Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

October 30, 1998
3F1098-17

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

Subject: Crystal River Unit 3 Response to Request for Additional Information – License Amendment Request #218 Related to Revised Analysis of Makeup System Letdown Line Failure Accident (TAC NO. M99571)

References: 1. NRC to FPC letter, 3N0998-05, dated September 18, 1998

2. NRC to Framatome Technologies, Inc. (FTI) letter dated February 18, 1997

Dear Sir:

In Reference 1, Florida Power Corporation (FPC) was requested to provide information demonstrating how each of the restrictions and conditions reflected in Reference 2 were met. Reference 2 is the NRC's acceptance for referencing Topical Report BAW-10192-P, "Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," which is based on RELAP-5/MOD2-B&W.

With support from Framatome Technologies, Inc., FPC prepared the attached response to the requested NRC information. Based on this assessment, the NRC's criteria for reliance on BAW-10192-P have been satisfied.

There are no new regulatory commitments established by this letter. Please contact Ms. Sherry Bernhoft, Manager, Nuclear Licensing at (352) 563-4566, if you have any questions regarding this letter.

Sincerely,

MW Rencheck

M.W. Rencheck
Director, Nuclear Engineering and Projects

MWR/twc

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Attachment

xc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

A001

# ASSESSMENT OF USING RELAP5 FOR LETDOWN LINE FAILURE ACCIDENT

## I. Background

In support of the spring 1996 refueling outage at the Crystal River Unit 3 (CR-3) plant, an analysis was performed of a break in the reactor coolant letdown line. The reactor coolant system (RCS) response for the letdown line failure accident was performed with the RELAP5/MOD2-B&W computer code (BAW-10164P). The computer code used in the analysis of record referenced in the CR-3 Final Safety Analysis Report (FSAR) was CRAFT2. The RELAP5 analysis was performed with the Framatome Technologies, Inc. (FTI) NRC-approved version of RELAP5/MOD2, and, with the exceptions described below, was done with the ECCS evaluation model (EM) as approved in BAW-10192-P.

The EM adequately determines the transient evolution; however, its primary purpose is to calculate a conservative peak cladding temperature (PCT) following a postulated LOCA. The EM models and methods are developed to predict core uncovering and resulting PCTs for transients with typical end times of less than two hours. The acceptance criterion for the letdown line failure accident, as reported in the CR-3 FSAR is the offsite dose released, and not core cooling. This break is limited in size and location, and can be isolated such that core cooling will always be assured due to isolation of the break and/or initiation of ESAS. Given that this analysis was specifically performed to address offsite dose (or release) consequences, two adjustments were made to the EM input deck. These changes were: (1) the break discharge volume initial pressure, and (2) the critical flow model. In addition, the INEL interphase drag model was used in the steam generator tubes instead of the B&W slug drag model. These adjustments are described in Section III, below.

## II. NRC Topical Report BAW-10192-P

The NRC documented its review of the subject topical report (dated February 1994), along with Framatome's May 6, 1996; October 11, 1996; and January 7, 1997 responses to NRC requests for additional information in a February 18, 1997 letter to Framatome.[1] On the basis of its review, the NRC concluded that BAW-10192 (hereinafter, the topical) is acceptable for referencing in licensing applications in the analysis of LOCA accidents for once-through steam generator plants. The NRC also noted that when the report is referenced in a license application (as exists in the CR-3 FSAR), it will not repeat its review of matters described in the report and found acceptable.

Enclosure 1 of the NRC's February 18, 1997, letter states, in part, that use of the topical methodology for reference in licensing applications involving large and small break LOCA analysis for B&W plants is acceptable, subject to eleven conditions. Accordingly, justifying use of the topical in licensee actions by comparing code use to the eleven conditions is necessary for plant-specific utilization of the topical. In that regard, this text addresses those eleven conditions and demonstrates that use of the topical in support of this license amendment is acceptable.

