SEABROOK

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SBK-L-14007
Docket No. 50-443

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Seabrook Station

Response to Request for Additional Information Regarding License Amendment Request 13-03, Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-13071, "License Amendment Request 13-03, Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," dated May 28, 2013. (Accession No. ML 13155A002)
2. U.S. Nuclear Regulatory Commission letter, Seabrook Station, Unit 1 – Request for Additional Information for License Amendment Request 13-03, "Application for Technical Specification Change Regarding Risk-Informed Justifications for Relocation of Specific Frequency Requirements to a Licensee-Controlled Program," dated December 11, 2013. (TAC No. MF1958)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted a request for an amendment to the Technical Specifications (TS) for Seabrook Station. The proposed amendment would modify Seabrook's TS by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04–10, "Risk-Informed Technical Specification Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies," using the Consolidated Line Item Improvement Process.

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NRR

This response does not modify the changes to the TS proposed in Reference 1 and does not alter the conclusion in Reference 1 that the changes do not present a significant hazards consideration.

Should you have any questions regarding this letter, please contact Mr. Michael Ossing, Licensing Manager, at (603) 773-7512.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on January 29, 2014.

Sincerely,

Kevin T. Walsh
Site Vice President
NextEra Energy Seabrook, LLC

Enclosure

cc: NRC Region I Administrator
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Enclosure to SBK-L-14007

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RAI #1

In Section 2.2 of the submittal, "Optional Changes and Variations," the licensee states in Item 3 that they propose relocation of surveillance frequencies for surveillances not covered by NUREG-1431, "Standard Technical Specifications Westinghouse Plants," and TSTF-425. The licensee further states that these changes will be evaluated using NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk- Informed Method for Control of Surveillance Frequencies" (Reference 3). Discuss what approach (risk, bounding, etc.) will be used and the estimated additional impact to risk, if any, these additions present.

NextEra Response

The TS changes for Westinghouse plants in TSTF-425 were based on the Improved Standard Technical Specifications (NUREG-1431). Additional surveillances are included because Seabrook Station has not converted to the Improved Standard Technical Specifications (ITS) and the Seabrook Tech Specs contain surveillance requirements that are not included in the ITS. These additional surveillances are identified in italic text in the Cross Reference Table included in Reference 1. These surveillances were either excluded or modified in the ITS or represent plant-specific features for which a TS surveillance was required (e.g., TS 3.7.5, "Verify valves operable for aligning startup feed pump"). As stated in Reference 1, NextEra has determined that the relocation of the frequencies for these Seabrook-specific surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

If changes to surveillance frequencies are proposed for these additional surveillances, either a qualitative risk analysis, a bounding quantitative risk analysis, or a more realistic quantitative risk analysis will be performed as applicable, consistent with the guidance of NEI 04-10 Rev 1

(Reference 3). The method of analysis will depend on whether the impacting components are modeled in the PRA and whether a bounding risk analysis is sufficient. The additional impact to risk from changes in the frequencies of these additional surveillances is expected to be minor and will be assessed consistent with the guidance of Reference 3.

RAI #2

Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, (Reference 4) states that, to demonstrate technical adequacy of the PRA, licensees should identify permanent plant changes (such as design or operational practices) that have an impact on those things modeled in the PRA, but have not been incorporated in the baseline PRA model. Please discuss Seabrook's approach to assessing plant changes not yet incorporated into the PRA model for this application.

NextEra Response

The Seabrook PRA model is updated periodically to reflect changes in plant design and operation. Changes that are identified in plant design and operation between periodic updates are tracked in the Model Change Database. Next Era Energy Fleet PRA procedures (EN-AA-105-1000) require that these pending changes be assessed for significance on the PRA. These pending changes are formally evaluated and incorporated into the PRA as appropriate during a subsequent PRA update and the model and documentation revised accordingly.

