVISION 3)**

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1). The proposed amendment would modify TMI Unit 1 TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled program. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. On June 17, 2010, draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On June 23, 2010, a teleconference was held between the NRC and Exelon to further discuss the additional information requested by the NRC. In Reference 2, the NRC formally issued the request for additional information (RAI). The questions are restated below along with Exelon's responses.

RAI-1

The LAR states that the changes presented are consistent with TSTF-425 and also includes a discussion of the differences in the application that result primarily from the custom TMI-1 TSs as compared to the STSs presented in TSTF-425 and NUREG-1430. The LAR, Attachment 4, "TSTF-425 (NUREG-1430) vs. TMI Unit 1 Cross-Reference," is provided to aid in the determination of consistency of the surveillances proposed for relocation as compared to TSTF-425. In order to verify that the surveillances proposed for relocation are consistent with TSTF-425 as the LAR asserts, the NRC staff requests that the licensee provide corresponding TSTF-425 cross references for the following surveillance frequencies proposed for relocation: Table 4.1-1, "Instrument Surveillance Requirements," Channel Description Nos. 11, 15, 17, 19e, 19f, 45, and 46.

RESPONSE

The corresponding TSTF-425 cross-references for the specified TMI TS Table 4.1-1 instrument channel descriptions are provided in the table below.

TMI TS Table 4.1-1		TSTF-425/ NUREG-1430 Equivalent	Comments
Item	**Description**		
11	"Reactor Coolant Pressure-Temperature Comparator"	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	STS Table 3.3.1-1, Item 5
15	"High Pressure Injection Analog Channels"	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	STS Table 3.3.5-1, Items 1 & 2
17	"Low Pressure Injection Analog Channels"	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	STS Table 3.3.5-1, Items 1 & 2

TMI TS Table 4.1-1		TSTF-425/ NUREG-1430 Equivalent	Comments
Item	**Description**		
19e	"Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)"	SR 3.3.15.1 SR 3.3.15.2 SR 3.3.15.3	
19f	"Line break isolation signal (ICCW & NSCCW)"	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	Line break isolation is a diverse method for Reactor Building Isolation. This is a TMI-specific signal which is redundant to a signal on Reactor Building high pressure and accomplishes the same function as STS Table 3.3.5-1, Item 3
45	"Loss of Feedwater Reactor Trip"	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	STS Table 3.3.1-1, Item 10
46	"Turbine Trip / Reactor Trip"	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	STS Table 3.3.1-1, Item 9

RAI-2

With reference to the LAR, Attachment 2, Table 2-1, each of the findings in the following table identified an issue or gap that, individually, might not significantly impact the results from a surveillance test interval (STI) risk evaluation performed via the NEI 04-10 methodology, but, when taken cumulatively, could prove significant. The NRC staff's concern associated with each is highlighted in italics. Please address whether, when taken cumulatively, their effects could prove significant to the risk evaluation for an STI TS change and, if not, why not.

RESPONSE

Subsequent to the LAR submittal, several of the gaps identified in this RAI were addressed and resolved. The following gaps have been resolved as described in the table below:

- IE-A5-01
- IE-A7-01
- LE-E4-01

Additionally, responses for the following three gaps are provided in the table below:

- IE-A4a-01
- QU-D5-01
- SC-C2-01

Based on the discussions provided in the table, these gaps are still considered to not impact the results of an STI evaluation.

A sensitivity calculation was performed to address LE-C8a-01. The sensitivity shows that there is no impact on the base model results, but additional sensitivities will be performed, if necessary, to support specific STI evaluations. Also, for QU-F5-01, the technical adequacy associated with this gap is accounted for in the NEI-04-10 process (see discussion in table).

This leaves only gap IE-A6-01 as not addressed; however, this is not expected to have an impact as described in the table below.

Since three of the gaps identified in the RAI are resolved, there are six open gaps remaining; four of these have no impact, and two will be addressed by sensitivities required by the NEI 04-10 methodology. As a result, there is no cumulative impact of these open gaps.

Finding	Issue/Gap	Status of Issue/Gap
IE-A4a-01	"The potential for common cause failures [CCFs] was included in examination of potential initiating events resulting from the systematic evaluation for potential initiating events." *As recommended per [Regulatory Guide] RG 1.200, Rev. 2, for Supporting Requirement (SR) IE-A6 (Capability Category [(CC)]-II), this examination should also include CCFs from routine system alignments that could result from preventive and corrective maintenance.*	The text of the comment provided for IE-A4a-01 in the LAR was misleading. In fact, the examination for potential initiating events did include common cause failures from routine system alignments that could result from preventive or corrective maintenance. Therefore, the italicized item is not an issue in the performance of STI evaluations.
IE-A5-01	"No documentation was found of incorporating: (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation; (b) events resulting in a controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation." *SR IE-A7 requires that, even if not documented, these events have to be incorporated.*	Subsequent to the LAR submittal, a new review was performed and documented for events meeting either (a) or (b) in SR IE-A7. The review covered events from January 1, 1990 to December 31, 2009. No new initiators were identified from this review. Therefore, this gap is resolved.

