FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy.
Point Beach Nuclear Plant

December 12, 2007

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10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Response to Request for Additional Information
License Amendment Request 256
One-Time Extension of Containment Integrated Leakage Rate Test Interval

Reference: (1) FPL Energy Point Beach, LLC to NRC Letter Dated October 12, 2007, License Amendment Request 256, One-Time Extension of Containment Integrated Leakage Rate Test Interval, (ML072910053)
(2) NRC to FPL Energy Point Beach, LLC Letter Dated November 15, 2007, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information Related to the Containment Integrated Leak Rate Testing (ILRT) Interval Extension for Point Beach, Units 1 and 2 (MD7013 and MD4014), (ML073170039)

Via Reference (1) above FPL Energy Point Beach, LLC (FPLE-PB) submitted a proposed license amendment request for Point Beach Nuclear Plant (PBNP), Units 1 and 2, for Commission review and approval pursuant to 10 CFR 50.90.

On November 13, 2007, a telephone conference was held between NRC and FPL Energy personnel. During the conference, License Amendment Request 256 was discussed and additional information was requested. It was agreed that the response to the request for additional information would be submitted by December 14, 2007. The request for additional information was received November 15, 2007, via Reference (2).

Enclosure 1 of this letter provides FPLE-PB's responses to the questions in Reference (2). Enclosure 2 resubmits Section 3.1 of Reference (1), which was revised to incorporate the data from the revised risk assessment in Enclosure 3. Enclosure 3 is the updated risk assessment and replaces Enclosure 3 in Reference (1). Enclosure 4 provides a White Paper, dated December 6, 2007, "Impact on the Point Beach Nuclear Plant Large Early Release Frequency (LERF) Due to Level 2 Modeling Enhancements."

This response to a request for additional information does not alter the environmental considerations and the no significant hazards consideration contained in Reference (1).

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NRR

This submittal has been reviewed by the Plant Operations Review Committee. The submittal contains no new commitments or revisions to existing commitments.

In accordance with 10 CFR 50.91, a copy of this response to a request for additional information is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 12, 2007

Very truly yours,

FPL Energy Point Beach, LLC

James H. McCarthy
Site Vice President

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
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ENCLOSURE 1

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 256
ONE-TIME EXTENSION OF CONTAINMENT
INTEGRATED LEAKAGE RATE TEST INTERVAL

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The following information is provided by FPL Energy Point Beach, LLC (FPLE-PB) in response to the NRC staff's request for additional information resulting from a letter dated November 15, 2007.

Question 1

The requested ILRT extension is estimated to result in an increase in large early release frequency (LERF) (for internal events) of approximately 3E-7 per year. In accordance with Regulatory Guide 1.174, such applications will be considered only if it can be reasonably shown that the total LERF is less than 1E-5 per year. As indicated in the license amendment request (LAR), the combined internal and external events LERF for the baseline PRA is approximately 2E-5 per year and exceeds the risk acceptance guideline. A further assessment of induced steam generator tube rupture (ISGTR) events in Attachment C-1 of the LAR indicates that the total LERF could be reduced below the acceptance guideline if the conditional probability of an ISGTR of 0.25 used in the baseline PRA (obtained from NUREG-1570 and applied generically to all sequences with high RCS pressure and dry steam generators to arrive at the frequency of an ISGTR), is replaced by a plant-specific assessment of the frequencies of the various sequences leading to high/dry conditions at Point Beach (particularly high/dry sequences with concurrent reactor coolant pump seal LOCA) and their respective conditional failure probabilities. However, this analysis continues to rely on conditional failure probabilities and steam generator flaw distributions from NUREG-1570, which is nearly 10 years old, and fails to reflect additional information developed subsequent to NUREG-1570. More recent NRC-sponsored thermal-hydraulic studies to evaluate steam generator tube integrity suggest that the conditional probability on an ISGTR could be higher than considered in NUREG-1570, i.e., ISL draft reports "SCDAP/RELAP Base Case Calculation for the Station Blackout Uncertainty Study," August 2006 (ML070220062), and "Evaluation of Uncertainties in SCDAP/RELAP5 Station Blackout Simulations," August 2006 (ML070220056). Discuss the applicability and implications of the aforementioned studies on the Point Beach LERF assessment. Provide additional assessments, as appropriate, to illustrate the sensitivity of the total LERF estimate to higher conditional probabilities of ISGTR as might be inferred from the more recent studies.

