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

April 20, 2011

Mr. Barry S. Allen
Site Vice President,
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1, LICENSE RENEWAL APPLICATION

Dear Mr. Allen:

By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew operating license NPF-003 for Davis-Besse Nuclear Operating Station, Unit 1, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Mr. Cliff Custer of your staff. The date for the response is within 60 days from the date of this letter. If you have any questions, please contact me at 301-415-2323 or e-mail at paula.cooper@nrc.gov.

Sincerely,

Paula Cooper

Paula E. Cooper, Project Manager
License Renewal Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
Requests for Additional Information

cc w/encl: Listserv

Request for Additional Information
Regarding the Analysis of Severe Accident Mitigation Alternatives
for the Davis-Besse Nuclear Power Station

1. Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

   a. Environmental Report (ER) Section E.3.1.1.2 explains that the SAMA evaluation is based on an updated version of the Davis-Besse Revision 4 PRA model that takes advantage of a 2008 "gap self assessment." This model, referred to as the "SAMA Analysis Model" represents a "freeze date" of July 9, 2009 for plant configuration, August 1, 2006 for component failure data and initiating event data, April 30, 2007 for equipment availability, and January 1, 2006 for non-Maintenance Rule unavailability. Identify any changes to the plant (physical and procedural modifications) since July 9, 2009 that could have a significant impact on the results of the PRA and/or SAMA analyses. Provide an assessment of their impact on the PRA and on the results of the SAMA evaluation.

   b. ER Section E.3.1.1.2 describes the PRA model history from 1993, when the IPE was issued, to July 2009 when the SAMA Analysis Model became effective. This section specifically discusses the model updates to Revision 2, 3, 4, and the SAMA Analysis Model. This section does not discuss the model revision from the IPE to the Revision 0, when the largest decrease in internal events CDF occurred (i.e., a decrease from 6.6E-05/yr to 1.4E-05/yr), or the update to Revision 1. Also, the reason for the drop in internal events CDF between the Revision 3 and 4 PRA models of approximately a factor of three is not apparent from the model update discussion. Provide a discussion of the PRA model changes that most impacted the change in total internal events CDF for the Revision 0, 1, and 4 PRA models. Also provide the effective dates of the Revision 0, 1, and 2 PRA models.

   c. Provide a brief description of the quality control process used for controlling changes to the PRA, including the process of monitoring potential plant changes, tracking items that may lead to model changes, making model changes (including frequency for model updates), documenting changes, software quality control, independent reviews, and qualification of PRA staff.

   d. ER Section E.3.1.1.2 identifies a Babcock and Wilcox (B&W) owner's group peer review of the internal events Level 1 and LERF PRA models performed on November 8, 1999 and states that no Level A and 18 Level B supporting requirements findings were identified. The ER further explains that following the review a Revision 3 PRA was issued to "close gaps to the draft industry standards." It is not clear from this statement whether all Level B findings were resolved by the Revision 3 PRA model. Section E.3.3 of the ER also discusses a B&W owner's group peer review that was finished in March 2000 which states that there were no Level A findings, and presents 5 Level B findings, three of which are closed and two that are still open. It is not clear whether this is the same B&W owner's group peer review comments described in Section 3.1.1.2, and if it is, why there are discrepancies in the two descriptions. The ER also states that in 2008 a "gap self assessment" was performed using a team of industry peers and internal staff

ENCLOSURE

that identified four Level A findings and 23 Level B findings associated with not meeting Capability Category 2 requirements of the 2005 ASME PRA standard. It is not clear from the description what the scope of this "gap self assessment" included. The ER does not identify any other peer reviews, technical reviews, or self assessments of the PRA. In light of these issues, provide the following:

i. Clarify whether there were one or two B&W owner's group peer reviews performed in late 1999 and early 2000 and the differences (e.g., scope) between these reviews if there were two. Clarify whether any Level A or B findings remain unresolved from this peer review (or these peer reviews) and if so, provide an assessment of their impact on the SAMA evaluation.