Finding	Issue/Gap	Status of Issue/Gap
IE-A6-01 and IE-A7-01	"No documentation was found of interviews with plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked ... No documentation of the review of plant-specific operating experience for initiating event precursors was found in the [probabilistic risk assessment] PRA notebooks." *Even if not documented, CC-II for both of these SRs requires that the interviews (SR IE-A8 [CC-II], with finding IE-A6-01) and reviews (SR IE-A9 [CC-II], with finding IE-A7-01) have been conducted.*	Subsequent to the LAR submittal, a new review was performed and documented for precursor events. The review covered events from January 1, 1990 to December 31, 2009. No new initiators were identified from this review. Therefore, gap IE-A7-01 is resolved. Recent interviews with plant personnel (IE-A6-01) are still outstanding. Based on completion and documentation of the review of plant-specific operating experience for precursors, previous (undocumented) plant personnel interviews, and other initiating event identification methods used for the TMI PRA, the likelihood of plant personnel interviews identifying additional potential plant-specific initiating events is low.
SC-C2-01	*SR SC-C2 requires that, even if not documented (or else still in the process of being documented), computer code "limitations or potential conservatisms" have to be addressed.*	For success criteria that were developed for the PRA, generally MAAP4 is used instead of using design basis success criteria. The overall conclusion from the EPRI MAAP Thermal-Hydraulic Qualification Studies was that MAAP had a wide range of applicability; however, a few limitations were identified. The current position on MAAP code limitations can be found on the MAAP4 web site. The significant limitation of MAAP for PWRs is Large LOCA behavior prior to reflood. The TMI PRA uses design basis criteria for Large LOCAs, so this limitation of MAAP4 has been addressed.

Finding	Issue/Gap	Status of Issue/Gap
QU-D5-01	"Some SSCs [structures, systems, and components] that are significant contributors to initiating events, but not to mitigation, are not explicitly identified in the documentation of significant contributors." *CC-II for SR QU-D6, against which this finding is cited, requires that significant contributors to core damage frequency, including initiating events, and SSCs and operator actions that contribute to initiating event frequencies, be identified. While "not explicitly identified" in the documentation, were these significant contributors to initiating events actually identified but just omitted from the documentation? If they were not identified, how were they known to be significant and to what extent?*	Significant contributors to initiating events were identified through a review of support system initiating event cutsets, but the individual contributors and cutsets were omitted from the quantification notebook. It should be noted that initiating event fault trees are re-quantified for any application affecting the components or configurations represented by these fault trees.
QU-F5-01	"[O]ther than the [large early release frequency] LERF truncation limitation, no evaluations of limitations were presented ..., [including] limitations of the model as they may apply to applications." *As implied by SR QU-F5, these limitations need to have been addressed.*	LERF truncation is the only identified limitation to the TMI PRA model for applications. Additional limitations may exist (e.g., STI components not modeled in the PRA), but the NEI 04-10 process (Step 8) requires an assessment of whether the STI change can be adequately characterized by the PRA.

Finding	Issue/Gap	Status of Issue/Gap
LE-C8a-01	"The Reactor Building fan coolers are undersized at TMI and have a little to no impact on containment [CNMT] pressure and temperature with respect to early containment failure." *SR LE-C9 (CC-II) requires justification for any credit taken for equipment survivability under adverse environmental conditions such that, even if the fan coolers were assumed to be failed, there would be "little to no impact" on CNMT pressure and temperature with respect to early CNMT failure.*	In response to this RAI, a sensitivity analysis was performed to determine the impact on the base (average) PRA model assuming the Reactor Building fan coolers were not available following core damage. There was no change to the LERF results (i.e., identical large early release cutsets and frequency). It is expected that this conclusion will be the same for most applications. However, there is still a potential that the assumption for the Reactor Building fan coolers surviving adverse environmental conditions may impact a specific STI evaluation being performed. To address this potential, the comment for this gap in the LAR states that it will be evaluated via a sensitivity analysis per NEI 04-10, if applicable to the STI.
LE-E4-01	"The level 2 results with the flag file are expected to be conservative. When the cutsets were reviewed, it was determined that there appears to be non-minimal cutsets in the level 2 model as quantified without the flag file ... Some sensitivities have been performed, although a conclusive determination has not been made regarding the current method for quantifying LERF ... ([T]he TMI model uses Forte 3.0c as the quantifier)." *SR LE-E4 requires that LERF be quantified consistently as with core damage frequency. This implies that the LERF quantification be conclusively determined as conservative, e.g., by quantifying LERF using Forte 3.0c at a greater truncation value just to assess whether the use of the flag file produces conservative results.*	Subsequent to the LAR submittal, this F&O/gap was resolved. A test was performed that was similar to that done by the peer reviewer. It determined that the reason for the higher FTREX results is because of the way that CAFTA calculates the total value of cutsets using Min Cut Upper Bound. The LERF results from FTREX without using the flag file have a significant number of events greater than or equal to 0.9. Using the EPRI Acube (beta) software, it was shown that the sum of the cutset values calculated without the flag file was less than when using the flag file. This is the result expected. Therefore, the method utilizing the flag file is conservative and acceptable.