FPLE-PB Response

The analysis documented in the application dated October 12, 2007, Enclosure 3, Appendix C has been revised to incorporate industry peer review comments, and includes components of the update to the Point Beach Nuclear Plant (PBNP) Level 2 probabilistic risk assessment (PRA). The revised analysis includes a plant-specific Level 1/Level 2 mapping and the current representation of the steam generator evaluation. While still conservative, this revision includes several plant-specific factors not considered in the original submittal. Based on a comprehensive review of the top 25 sequences that were binned into Source Term Category (STC) 8, the plant-specific differences between the reference plant (Surry) used in NUREG-1570 and PBNP have been identified and incorporated. STC 8 is early steam

generator tube ruptures (SGTRs), including induced steam generator tube ruptures (ISGTRs), with the top 25 sequences representing about 90% of the STC 8 frequency. Including these plant-specific features results in internal events LERFs of 2.11E-6/Yr and 2.19E-6/Yr for Unit 1 and Unit 2, respectively. These LERF reductions are attributable to the reduction in ISGTR. The ISGTR reductions are due primarily to four factors: 1) correction of an error in a decomposition event tree that provides sorting for the containment event tree. The error caused sequences with successful auxiliary feedwater or successful main feedwater to be classified as "dry", and thus subject to a possible ISGTR, 2) the modification of the accident progression event tree (APET) fault tree to use two steam generators (SG) versus three SGs, 3) the development of a plant-specific reactor coolant pump (RCP) seal loss-of-coolant-accident (LOCA) sequence, and 4) accounting for the actual condition of the PBNP SGs. The LERFs associated with the external events also are reduced to 1.53E-6/Yr for both Unit 1 and Unit 2. The combined internal/external events contribution to LERF is 3.64E-6/Yr and 3.72E-6/Yr for Unit 1 and Unit 2, respectively.

PBNP has reviewed the more recent NRC-sponsored thermal-hydraulic studies (mentioned in this question) to evaluate SG tube integrity. The discussion provided in the Information Systems Laboratories (ISL) draft report was considered. The analyses presented in this report were performed to evaluate plant behavior during hypothetical event sequences with a potential for leading to a severe accident. The occurrence of the event sequences is unlikely due to multiple assumed concurrent failures of systems and components. A few of the key assumptions for the station blackout base case accident sequence are:

- Loss of off-site power for an extended period;
- Failure of all diesel-electric generators to start;
- Failure of the turbine-driven auxiliary feedwater system to operate;
- 21 gpm (equivalent hole size) reactor coolant pump shaft seal leakage;
- Steam leakage causes both SGs to depressurize.

These assumptions result in a "high-dry" condition with one or both SGs depressurized by the time any primary system ruptures are predicted to occur. No operator intervention for mitigating the accident is accounted for. The analyses therefore, do not represent best-estimate plant behavior, nor do the results indicate the most likely outcomes of the event sequences. The results can only be placed into perspective with appropriate consideration for the probability of such events occurring. The predicted results apply only for the specific analysis assumptions and may vary considerably as assumptions are changed (for example, greater reactor coolant pump shaft seal leakage rates can eliminate SG tube failures which are predicted at smaller leakage rates). These considerations were accounted for in a revised LERF evaluation including the severe accident ISGTR failures.

Industry experts on severe accident ISGTR were also consulted to provide an assessment of the applicability of the ongoing analyses performed by the NRC and consultants as well as analyses performed by EPRI and documented in TR-107623, "Steam Generator Tube Integrity Risk Assessment Methodology." FPLE-PB continues to follow these ongoing activities. While the various analytical approaches have been discussed, the site PRA and risk analyses based their evaluations on the EPRI analyses because of the extensive benchmarks with the EPRI/NRC sponsored 1/7 scale SG tests as well as the TMI-2 benchmark that shows a very conservative representation of the upper plenum temperature; the driving potential for natural circulation to the inverted U-tube SGs. Both of these indicate that the current industry (EPRI) evaluations are conservative.

There is still considerable conservatism in the LERF model, as outlined in the response to Question 4. FPLE-PB believes that the conservatism in the LERF model and the margin provided in the revised analysis adequately addresses the uncertainties associated with the conditional probabilities of ISGTR as might be inferred from the referenced recent studies.

Question 2

Provide a description of the reactor coolant pump seal LOCA model used in the Point Beach PRA. The NRC staff has approved the use of the "WOG 2000 RCP Seal Leakage Model for Westinghouse PWRs" described in WCAP-15603, Revision 1 (ML032040132). If a different model has been used in the Point Beach PRA, provide an assessment of the impact on core damage frequency and total LERF of using the approved seal leakage model.

FPLE-PB Response

The PBNP PRA Model uses RCP seal LOCA probability results from "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," WCAP-15603, Rev 1-A, dated June 2003 (ML031400376). This report includes the NRC Safety Evaluation Report. The only split fraction for the RCP seal LOCA used in the PBNP model is the probability of nominal leakage of 21 gpm per pump (0.79) and the complementary probability of above-nominal leakage (0.21) upon loss of RCP seal cooling. A sequence that results in above-nominal leakage due to a loss of RCP seal cooling requires reactor coolant system injection (high head safety injection or residual heat removal) for success. Success for depressurization and prevention of core damage was demonstrated using the MAAP4 thermal hydraulics computer code and a PBNP-specific model. This WCAP was used in the initial application dated October 12, 2007.

Question 3

Confirm that the treatment of the turbine-driven auxiliary feedwater pump in the PRA on which the LAR is based is consistent with plant-specific operating experience to-date with this system. Discuss the recent operating experience and how it has been reflected in the PRA.

FPLE-PB Response

During 2007, both PBNP units experienced problems with their associated turbine-driven auxiliary feedwater pumps (TDAFP). On Unit 1, the problem was high turbine bearing temperatures experienced following a turbine overhaul which was performed during the spring 2007 refueling outage. On Unit 2, a problem with water intrusion into the bearing oil was exacerbated during bearing maintenance conducted in September. Both of these problems resulted in additional unavailability of the pumps for troubleshooting and repairs.

The TDAFP failure probabilities used in the PRA model that served as the basis for the ILRT analyses (PRA Model Rev 3.17) were compared to those from a pending PRA model update (Rev 4.00), mitigating systems performance indicator (MSPI) derived test and

maintenance (TM) unavailability from 36 data months ending in October 2007, and to the probabilities from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, January 2007. A discussion of this data comparison follows:

- The TDAFP failure-to-run (FR) probability used in the ILRT study is more than ten times higher than the more recent analysis and the NUREG generic value. This is due to a run failure experienced at the plant during the Rev 3.17 data analysis time period.
- The TDAFP failure-to-start (FS) probability used in the ILRT study is about three times lower than the more recent analysis and about 2.5 times lower than the NUREG value. PBNP formerly ran the TDAFP for plant startups and shutdowns. Although there was one start failure at the plant during the Rev 3.17 data time period, there were also more starts in the data period, so the impact after Bayesian analysis was lower than the generic mean.
- The TDAFP TM probability used in the ILRT study is higher than both the more recent plant analysis and the NUREG generic value

To ensure that the ILRT risk impact assessment used data that bounds current AFW performance, a sensitivity analysis was performed on the Level 1 and Level 2 PRA analyses for Units 1 and 2 with more recent failure probabilities for the TDAFPs. The updated FS and FR data was obtained from the draft Rev 4.00 of the PBNP PRA model that used plant operating experience from 2000 through 2005. The TM unavailability was derived from MSPI information through October 2007. This is the most recent performance data.

Basic Event	Rev 3.17 Value	Sensitivity Value	NUREG/CR-6928
AF--TDP-FR--1P29	3.21E-02	2.47E-03	1.76E-03 *
AF--TDP-FR--2P29	3.21E-02	2.47E-03	1.76E-03 *
AF--TDP-FS--1P29	4.31E-03	1.36E-02	9.52E-03 **
AF--TDP-FS--2P29	4.31E-03	1.36E-02	9.52E-03 **
AF--TDP-TM--1P29	7.76E-03	9.13E-03	5.44E-03
AF--TDP-TM--2P29	7.76E-03	9.13E-03	5.44E-03

* NUREG per-hour value multiplied by 24 hours
** NUREG values for FTS and FTR <1hr added together

The TDAFP common cause failure probabilities were not modified for the sake of simplicity. This is conservative since the FR common cause value should decrease significantly (about a factor of 10) while the FS common cause value should increase much less (about a factor of three).

The PBNP Unit 1 internal events LERF resulting from this AFW sensitivity study is 1.98E-6/yr. This represents a reduction of about 6.1% in the base Unit 1 internal events LERF of 2.11E-6/yr.

The PBNP Unit 2 internal events LERF resulting from this AFW sensitivity study is 2.08E-6/yr. This represents a reduction of about 4.9% in the base Unit 2 internal events LERF of 2.19E-6/yr.

As is demonstrated by the sensitivity study results above, when current performance of the TDAFP is taken into account, the results are bounded by the results from the ILRT risk impact assessment. This includes the impact of additional unavailability of the TDAFP experienced over the past six months. This sensitivity study was performed to address this question. These results were not incorporated into the revised risk assessment in Enclosure 3.

Question 4

Provide a breakdown of the key contributors to the total LERF estimate for internal and external events. Discuss the degree of realism, conservatism, or non-conservatism associated with each key contributor and the rationale for this characterization, including the underlying models and assumptions that may contribute to this view. Provide an estimate of the extent to which the total LERF would be impacted if other areas of the analysis (besides ISGTR) were treated more realistically.

FPLE-PB Response

Tables 2 and 3 of Enclosure 4 summarize the key contributors to the total LERF estimate for PBNP Unit 1 and Unit 2, respectively. These key contributors are early liner failures, containment isolation failures, interfacing systems loss-of-coolant-accidents (ISLOCA), initiating event SGTRs and main steam and feed line break-initiated SGTRs with early release and ISGTRs. Plant-specific features were included to reflect SG configurations. Considerable conservatisms still exist in the Level 1 PRA model and these are summarized below. These conservatisms indirectly affect the Level 2 model as well.

1. Internal events

   Component Failure data and CCF data

   The model used for the ILRT interval extension was based on a set of older data. The plant-specific failure rates were taken from plant operating history from September 1990 through December 1999. The test and maintenance unavailability data was taken from plant operations from January 1997 through December 1999. The common cause factors are taken from NUREG/CR-5497. A data update was performed after the ILRT interval extension project was initiated. In this data update, the plant-specific failure rates were taken from plant operating history from January 2000 through December 2005. The test and maintenance unavailability data was taken from plant operations from January 2003 through December 2005. Common cause factors were derived from a NRC database located at the following Internet address: http://nrcoe.inel.gov/results/CCF/ParamEst2003/ccfparamest.htm under "Common Cause Failures (CCFDB)." A comparison of the new and the previously used data indicates that the core damage frequency (CDF) for the high dry sequences involving AFW is conservative based on the previous data.

   Credit for operator actions

   After review of the dominant cutsets, additional credit for operator actions may be included to mitigate some accident scenarios. In addition, some human error probability

may be reduced if more realistic scenario-specific timing is used rather than the bounding values used in the model supporting the ILRT interval extension.

2. External Events:

   Similar to that identified above for internal events, the same factors rendering the higher LERF estimates for internal events also yield higher LERF estimates for external events. The core damage frequency of external events is based on the individual plant examination for external events (IPEEE) and may be conservative due to the scoping nature of the estimate.

Although the conservatisms identified above reduce the CDF directly, they will reduce the LERF as well. A detailed LERF quantification considering the data update and credit for operator actions has not been performed. A scoping estimate using the current data contained in draft Revision 4.0 indicates that the baseline CDF using the more current data yields a CDF reduction of approximately 15% to 20%. Credit for proceduralized operator actions is expected to reduce applicable sequences even more.

Question 5

Describe the internal and external industry/peer reviews of the Point Beach Level 2/LERF model (or its predecessor), the findings from these reviews, and the actions taken to address the peer review findings. Specifically address any comments that relate to the conservatism or non-conservatism of key models or assumptions that impact the total LERF, and how these comments have been addressed within the PRA version used to support the LAR.

FPLE-PB Response

The PBNP PRA model, including the Level 2 portion, was reviewed by a team of six PRA engineers in June 2001 as a part of the Westinghouse Owners Group peer review effort. The model version reviewed by this team was a draft of Revision 3.00. The final report of this review was received in November of 2002. In this report, PBNP received a total of three Significance Level A facts and observations (F&Os) and 30 Significance Level B F&Os. In the Containment Performance (Level 2) element, the PBNP model was assigned no Significance Level A F&Os, two Significance Level B F&Os and three Significance Level C F&Os.

The first Level B F&O in the Containment Performance element dealt with a lack of evidence that equipment in containment, are qualified to operate in a post-core damage environment. The fan coolers and power-operated-relief-valves were of specific concern. The F&O describes this as a concern for long-term containment heat removal and possible eventual containment failure on overpressure and states that LERF is not affected by these considerations. This F&O remains open.

The second Level B F&O noted that the LERF model at the time of the review contained one conservatism and one non-conservatism. The conservative aspect of that LERF model was that all SGTR-initiated core damage sequences were considered LERF regardless of whether the core damage occurred early or late. The non-conservative aspect of that LERF model was that pre-core damage ISGTRs from a faulted SG (steam line break or a feed line break inside containment) were not included as LERF. Both of these corrections have been made in the current LERF model. In addition, the current LERF model includes the impact of pressure-induced and thermal-induced SGTR as is discussed in the response to Question 1 above and in the attached description of the LERF model.

Of the three Significance Level C F&Os in the Containment Performance element, two dealt with the lack of containment failure modes from core melt energetic releases. The first observed that an engineering justification was not provided to demonstrate that failure would not occur. The second F&O observed that containment temperature was apparently not considered along with pressure for the containment strength analysis. The third Significance Level C F&O observed that the Level 2 analysis did not include operator actions to depressurize the reactor coolant system (RCS) after the core uncovers to mitigate the effects on containment of a high pressure melt ejection. All three of these observations were assigned a Level C significance because they have no impact on LERF. The potential for steam explosions has been included in the 2007 Level 2 update. The other observations remain open.

In December 2006 and May 2007 a gap assessment of the PBNP PRA model was performed by a team of two contractors and two offsite PRA engineers against the requirements of RG 1.200 and ASME Standard RA-Sb-2005. This gap assessment included a Level 1 review, but did not include the LERF portion of the Level 2 PRA because the 2007 Level 2 update was still in progress. Observations from this gap assessment that dealt with the remainder of the PBNP PRA model will be addressed during the on-going RG 1.200 upgrade project.

Finally, the methodology used for the most recent update of the PBNP Level 2 / LERF model was reviewed by two industry experts in containment performance phenomenology. While this was not a comprehensive peer review, their comments have been addressed in this current model used for the revised ILRT risk impact assessment.

Question 6

In the discussion regarding very small increases in LERF, Regulatory Guide 1.174 (Section 2.2.4) notes that if there is indication that total LERF may be considerably higher than 1E-5 per year, the focus should be on finding ways to decrease LERF. Discuss any activities planned or in progress at Point Beach to reduce the total LERF for the plant.

FPLE-PB Response

The LERF of 2E-5/yr was calculated using non-plant specific information. The recalculated LERF using plant specific data is approximately 4.0E-6/yr, significantly below 1E-5/yr. During the evaluation of Regulatory Guide 1.200 requirements, it is anticipated that the conservatism described in the response to Question 4 will be re-evaluated.

ENCLOSURE 2

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 256
ONE-TIME EXTENSION OF CONTAINMENT
NTEGRATED LEAKAGE RATE TEST INTERVAL

REVISED SECTION 3.1 RISK ASSESSMENT

2 pages follow

3.1 Risk Assessment

3.1.1 Methodology

An evaluation was performed to assess the risk impact of a one-time extension of the currently allowed containment Type A ILRT frequency from 10 years to 15-1/2 years.

The risk assessment follows the guidelines from NEI 94-01 and the NRC regulatory guidance, as outlined in RG 1.174, on the use of PRA findings and risk insights in support of the request to change PBNP's licensing basis. This methodology is similar to that presented in EPRI TR-104285 and NUREG-1493.

The potential impact of age-related corrosion of the steel containment vessel on the risk associated with extending the ILRT interval has also been determined. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP), and subsequently used in other submittals including those for Comanche Peak and D. C. Cook. The details of this assessment are contained in Enclosure 3

3.1.2 Input Information

The risk assessment utilizes input of population doses for containment failure modes provided in the Environmental Report Operating License Renewal Stage, Point Beach Nuclear Plant, Units 1 and 2, dated February 2004. The updated PRA total Core Damage Frequency (CDF) and frequency of various release categories are based upon the calculations done for the Environmental Report. Data from NUREG-1493 and the EPRI Interim Guidance were used to calculate the probabilities of a liner leak size.

Point Beach PRA Model Revision 3.17 was used for the risk impact assessment. A draft version of PRA Model Revision 3 underwent a peer review in June of 2001. The peer review team concluded that the model was of sufficient quality to support risk-informed applications supported by deterministic analyses, provided the Level A and Level B observations were addressed. Level A observations and Level B observations pertinent to Large Early Release Frequency (LERF) analyses have been resolved.

3.1.3 Results

The combined internal and external events increase in risk for those accident sequences influenced by Type A testing, compared with the total integrated plant risk, given the change from a one-in-10 year test interval to a one-in-15-1/2 year interval, was found to be 4.4 percent (0.2 person-rem/yr) for Unit 1 and 2.1 percent (0.1 person-rem/yr) for Unit 2. Given the low total risk to the public, these values are not significant increases in risk.

The realistic combined internal/external events contribution to LERF at PBNP is 3.64E-06 for Unit 1 and 3.72E-06 for Unit 2 as described in the analysis provided in Enclosure 3. The change in the realistic combined internal events/external events LERF associated with increasing the ILRT interval at PBNP is 3.08E-07 for Unit 1 and 2.62E-07 for Unit 2. Because RG 1.174 defines small changes in LERF as below 1E-6/yr, an increase in the ILRT interval at PBNP represents a small change in plant risk

from the realistic LERF perspective. Similarly, the change in realistic values of LERF of 7.00E-07 for Unit 1 and 5.97E-07/yr for Unit 2 falls into Region II, Small Change in Risk, of the acceptance guidelines in RG 1.174.

The change in conditional containment failure probability due to the proposed change in ILRT frequency is 0.36 percent for Unit 1 and 0.30 percent for Unit 2. This change is small compared to the total containment failure probability.

The impact of age-related corrosion of the steel containment is very small on each of the risk measures associated with the extension of the Type A ILRT test frequency. This conclusion remains valid even including consideration of corrosion.