ii. Clarify the scope of the 2008 "gap self assessment" including whether it covered Level 1 and 2 internal events, internal flooding, and the high winds hazard. Also, identify the open Level A and B findings from this self assessment and provide an assessment of their impact on the SAMA evaluation.

iii. Provide a summary of the scope of any other PRA model internal and external reviews, a discussion of each unresolved finding, and an assessment of the impact of all unresolved findings on the SAMA evaluation.

e. ER Section E.3.1.1.1 states that the Davis-Besse Level 1 PRA internal events CDF is estimated to be 9.2E-6/yr, but further explains that if high winds and internal flooding is included that the CDF is estimated to be 9.8E-6/yr. Regarding the internal events CDF, provide the following:

i. The ER provides a caveat about the "tornado high winds" analysis in Section E.3.1.2.3 saying that the model does not include tornado-generated missiles. Based on the top 100 cutsets presented in Table E.5-1, the contribution to the total CDF from tornadoes does not appear to be significant (i.e. Cutset #1 = 3.0E-8/yr, #30 = 2.8E-8/r, #69 = 1.2E-8/yr, and #87 = 1.2E-8/yr). The NRC staff notes that the contribution to the internal events CDF from internal flooding is typically included in the internal events CDF whereas the contribution from high winds is generally not included. In light of this and given the high winds analysis is not complete, provide the internal events CDF including flooding but excluding high winds.

ii. ER Table E.3-1 presents dominant internal event sequences by initiating event and their percentage contribution to CDF that includes a contribution from internal flooding (i.e., F3AM and F7L). The calculated contribution percentages in Table E.3-1 appear to be based on a CDF of 9.2E-06/yr. This is consistent with the CDF reported in Section E.3.1.1.1 for the internal events CDF that does not include internal flooding and external wind, rather than the CDF of 9.2E-06/yr that does includes internal flooding and external winds. Clarify this apparent discrepancy. Also, clarify which model the Level 2 PRA was based on (i.e., with or without inclusion of internal flooding and external wind).

f. In ER Table E.3-1, initiating event T2B-1 listed as "SP6A fails to throttle" and T2A-1 listed as "SP6B fails to throttle" appear to have mismatching nomenclature and descriptions. Also it is not clear which valves are being referred to or what their function is in the plant. Initiating event T2A-2 listed as "FICICS35B fails high" and T2B-2 listed as "FICICS35A fails high" also appear to have mismatching nomenclature and descriptions. It is also unclear for these initiating events which components are being referred to or what their function is in the plant. Clarify these apparent discrepancies and provide layman descriptions for these four initiators.

2. Provide the following information relative to the Level 2 analysis:

   a. ER Section E.3.1.1.1 states that the Level 1 PRA quantification was performed using a "truncation cutoff" of 5E-13/yr, but no reference is made to the Level 2 truncation cutoff. Provide the Level 2 PRA truncation cutoff.

   b. ER section E.3.2.1 states that "The CET provides the framework for evaluating containment failure modes and conditions that would affect the magnitude of the release." The ER also explains that "The probabilities of the CET end states were quantified for each PDS." However, the Containment Event Tree (CET) is not presented in the ER nor is a description of its structure and composition provided. Provide the CET or a description of the CET used in the Level 2 analysis. Include in the response a discussion of how the CET top events were selected and how branch points probabilities were determined, including how phenomenological versus system failure mode branch point probabilities were determined.

   c. ER Section 3.1.1.2 states that an explicit LERF model was added to the PRA. ER Section 3.2.1 states that 14 additional PDSs were added to better define the status of certain containment systems. Clarify how the Level 2 model used in the SAMA evaluation differs from the IPE analysis.

   d. Identify the version of MAAP used in the SAMA analysis.

   e. Identify the release categories that compose the large early release frequency (LERF) from those presented in Table E.3-4 (Release Categories 1.1 through 9.2). Confirm that the identified release categories are those reviewed in Table E.5-3 (Basic Event LERF Importance).

3. Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

   a. For each of the four dominant fire areas identified in ER Section E.3.1.2.1, provide the following:

      i. Explain what measures have already been taken to reduce risk. Include in the response specific consideration of improvements to detection systems, enhancements to suppression capabilities, changes that would improve cable separation and drain separation, and monitoring and controlling the quantity of combustible materials in critical process areas.

ii. Review to identify potential SAMA candidates to reduce fire risk. Provide a Phase I and II assessment, as applicable, of each SAMA candidate. If no SAMA candidates are identified, explain why the fire CDF cannot be further reduced in a cost effective manner through implementation of SAMAs specific to fire events.

b. ER Section E.3.1.2.1 presents the four fire areas identified in the IPEEE that had an estimated CDF above the screening criteria of 1E-06/yr. It also presents the summation of those fire area CDFs to be 2.5E-05/yr which is then used as the basis to develop an external events multiplier. The IPEEE SER (Enclosure 3, Section 2.1.7) explains that the total frequency of the fire area CDFs which had been screened out after detailed analysis (some of which had revised CDFs greater than 1E-06/yr) is 3.8E-06/yr, which results in a total fire CDF of 2.9E-05/yr. Identify the fire compartments that were screened after detailed analysis and the corresponding CDFs and provide a review of these fire compartments for potential SAMAs.

c. ER section E.3.1.2.4 presents the basis for an external events multiplier of 3 based on a "conservatively" estimated fire CDF of 2.5E-05/yr developed using the FIVE methodology and the assumption that a "realistic" fire CDF is a factor of 3 less than this FIVE-produced fire CDF. The NRC staff disagrees that a fire CDF produced using the FIVE screening methodology is necessarily conservative in light of more recent research and guidance on hot short probabilities, fire ignition frequencies, and non-suppression probabilities (i.e., NUREG/CR-6850). The NRC staff particularly notes that the minimal or non-treatment of hot shorts in the IPEEE FIVE analysis may more than offset other conservatisms in the FIVE analysis. Based on this, and the previous RAI, the NRC staff believes the best estimate of the fire CDF for Davis-Besse is 2.9E-05/yr. In addition, the USGS issued updated seismic hazard curves for much of the U.S. in 2008. Using this data, the NRC staff estimated a "weakest link model" seismic CDF for Davis-Besse of 6.7E-06/yr (see NRC Information Notice 2010-18 regarding Generic Issue 199). Based on a fire CDF of 2.9E-05/yr, a seismic CDF of 6.7E-06/yr, and an internal events CDF of 9.8E-06/yr, the NRC staff estimates the external events multiplier to be 3.6. In light of this, provide a revised SAMA evaluation using an external events multiplier of 3.6 or alternatively provide justification for an evaluation of a different multiplier based on this updated USGS information.

4. Provide the following information concerning the Level 3 analysis:

a. Regarding ER Section E.3.4.7, clarify that the core inventory is based on the rated thermal power of 2,817 MWt and, if not, provide justification for the thermal power used.

b. Table 2.6-1 identifies that the year 2000 population living within the 50-mile site boundary is 2,375,624. Table E.3-11 identifies that the escalated population to year 2040 is only 2,227,192. The year 2040 population was stated to be a 4.7% escalation per decade from year 2000. Clarify this discrepancy. Also, in ER Section E.3.4.2, the statement that actual population within the 50-mile radius decreases appears to be incorrect. This statement appears to apply only to the US population groups within a

20-mile radius. Clarify that this understanding is correct.

c. Three SECPOP2000 code errors have been publicized, specifically: 1) incorrect column formatting of the output file, 2) incorrect 1997 economic database file end character resulting in the selection of data from wrong counties, and 3) gaps in the 1997 economic database numbering scheme resulting in the selection of data from wrong counties. Address whether these errors were corrected in the Davis-Besse analysis. If they were not corrected, then provide a revised cost-benefit evaluation of each SAMA with the errors corrected.

d. ER Section E.3.4.6.2 does not identify the population base/year reference for the emergency planning zone (EPZ) evacuation speed. Describe how/whether the EPZ evacuation time was corrected for the year 2040 population (and address the population discrepancy noted in RAI 4.b).

e. In ER Section E.3.5.2.3, for Case A1, identify the heat release energy (e.g. thermal, 1 MW) assumed for both the base and sensitivity cases.

5. Provide the following with regard to the SAMA identification and screening process:

a. ER Section E.5.2 describes major contributors to plant CDF, suggested improvements from the IPE study, and specific SAMA candidates identified to address the major contributors and suggested improvements. In addition to the suggested improvements identified in the ER, the IPE (in Section 3, Other Potential Plant Improvements) identifies four potential plant improvements related to the "back-end analysis": 1) BWST level at switchover to sump recirculation, 2) operator actions for inadequate core cooling, 3) emergency plan evacuation criteria, and 4) monitoring of carbon monoxide levels in containment. Describe the status of the implementation of each of these suggested improvements and identify and assess SAMAs to address each unimplemented improvement.

b. ER Section E.5.2 indicates that no plant-specific vulnerabilities that would affect the PRA CDF were identified in the IPEEE. NRC staff notes that the IPEEE safety evaluation report (Section 3.0, of the seismic attachment) states that "The aggregate of the material provided in the submittal and the licensees response to the RAIs is not quite sufficient to meet NUREG 1407" but that "The license did provide an incomplete list of HCLPF values for the plant, with the lowest HCLPF value being 0.26g" and so concluded that the submittal "did come close to meeting the objectives of a focused scope analysis." A FirstEnergy response to an NRC staff RAI on the IPEEE dated May 25, 2000 identifies a number of plant components with high-confidence low probability of failure (HCLPF) values less than 0.3g:

- Borated Water Storage Tank roof from sloshing (0.28g)
- Masonry Wall No. 2367 associated with 480 V Essential MCC (0.26g)
- Masonry Wall No. 3407 associated with Component cooling water room (0.27g)
- Masonry Wall No. 4786 associated with Essential Distribution Panel "D2N" (0.27g)

- Masonry Wall No. 6107 associated with Control Room Emergency Vent Fan Temperature Switch (0.29g)

Discuss whether plant improvements to meet 0.3g for these components has been implemented at the plant and, if not, identify and evaluate SAMAs to improve the seismic capacities of each of these components.

c. None of the SAMA candidates identified in Table E.5-4 appear to be plant-specific SAMAs identified from plant-specific risk insights based on the current PRA model. Clarify how the importance lists were used to develop plant-specific SAMA candidates and justify the apparent absence of any plant-specific SAMA candidates. Also, the basic events identified in importance analysis Tables E.5-2 and E.5-3 are not linked to SAMA candidates. Sections E.5.4 and E.5.5 only discuss the SAMA candidates identified to address basic events with high risk reduction worth (RRW) values. Identify, for each basic event having a RRW benefit value (averted cost risk) greater than the minimum cost of a procedure change at Davis-Besse, the specific SAMA(s) that address each event and describe how the SAMA(s) address the basic event. Identify and evaluate SAMAs for basic events not addressed by an existing SAMA (e.g., flooding related basic events and initiators, including WHAF3ISE, SHAF2ISE, F3AM, and F7L). For any basic event for which no SAMA is identified, provide justification for not identifying a SAMA(s).

d. ER Section E.5.3, E.5.4, and E.5.5 discuss significant contributors to core damage frequency (CDF) and large early release frequency (LERF). These sections and the associated tables show that there are a number of operator errors and non-recovery actions that occur in these listings, but report that no weaknesses in training or procedures were identified. Given: 1) the significant number of operator errors in these lists, 2) that human errors are among the most dominant failure modes presented in the importance Tables E.5-2 (i.e., the first 9 basic events listed by RRW are human error events) and E.5-3, and 3) that operator errors often have relatively high failure probabilities, provide the following:

   i. Explain the process used to make the determination that there were no opportunities to improve procedures and training.

   ii. Discuss whether any of the risk significant operator action failures could be addressed by a SAMA to automate the function (i.e., automating tripping of the RCPs after a loss of seal cooling – see RAI 7.a).

e. Table E.5-2 identifies events QMBAFP11 and QMBAFP12 representing unavailability of Auxiliary Feedwater (AFW) Trains 1 and 2, respectively, due to maintenance. Provide an evaluation of a SAMA to improve the availability of the AFW pumps by making improvements to maintenance practices or by making hardware modifications.

f. Table E.5-4 does not provide the source for identifying SAMAs CC-19, CW-24, and CW-25. ER Section E.5.2 implies that CW-24 and CW-25 were identified to address IPE risk insights. Clarify the basis for identifying these SAMA candidates.

g. Several SAMA candidates identified in Table E.6-1 are subsumed in another SAMA candidate (e.g., AC/DC-06, AC/DC-09, AC/DC-20). For each subsumed SAMA candidate, provide an assessment of its implementation cost relative to that of the SAMA into which it is subsumed. If the implementation cost of the subsumed SAMA is less, provide a revised basis for the Phase I screening and a Phase II cost-benefit evaluation if it meets Criterion F.

h. A few SAMA candidates identified in Table E.6-1 are screened for Very Low Benefit based on low contribution to LERF (e.g., CB-02, CP-21, OT-07). The ER does not provide sufficient information to assess the contribution of LERF to population dose-risk and offsite economic cost-risk relative to the total contribution from all release categories. Considering that the benefit of a SAMA is potentially based on the contribution from multiple release categories, provide additional justification for screening these SAMAs on Very Low Benefit.

i. SAMA CB-18, "direct steam generator flooding after a steam generator tube rupture (SGTR), prior to core damage," was screened in Table E.6-1 because it could impact efforts to mitigate the SGTR. This SAMA was determined to be potentially cost-beneficial in previous SAMA analyses (e.g., Diablo Canyon, TMI-1). Provide a cost-benefit evaluation of this SAMA.

6. Provide the following with regard to the Phase II cost-benefit evaluations:

   a. ER Section E.7.2 states that an expert panel developed the implementation cost estimates for each of the SAMAs. Briefly, describe the level of detail used to develop the cost estimates (i.e., the general cost categories considered). Also, clarify whether the cost estimates accounted for inflation, contingency costs associated with unforeseen implementation obstacles, replacement power during extended outages required to implement the modifications, and maintenance and surveillance costs during plant operation.

   b. SAMA CC-19, "provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs," has an estimated implementation cost of $1.5M. Table E.6-1 states that Davis-Besse already has this capability but that the feature has been deactivated, and that the cost would be minor to reactivate this feature. The estimated cost of $1.5M seems very high based on this description. Furthermore, other SAMA analyses have estimated the cost of this SAMA to range from $265K (Robinson) to $1M (Catawba). Provide a more detailed description of this modification and justification for the estimated cost.

   c. SAMA AC/DC-25, "provide a dedicated DC power system (battery/battery charger) for the TDAFW control valve and NNI-X for steam generator level indication," has an estimated implementation cost of $2M. This cost seems quite high for a system dedicated to just the TDAFW control valves and in light of the estimated costs for AC/DC-01 and AC/DC-03. Provide a more detailed description of this modification and justification for the estimated cost. Also, consider whether a portable system can provide the same benefit at a lower cost.

d. SAMA CW-24, "replace the standby CCW pump with a pump diverse from the other two CCW pumps," has an estimated implementation cost of $7.5M. This cost seems quite high for a pump replacement. Provide a more detailed description of this modification and justification for the estimated cost.

e. As reported in Table E.7-2, the population dose risk reduction is either 10.00% (for 3 SAMAs) or 0.00% (for all other SAMAs). Explain how population dose risk was calculated and justify the result for each SAMA individually.