RAI-3

With reference to the LAR, Attachment 2, Table 2-2, Finding DA-B2-01 states: "There is no evidence that the intent of this SR was met. Although the component failure rates are grouped by system and component type, that does not guarantee that outliers are not included in a group." SR DA-B2 (CC-II) requires exclusion of outliers in the definition of system/component failure groups. Were outliers appropriately excluded from group definitions? If not, will their exclusion be part of the sensitivity analysis for an STI evaluation?

RESPONSE

A review of the component grouping has subsequently been performed. There is no indication of outliers due to testing or operational characteristics (except potentially for manual valves), nor due to poor performance of certain components or systems. Operational characteristics (normal position and frequency of manipulation) for manual valves was not taken into account (i.e., for failure rate purposes, all manual valves were grouped together). However, the risk significance of manual valves is negligible; therefore, no impact on the results would be expected if they were grouped differently.

RAI-4

With reference to LAR, Attachment 2, Table 2-2, Finding IFEV-A5-01 states: "Several requirements in establishing flood initiating event frequencies are not met." Specifically cited are SRs IFEV-A5 through IFEV-A7, which require inclusion of plant-specific information and consideration of human-induced floods during maintenance (CC-II). Are any of the valves that may be assigned new STIs potential flooding sources, such that increasing the STI could increase the frequency of a flood due to miscalibration, etc., of one of these valves?

RESPONSE

Which valves, if any, are assigned new STIs using the NEI-04-10 process is unknown at this time. However, as indicated in the LAR submittal for this item, the methodology requires sensitivities for assumptions in the PRA model that may affect the results of the analysis or of any gaps to Capability Category II. This would lead to these issues being appropriately addressed for any valves associated with a surveillance interval change analysis.

REFERENCES:

1. Letter from Pamela B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated March 24, 2010.

2. Letter from Peter Bamford, U.S. Nuclear Regulatory Commission, to Michael J. Pacilio, Exelon Nuclear, "Three Mile Island Nuclear Station - Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Relocation of Surveillance Frequencies to a Licensee Controlled Program (TAC No. ME3587)," dated July 2, 2010.

ATTACHMENT 2

License Amendment Request

Three Mile Island Nuclear Station, Unit 1
Docket No. 50-289

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Revised Proposed Technical Specifications/Bases Pages

4-2a
4-47

Bases (Cont'd)

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked **in accordance with the Surveillance Frequency Control Program against a heat balance standard** and calibrated if necessary~~, every shift against a heat balance standard~~. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals ~~of each refueling period~~**specified in the Surveillance Frequency Control Program**.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth **in the Surveillance Frequency Control Program** are considered acceptable.

Testing

On-line testing of reactor protection channels is required ~~semi-annually~~ **in accordance with the Surveillance Frequency Control Program** on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel (Reference 1). **Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.**

~~The rotation schedule for the reactor protection channels is as follows:~~

~~a) Deleted~~
~~b) Semi-annually with one channel being tested every 46 days on a continuous sequential rotation.~~

~~The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested every 46 days. The frequency of every 46 days on a continuous sequential rotation is consistent with the calculations of Reference 2 that indicate the RPS retains a high level of reliability for this interval.~~

Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested **in accordance with the Surveillance Frequency Control Program**~~quarterly with one channel being tested every 23 days on a continuous sequential rotation. Calculations have shown that the frequency of every 23 days maintains a high level of reliability of the Reactor Trip System in Reference 4~~. The trip test checks all logic combinations and is to be performed on a rotational basis.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

d. The battery will be subjected to a load test ~~on a refueling interval basis~~ **in accordance with the Surveillance Frequency Control Program.**

(1) Verify battery capacity exceeds that required to meet design loads.

(2) Any battery which is demonstrated to have less than 85% of manufacturers ratings during a capacity discharge test shall be replaced during the subsequent refueling outage.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted ~~at least once each refueling~~ **in accordance with the Surveillance Frequency Control Program**:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the emergency power bus and energized. Upon completion of this test, the heaters shall be returned to their normal power bus.

(2) Demonstrate that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verify that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The intent of the ~~monthly~~ **periodic** tests is to demonstrate the diesel capability to carry design basis accident (LOOP/LOCA) load. The test should not exceed the diesel 2000-hr. rating of 3000 kW. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential overload condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The **S**~~s~~urveillance ~~specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails~~**Frequencies are controlled under the Surveillance Frequency Control Program**.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valve is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation.