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

FPL Energy

Point Beach Nuclear Plant

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10 CFR 50.59(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Annual 10 CFR 50.59 Summary Report for 2007

FPL Energy Point Beach, LLC, is submitting this annual 10 CFR 50.59 Summary Report for the Point Beach Nuclear Plant (PBNP), Units 1 and 2.

This report consists of two enclosures. Enclosure 1 contains descriptions of facility changes, tests and experiments evaluated in accordance with 10 CFR 50.59 during 2007. Enclosure 2 contains commitment change evaluations completed in 2007.

This letter contains no new commitments and no revisions to existing commitments.

Very truly yours,

FPL Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures (2)

cc: Incident Response Center, Region III (copy of PBNP EP Manual on CD-ROM)
Resident Inspector, Point Beach Nuclear Plant, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC

ENCLOSURE 1

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

ANNUAL 10 CFR 50.59 SUMMARY REPORTS FOR 2007
MODIFICATIONS FSAR CHANGES AND OTHER EVALUATIONS

FSAR Change, Revision of FSAR Section 14.2.5, "Rupture of a Steam Pipe."

Activity Description: Calculation CN-TA-96-075 reduced the main feedwater temperature assumed in analyses performed for zero power conditions from 100°F to 35°F. By extension, revising the inputs of the analyses is a change to the analyzed operation of the facility. Therefore, this proposed activity includes physically lowering the minimum feedwater temperature at zero power conditions.

Summary of 10 CFR 50.59 Evaluation: FSAR Section 14.2.5, "Rupture of a Steam Pipe," does not directly mention or discuss the feedwater temperature; there is an indirect connection that ties the parameter to the analysis. The "Method of Analysis" states the analysis is performed to determine the core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. Although the FSAR does not imply that core heat flux is affected by feedwater temperature during this event, the underlying analysis uses the parameter for an input.

The analysis has been re-performed using the reduced feedwater temperature as an input, and the impact on core heat flux, RCS temperature, and pressure is acceptable. The change results in a less than minimal increase in peak heat flux, and the heat flux remains within the DNBR and kW/ft design limits. (EVAL 2007-001)

Maintenance Work Activity, Defeat Pressurizer Low Level Letdown and Heater Cutoff to Replace Bistables Online.

Activity Description: The pressurizer level low bistables were replaced online while preventing CVCS letdown isolation from occurring and maintaining pressurizer backup heaters in service. CVCS letdown isolation is defeated by opening a slider in 2C-004, "Reactor and CVCS Control Board," that inhibits the letdown isolation control signal from being generated by either pressurizer level low bistable. The backup heater cutout function is inhibited by physically holding the heater cutout relay solenoid in the normally energized position, which prevents the cutout relay contacts from changing state. The heater cutout relays are inhibited one at a time and coordinated with replacement of its associated pressurizer level low bistable. During this evolution, a Dedicated Operator is stationed to monitor pressurizer level and manually initiate letdown isolation and deenergize pressurizer heaters should indicated level drop below 20%.

Summary of 10 CFR 50.59 Evaluation: This activity affects the following functions:
1) Pressurizer level low backup heater banks A, B, C, D, E cutoff (this function is not, nor does it support any Design Basis Functions). The pressurizer backup heaters are deenergized to prevent uncovering and damaging heaters on a lowering pressurizer level at 12% indication. The backup heater automatic cutoff function is provided by 2LC-00427B-X1 and

2LC-00428D-XI relays that are driven by the 2LC-O0427B/D and 2LC-00428D bistables, respectively.

2) Pressurizer level low CVCS letdown isolation (this design function remotely operate the valve located near the main coolant loop on low pressurizer level to prevent supplementary loss of coolant through a letdown line rupture per FSAR 9.3-7, however, there is no limiting safety system setting associated with letdown isolation.): The pressurizer level low CVCS letdown isolation cuts off letdown flow to maintain reactor coolant inventory in the event of a lowering pressurizer level at 12% indication. The CVCS letdown isolation function is provided by 2LC-00427B-X2 and 2LC-00428D-X2 relays that are driven by the 2LC-O0427B/D and 2LC-00428D bistables, respectively.

3) Pressurizer level high and level low (HI-LO) alarm (this function is not, nor does it support any Design Basis Functions): The pressurizer HI-LO alarm function provides the operator with indication of an abnormally low pressurizer water level at 12% and indication of an abnormally high pressurizer water level at 70%. The pressurizer HI-LO alarm is activated on a low level condition provided by 2LC-00427B-X2 and 2LC-00428D-X2 relays that are driven by the 2LC-O0427B/D and 2LC-00428D bistables respectively. The pressurizer HI-LO alarm is also activated on a high level condition provided by 2LC-00427D-X relay driven by the 2LC-O0427B/D bistable.

During a normal lowering pressurizer level condition, the automatic function would trip at 12%. The pressurizer low level letdown and heater cutoff bistable causes the RCS Loop B Cold Leg to CVCS motor operated valve (MOV) to shut to isolate letdown flow, and opens a contact in the pressurizer heater controller to deenergize the pressurizer heaters. The manual action would perform the letdown isolation by shutting the MOV and deenergizing the pressurizer heaters from the control room during a lowering pressurizer level condition upon to reaching 20%. The dedicated operator monitors pressurizer level instruments L-427 and L-428 in the control room. These indications are adequate based on the fact that they are the normally relied upon indications that operators use to monitor pressurizer level and, as stated in FSAR 7.2.3.2.d, "Level alarms are available and ample time exists for operator action." The manning requirement of one operator was based on the need to only monitor one parameter and the controls that would need to be manipulated are present at the dedicated operator station.

The automatic pressurizer low level letdown and heater cutoff actuation are not safety related functions, are not relied upon in the safety analysis, and are not initiators of an accident, thus their failure would not increase the probability or consequence of an accident or a malfunction of equipment important to safety. The requirement to be capable of closure subsequent to a line-break (LOCA) will not be inhibited. The auto close feature to isolate letdown on low pressurizer level prevents uncovering the pressurizer heaters and is for component protection and is considered non-safety related (RC-427 IST Valve Data Sheet). Since the current accident analysis does not rely upon the pressurizer low level letdown and heater cutoff, the current analysis is bounding. (EVAL 2007-003)

ENCLOSURE 2

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

ANNUAL 10 CFR 50.59 SUMMARY REPORTS FOR 2007
COMMITMENT CHANGE EVALUATIONS

Steam Generator Secondary-Side Inspection Interval Change: The original commitment stated that as part of the Steam Generator Integrity Program, visual inspections of accessible areas to verify the integrity of steam generator secondary side components will be performed at least every six years, with one steam generator being inspected every three years on an alternating basis. Any indications of degradation or unacceptable conditions will be evaluated through the corrective action program, including the extent of condition.

Justification for Change: NRC Safety Evaluation dated August 22, 2006, allowed an increase in the time between Unit 2 steam generator primary side inspections. Although there is no requirement to inspect primary and secondary side of the steam generators in sync, tube degradation sizing and tube repairs (plugging) can only be performed from the primary side. Therefore, secondary side inspections are indirectly tied to primary side inspections for potential repair reasons.

Visual inspections of accessible areas to verify the integrity of steam generator secondary side components will be performed at least every six effective full power years (EFPY). Indications of degradation or unacceptable conditions will be evaluated within the corrective action program, including the extent of condition. (CCE 2007-002)

Formal Audit Checklists: The original commitment required an audit plan or checklist for Quality Assurance review activities.

Justification for Change: The Quality Assurance Topical Report implements ASME NQA-I-1994. Section 3.1, Audit Plan, of NQA-1 requires that the auditing organization develop and document an audit plan that identifies the written procedures or checklists.

This commitment is being cancelled based on it being incorporated into the Quality Assurance Topical Report (QATR) and its implementing documents. (CCE 2007-004)

Response Time for Audit Findings: The original commitment required a 30-day response to in-plant audit findings, as well as providing for notifying the Plant Manager and the Vice President should audit findings not be responded to within 30 and 60 days, respectively. Commitment Change 99-013 revised the commitment to state that nonconformances identified during audits, surveillances or work monitoring activities are documented in condition reports. The condition reporting system requires evaluations to be performed with urgency commensurate with safety significance. Quality Assurance (QA) Program deficiencies shall be documented in quality condition reports/condition reports and shall require a 30-day evaluation.

Justification of Change: Confirmatory Action Letter (CAL 3-04-001) Commitment, OR-02-001.07.C, "Nuclear Oversight (NOS) is effective in assuring management response to QA findings," requires that NOS implement a process for reporting NOS significant QA findings. This process is implemented by a procedure that requires a monthly report to management that provides the status of open QA findings.

This commitment has been superseded by CAL Commitment (CAL 3-04-001) OR-02-001.07.C. (CCE 2007-005)