f. The model approach for SAMA AC/DC-01, "provide additional DC battery capacity," assumes a seven hour battery life. Provide the battery life assumed in the base PRA model, the basis for assuming a seven hour battery life in the SAMA analysis, and justification for the estimated implementation cost of $1.75M. .

g. The model approach for SAMA AC/DC-14, "install a gas turbine generator," assumes failure of the station blackout (SBO) diesel generator is eliminated. This assumption does not provide credit for the gas turbine generator in the situation where all the emergency diesel generators (EDGs) are unavailable. Provide an assessment of the impact of this omission.

h. The model approach for SAMA CB-21, "install pressure measurements between the two DHR suction valves in the line from the RCS hot leg," assumes latent failures of the upstream valve are eliminated. It is unclear what is meant by "latent failures." Provide a more detailed description of the PRA model changes made to evaluate this SAMA.

i. ER Section E.8.6 discusses six sensitivity cases. Relative to these sensitivity cases, provide the following:

    i. Insufficient information is provided to understand the specific changes made to the baseline analysis assumptions for the first and fourth sensitivity cases. Provide a more detailed description of the analysis assumptions and methodology for these two cases.

    ii. The description of the sixth sensitivity case states that off-site economic cost was increased by 25 percent. Table E.8-1 indicates that the total benefit for each of the SAMA candidates was increased by the same amount of $19,632, the offsite economic cost (AOC) value. Clarify how the increase of 25 percent in off-site economic cost correlates to the increase in total benefits of $19,632 for each SAMA.

j. ER Section 8.3 discusses a sensitivity case using a higher evacuation speed. Provide the evacuation speed used for this analysis. Also, Table E.3-31 shows that the population dose decreased compared to the base case yet Table E.8-1 shows the total net benefit increased by $1,963 for each SAMA. Explain this anomalous result and describe the methodology for developing the $1,963 used for each SAMA.

k. The ER provides no assessment of the uncertainty distribution for CDF. Relative to the uncertainty distribution, address the following:

- Provide the uncertainty distribution (5th, mean, and 95th percentiles) for the Davis-Besse PRA model CDF and describe how the distribution was developed.

- Provide an assessment of whether an uncertainty analysis using the 95th percentile CDF and the external events multiplier of 3.6 developed in RAI 3.c is bounded by the Multiplier Case sensitivity analysis. If not bounded, provide an uncertainty analysis using the 95th percentile CDF. In this analysis, provide an assessment of each Phase 1 SAMA eliminated using Screening Criterion D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.

- If the Multiplier Case is bounding, provide an assessment of each Phase 1 SAMA eliminated using Screening Criterion D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.

7. For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other Babcock and Wilcox plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at Davis-Besse Nuclear Power Station.

   a. Automate reactor coolant pump trip on high motor bearing cooling temperature.

   b. Use the decay heat removal (DHR) system as an alternate suction source for high pressure injection (HPI).

   c. Automate HPI injection on low pressurizer level (in loss of secondary side heat removal cases where the reactor coolant system (RCS) pressure remains high while the RCS level drops) – Three Mile Island SAMA 16.

   d. Automate refill of the borated water storage tank (BWST).

   e. Automate start of auxiliary feedwater (AFW) pump in the event the automated emergency feedwater (EFW) system is unavailable.

   f. Purchase or manufacture of a “gagging device” that could be used to close a stuck-open steam generator safety valve for a SGTR event prior to core damage.

April 20, 2011

Mr. Barry S. Allen
Site Vice President,
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Oak Harbor, OH 43449-9760

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1, LICENSE RENEWAL APPLICATION

Dear Mr. Allen:

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Sincerely,
/RA/

Paula E. Cooper, Project Manager
License Renewal Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
Requests for Additional Information

cc w/encl: Listserv

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DATE	4/6/11	4/14/11	4/20/11	4/20/11

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Letter to Barry S. Allen from Paula E. Cooper dated April 20, 2011

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1, LICENSE RENEWAL APPLICATION

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