When proposed changes to surveillance frequencies are evaluated, the Fleet implementation process (EN-AA-105-1006) ensures that the Model Change Database be reviewed to identify any plant changes that have not been incorporated in the PRA. These pending changes will be addressed qualitatively or with a sensitivity analysis, as applicable.

RAI #3

The submittal indicates that the internal fire and seismic events portions of the Seabrook PRA have not been subjected to peer reviews or self-assessments. The submittal does not indicate how the PRAs will be utilized (e.g., detailed evaluations, bounding analyses). Please describe, in more detail, how fire and seismic events would be assessed in terms of NEI 04-10 guidance. If those portions of the PRA are to be used for detailed evaluations such as Step 12 of NEI 04-10, please demonstrate how they are technically adequate in accordance with Regulatory Guide 1.200 (Reference 4).

NextEra Response

The Seabrook PRA includes both internal fire and seismic events, along with internal events and internal flooding in an integrated-scope PRA. Quantitative risk assessments performed with the Seabrook PRA include all contributions to risk, including fire and seismic. This integrated-scope PRA will be used in any quantitative assessment of the risk from changes to surveillance intervals, whether that assessment is a bounding analysis or a detailed evaluation. The technical adequacy of the Fire PRA and Seismic PRA portions of the Seabrook PRA is addressed below.

<u>Fire PRA</u>

While the Fire PRA portion of the Seabrook PRA has not been assessed against the PRA Standard referenced in RG1.200 Rev. 2, the technical adequacy has been established by the experience level of the practitioners who performed the work and the detailed internal reviews that were performed. The fire PRA was performed for the original Seabrook PRA (1983), was revised for the IPEEE (1992), and was revised again in 2004. Each revision improved the fire risk assessment, included plant walkdowns, and utilized the latest fire ignition data.

Specifically, the current Fire PRA contains the elements of a fire PRA as specified in Reg Guide 1.200 Rev 1, Section 1.2.4, including screening analysis, fire initiation analysis, fire damage analysis, plant response model, and detailed quantification. The structure of the fire PRA is described below:

1. Definition of Fire Areas

The Appendix R fire areas and zones were used as a basis for the Seabrook FPRA since they include all safety related locations and some additional locations. Some areas / zones were combined for this analysis in order to streamline the screening process.

2. Development of a Spatial Database

A record for each key component within each fire area was created, which included the component location, power and control cable location, motive and control power supplies and locations, etc. Appendix R was the principle source for equipment location and cable routing. Information for equipment not addressed in Appendix R was obtained from plant arrangement drawings, walkdowns, and the CASP computer code cable routings.

3. Qualitatively Screening

Any areas which did not contain equipment or cables which could cause or mitigate an initiating event were screened out qualitatively.

4. Development of Fire Ignition Frequencies

For those areas that did not qualitatively screen out, fire ignition frequencies were developed. These frequencies used the latest industry fire frequency data for components and areas and applied them to Seabrook Station areas based on the actual types and numbers of components in the defined fire areas.

5. Quantitatively Screening

For this quantitative screen, it was conservatively assumed that all equipment and cables in the area failed at the calculated fire frequency. The resulting fire initiating events are compared to the existing internal events. If their contribution to the same internal event is negligible (less than about five percent), they are screened from the further analysis. If all possible fire initiating events in a specific area are screened out, the entire area was considered to be quantitatively screened out.

6. Detailed Fire Hazards Analysis

Areas remaining after the quantitative screening received detailed hazards analysis. The methodology for the hazards analysis was based upon the quantitative fire hazard equations and FIVE Methodology. The analysis includes:

- Identifying the location of critical equipment (i.e., target sets) and the severity of a fire needed to disable this equipment; and
- Estimating the frequency of a disabling fire based on ignition sources, estimated severity factor and detection and suppression system availability.

7. Plant Response Model

Impacts on the plant from failures of those target sets that the hazards analysis identified as potentially important were modeled in detail. This included evaluating operator actions needed to respond to the fire, such as control room evacuation and operation of equipment from remote locations.

8. Quantification

The fire PRA quantification is performed in RISKMAN with specific fire initiating events. The fire PRA is integrated into the internal events PRA and uses the same event tree structures, adjusted to account for fire impacts. The fire PRA accounts for about 11% of the total at-power CDF.

<u>Seismic PRA</u>

Similarly, the Seismic PRA portion of the Seabrook PRA has not been assessed against the PRA Standard referenced in Reg Guide 1.200 Rev. 2. The technical adequacy has been established by the experience level of the seismic risk analysts who performed the work and the detailed internal reviews that were performed. The seismic PRA was performed for the original Seabrook PRA (1983), was revised for the IPEEE (1992), and was revised again in 2005 to account for changes in the seismic hazard, more detailed equipment fragilities, including relay chatter, and additional walkdowns.

Specifically, the seismic PRA contains the elements specified in Reg Guide 1.200 Rev 1, Section 1.2.5, including seismic hazard analysis, structural and equipment fragility analysis, plant response analysis, and detailed quantification. The structure of the seismic PRA is described below:

1. Hazard Analysis

The seismic hazard analysis is the determination of the frequency of ground motions of various magnitudes at the site. A Seabrook site-specific probabilistic hazard curve was developed for the original SPRA (1983). The current PRA model has been updated to a more recent EPRI hazard (EPRI NP-6395-D, 1989), including the EPRI uniform hazard spectrum, developed for the Seabrook site. The EPRI uniform hazard spectrum was the basis for the revised equipment fragilities in the 2005 update. The sensitivity of different hazard inputs is addressed in the seismic PRA.

2. Fragility Analysis

The seismic fragility analysis is the determination of the seismically initiated ground acceleration at which plant structures and components are predicted to fail. The probabilistic estimates of seismic capacity of structures and components have been updated to reflect component-specific fragility information and a site-specific uniform hazard spectrum. Structures and components with a median capacity of 2.5g or greater were screened out of the analysis.

3. Plant Response Model

The plant response model is built from the seismic equipment list and integrates the results of the hazard and fragilities into the PRA model. The RISKMAN software is used to combine hazard and fragility inputs, develop initiating event frequencies, and fragility values at discrete hazard levels to yield conditional system failure probabilities. These are used to provide initiating event and split fraction values for the plant model. The PRA model event sequence trees used for internal events are used to integrate seismic initiators and seismic-initiated component failures with random (non-seismically caused) hardware failures and maintenance unavailability.

4. Quantification

The seismic PRA quantification is performed in RISKMAN with specific seismic initiating events. The seismic PRA is integrated into the internal events PRA and uses the same event tree structures, adjusted to account for seismic impacts. The seismic PRA accounts for about 25% of the total at-power CDF.

RAI #4

Attachment 1, Section 3.4, "Part 5 Seismic Events," states that the most recent 2005 upgrade to the Seabrook Seismic PRA (SPRA) did not include an update to the seismic hazard curve. The NRC issued Information Notice 2010-18, "Generic Issue [GI] 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" on September 2, 2010, to nuclear power plants and independent fuel storage installations. GI-199 activities are being addressed in the recently issued Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.54(f) (50.54(f)) letters on items 2.1 and 2.3 of the Japanese Near-Term Task Force recommendations. Please describe how NextEra ensures the Seabrook SPRA is technically adequate prior to completing the demanded seismic risk evaluation.

NextEra Response

The seismic hazard included in the current Seabrook Seismic PRA represented the best estimate hazard available for the Seabrook site until receipt of the new ground motion response spectrum (GMRS). This new GMRS was received at the end of September 2013 and will be included in a future Seismic PRA update. In the meantime, this new hazard curve will be treated similar to other new information that may impact the PRA model. The change in seismic hazard curves is tracked in the Model Change Database (MC#904).

When proposed changes to surveillance frequencies are evaluated, the Model Change Database will be reviewed to identify any plant changes or other new information that have not been incorporated in the PRA. These pending changes will be addressed qualitatively or with a sensitivity analysis, as applicable. Specifically, to address the potential risk from the new hazard curve, a qualitative assessment or a sensitivity study (using a scaling factor based on the change in the hazard) will be performed for any surveillance interval change. Such qualitative assessments or sensitivity studies will be performed until such time as the Seismic PRA is updated to include the new hazard.

RAI #5a

The following questions relate to facts and observations (F&Os) entries in Attachments A and B of the application which identifies gaps to Capability Category II of the American Society of Mechanical Engineers (ASME) PRA standard.

For F&O HR-G4-1, clarify the use of "time windows" and confirm that NextEra specifies the point in time at which operators are expected to receive relevant indications.

NextEra Response

The "time window" used in the human reliability analysis (HRA) refers to the time available for operator response from the time of the cue to the time that some irreversible condition occurs. In the Seabrook PRA, the time window is generally based on plant-specific thermal/hydraulic analyses for the specific accident sequence.

For each operator action, the HRA specifies the time of the cue, the point in time at which operators are expected to receive relevant indications to initiate the action.

In addition to the time window and the delay time related to the cue, the timing analysis in the HRA Calculator also includes the time required for the action, and the execution time to perform the physical steps of the action.

RAI #5b

For F&O HR-G7-1, for multiple human actions in the same accident sequence, confirm that NextEra accounts for the influence of factors that could lead to dependence such as common practices and procedures.

NextEra Response

For multiple human actions in the same accident sequence, the Seabrook PRA accounts for influence factors that could lead to dependencies. Operator dependencies are analyzed based on such influence factors as the sequence of actions (the first action is regarded as independent), the time frame (actions that occur in the long term are generally considered independent), and functional dependencies (an action that requires the success of a prior action is considered completely dependent).

Operator dependencies are then evaluated using a matrix that accounts for the possible combinations of actions in specific sets of sequences (e.g., ATWS, LOCA, SGTR, etc.). In many cases, the operator dependency analysis is simplified by assuming no credit for some operator actions. For example, with loss of offsite power and failure of the normally running PCC pump, the operator is modeled to start the standby PCC pump. However, if the initiating event is a LOCA, this action is assumed to be failed due to the complexity of the sequence and the potential for competing resources with other actions.

In other cases, complete dependence is assumed. For example, for a transient and failure of EFW pumps to start due to loss of the auto-start signal, an operator action is modeled to manually start the EFW pumps. If this action fails, no credit is given for operators manually starting the Startup Feed Pump due to the common function and the simplicity of the first action.

RAI #5c

For F&O LE-C10-01 for LE-C10 and LE-C12, confirm that NextEra conducted the review to achieve Capability Category II and used conservative or a combination of conservative and realistic treatment for nonsignificant accident progression sequences.

NextEra Response

The finding LE-C10-01 addresses the lack of documentation regarding the possibility of crediting continued equipment operation or operator actions in adverse environments of post containment failure. This involves attempting to provide a realistic assessment of accident sequences that end as LERF by appropriately crediting systems and actions in the Level 2 model.

This issue is tracked by the Model Change Database #883 and has been resolved, as described below. The resolution to this issue is documentation only, with no change to the LERF model.

<u>Resolution of F&O LE-C10-01</u>

The total LERF contribution as determined in the current model of record (SB2011) is 9.2E-08/yr. The ASME standard defines the significant accident progression sequences as those sequences that sum to 95% LERF or with the individual sequence percentage of 1% LERF. The significant accident progression sequences are identified below, based on initiating events that contribute 95% (summed) and 1% (individual) to LERF include the following:

Initiator	Description	LERF (per yr)	Contribution	Cumulative
SGTR	SG Tube Rupture	4.45E-08	48.3%	48.3%
LOC1VI	Interfacing Systems LOCA	1.80E-08	19.5%	67.8%
E25L	Seismic LOCA (2.5 g)	7.29E-09	7.9%	75.7%
E18L	Seismic LOCA (2.5 g)	4.36E-09	4.7%	80.4%
E18T	Seismic Transient (1.8g)	3.83E-09	4.2%	84.6%
E14A	Seismic ATWS (1.4g)	3.21E-09	3.5%	88.1%
E18A	Seismic ATWS (1.8g)	2.83E-09	3.1%	91.2%
E25A	Seismic ATWS (2.5g)	2.65E-09	2.9%	94.1%
E25T	Seismic Transient (2.5g)	1.80E-09	2.0%	96.1%
E10A	Seismic ATWS (1.0g)	1.32E-09	1.4%	97.5%
Other	Other LERF sequences	<1E-09	<1%	--

The above LERF initiators and associated significant accident progression sequences are reviewed for possible additional credit of equipment or operator actions that could be applied <u>during</u> the accident progression or <u>after</u> containment failure to further reduce the LERF contribution. The conclusions of this review apply to both significant and non-significant accident progression sequences. It is noted that non-significant sequences contribute minimally to LERF and therefore are less important when considering possible realistic/conservative measures to yet further reduce their already low LERF contribution.

<u>SGTR Sequences</u> – SGTR sequences contribute 4.45E-08/yr to LERF. The SGTR initiator models steam generator tube rupture occurring randomly. SGTR with subsequent failures can result in core damage with containment bypass. The top ranking SGTR sequences involve a tube rupture with successful SG cooling and primary system makeup but with failure of the main steam safety valve(s) to reseat, failure of the operator to terminate SI, and failure of decay heat removal via long-term primary inventory makeup including credit for decay heat removal using feed and bleed cooling. Also modeled is failure of the operator to establish feed to the faulted SG after core damage but prior to a significant release (scrubbing).

Possible additional actions to further reduce LERF are evaluated below:

- Use the fire suppression system as an additional means of injecting water into the ruptured SG to maintain level above the tube breach (effect scrubbing). However, this could only be accomplished if the pressure in the affected SG was low (below 100 psi) which is unlikely to occur in the short term. Therefore, credit for this action cannot be justified.
- Use the fire suppression system to spray water above the safety valve discharge; this would tend to reduce the release as a result of scrubbing. However, given the location of the safety valve discharge, equipment needed, limited timing, personnel habitability concerns, and limited scrubbing effectiveness, credit for this action cannot be justified.
- Apply a manual gagging device to the failed-open main steam safety valve. However, given the location of the main steam safety valves and potential for personnel habitability concerns, credit for this action cannot be justified.

Conclusion: The SGTR event sufficiently credits equipment and operator actions to realistically mitigate the SGTR contribution to LERF during the accident progression and containment bypass failure. There are no additional, practical mitigative actions or equipment that warrant further specific consideration and justification to further reduce the LERF contribution to less than 4.45E-08/yr. The existing SGTR LERF sequences and modeling are judged adequate and no further modeling changes are necessary to reduce LERF from SGTR.

LOC1VI Sequences – LOC1VI sequences contribute 1.8E-08/yr to LERF. LOC1VI models an interfacing system LOCA (ISLOCA) resulting in containment bypass sequences. The LOC1VI event occurs in the low pressure RHR system due to failure of dual normally closed discharge check valves in one of four injection lines. LOC1VI sequences involve failure of the RHR pipe/heat exchanger or failure of the RHR pump mechanical seal. Sequences that involve RHR pipe or heat exchanger failure are given minimal credit for mitigation because injection water is lost out of the break (located outside containment) and long term sump recirculation is not available. Sufficient time is not available to refill the RWST for long term injection. This break location is assumed to be high in the RHR Equipment Vault, thus the release path is outside containment without scrubbing through a pool of water in the equipment vault.

With the break location at the RHR pump mechanical seal, the release path is also outside of containment but with the benefit of scrubbing credited through a pool of water in the equipment vault. The RHR pumps are assumed to not survive the harsh environment given their location relative to the ISLOCA event. However, the Charging pumps are credited (located in a different building) along with the SI pumps (minimal credit due to their location upper elevation of the RHR vaults) for injection. Operator actions to depressurize the RCS (using SG depressurization and SG makeup with EFW) to limit the rate of RCS inventory loss, and to refill the RWST for long term RCS injection capability are modeled in the RHR pump seal failure event.

Possible additional action to further reduce LERF is evaluated below:

- Use the fire suppression system to spray water above the RHR pipe rupture (release scrubbing) or to add pool inventory to the RHR vault so as to submerge the pipe break opening to provide release scrubbing. However, given the location of the RHR vault and numerous possible locations for the pipe break within the vault, and personnel habitability concerns, it would be impractical to access the area to implement fire water spray or to flood the equipment vault. Therefore, credit for this action cannot be justified.

Conclusion: The ISLOCA LOC1VI event sufficiently credits equipment and operator actions to realistically mitigate the ISLOCA LERF during the accident progression and containment bypass failure. There are no additional, practical mitigative actions or equipment that warrant further consideration and justification to reduce LERF. It is noted that ISLOCA events are of very low frequency and subject to considerable uncertainty as a result of the state-of-knowledge correlations of common or similar components. Given these uncertainties, the existing ISLOCA LERF sequences and modeling are judged adequate and no further modeling changes to reduce LERF are needed.

Seismic Initiating Event (E) Sequences – Seismic event LERF sequences contribute a total of 2.73E-08/yr to LERF. The seismic initiators, E25L through E10A, represent extreme seismic events (1.0g and larger). These events are assumed to impair/delay effective evacuation and thus contribute to LERF. Without the assumption of impaired evacuation, these seismic event sequences would bin to "late" release bins. Sequences associated with these seismic events include large LOCA seismic initiators with failure of RCS inventory injection (accumulators and low-head injection), small LOCA seismic initiator with failure RCS inventory makeup (high-head injection), ATWS seismic initiator during unfavorable exposure time with resultant overpressure condition that ruptures the RPV, and transient seismic events with failure of decay heat removal equipment. For all of these extreme seismic events, the dominant failure mode of plant equipment is seismic-induced failure.

Possible additional action to further reduce LERF is evaluated below:

- Given containment failure, use the fire suppression system to spray water at the containment failure location to provide release scrubbing. However, given the severity of these seismic events, the fire protection system is also likely to fail. Therefore, credit for this action cannot be justified.
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Conclusion: Seismic event sequences sufficiently credit plant equipment and operator actions to realistically mitigate the seismic-induced LERF during the accident progression and containment failure. Given the severity of the seismic events represented by these sequences, there are no additional, practical mitigative actions or equipment that warrant further consideration and justification to reduce LERF.

RAI #6

Do the failure probabilities of structures, systems, and components that are in standby mode for extended periods, as modeled in the Seabrook PRA, include a standby time-related contribution and a cyclic demand-related contribution? Please describe how you address the standby time-related contribution for extended surveillances.

NextEra Response

The Seabrook PRA models the failure probabilities of standby components primarily as demand-related contributions, without a standby time-related contribution. However, when changes are considered to surveillance intervals, the unavailability models for impacted components will be converted from demand-related contributions to standby time-related contributions. This will allow the assessment of the increased risk due to the extended time between surveillances.

References

1. Letter from K.T. Walsh (NextEra) to NRC, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program," dated May 28, 2013. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 13155A002)

2. Technical Specification Task Force (TSTF) Traveler 425, "Relocate Surveillance Frequencies to Licensee Controi-RITSTF [Risk-Informed Technical Specification Task Force] Initiative 5b." (ADAMS Accession No. ML090850642)

3. NEI 04-10, Rev. 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." (ADAMS Accession No. ML071360456)

4. Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." (ADAMS Accession No. ML062060184)