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[1] See letter from James E. Lyons, Acting Chief, Reactor Systems Branch, NRC, to J.H. Taylor, Manager, Licensing Services, Framatome Technologies Inc.; "Acceptance for Referencing of Topical Report BAW-10192-P, 'Loss-of-Coolant Accident Evaluation Model For Once-Through Steam Generator Plants' (TAC No. M89400)."

III. NRC Conditions for Licensee Use of BAW-10192P

FTI provides below, its evaluation of the eleven conditions that must be satisfied for a licensee to justify use of the Topical:

1. *The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model.*

   Response

   FPC (with FTI being the user of the code) has satisfied all NRC restrictions placed on the use of RELAP5 as defined in the evaluation model presented by Framatome to the NRC in its letters dated February 1994 and as supplemented in correspondence dated May 6, 1996 except for the input options noted in Item I and described in detail in Item III.2 below.

2. *The guidelines, code options, and prescribed input specified in Tables 9-1 and 9-2 in both Volume I and Volume II of BAW-10192P should be used in LBLOCA and SBLOCA evaluation model applications, respectively.*

   Response

   Given that this analysis was not specifically performed to address core cooling, two adjustments were made to the EM input deck. These changes were: (1) the break discharge volume initial pressure, and (2) the critical flow model. In addition, the INEL interphase drag model was used in the steam generator tubes instead of the B&W slug drag model.

   The letdown line failure accident is represented by a break in the letdown line downstream of the outboard isolation valve (inside the auxiliary building). The break discharge control volume pressure was set to 34 psia to simulate the presence of letdown line resistance before the break was opened. The transient output was reviewed and it was confirmed that the break flow remained choked throughout the transient. Therefore, no other changes to sink pressure were required.

   The EM critical flow model was initially used, but the break void fraction exceeded 70 percent. At this value, the static model may not be valid because it could under predict the break flow. Therefore, the critical flow model was changed and separate cases were run to compare the integrated break flow. The first cases calculated break flow based on stagnation properties. The next two cases investigated the Ransom-Trapp critical flow model and the EM model with a 0.7 multiplier on the transition and saturated conditions. The limiting case, the one that produced the greatest mass loss, was reported in the FSAR.

   The INEL interphase drag model was used on the inside of the steam generator tubes instead of the B&W slug drag model. The drag models calculate the interfacial effects between the two phases. However, in this analysis, the primary tube regions remained in single phase conditions, therefore there was no effect on the results.

3. *The limiting linear heat rate for LOCA limits is determined by the power level and the product of the axial and radial peaking factors. An appropriate axial peaking factor for use in determining LOCA limits is one that is representative of the fuel and core design and that may*

*occur over the core lifetime. The radial peaking factor is then set to obtain the limiting linear heat rate. For this demonstration, calculations were performed with axial peak of 1.7. The general approach is acceptable for demonstrating the LOCA limits methodology. However, as future fuel or core designs evolve, the basic approaches that were used to establish these conclusions may change. FTI must revalidate the acceptability of the evaluation model peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limits differ appreciably from those used to demonstrate Appendix K compliance.*

Response

The analyses performed did not relate to core kw/ft limits, so this restriction does not apply.

4. *The mechanistic ECCS bypass model is acceptable for cold leg transition (0.75 $ft^2$ to 2.0 $ft^2$) and hot let break calculations. The nonmechanistic ECCS bypass model must be used in the large cold leg break (> 2.0 $ft^2$) methodology since the demonstration calculations and sensitivities were run with this model.*

Response

The letdown line break is considered an isolatable SBLOCA with the limiting break location being outside the reactor building. For SBLOCA applications, the ECCS bypass model is typically not executed. ECCS bypass is a phenomenon applicable to blowdown during a large break of the reactor coolant piping in the reactor coolant pump leg. Therefore, this restriction does not apply to the letdown line break analysis.

5 *Time-in-life LOCA limits must be determined with, or shown to be bounded by a specific application of the NRC-approval evaluation model.*

Response

No LOCA limits were calculated; therefore, the restriction does not apply.

6. *LOCA limits for three pump operation must be established for each class of plants by application of the methodology described in this report. An acceptable approach is to demonstrate that three pump operation is bounded by four pump LHR limits.*

Response

No LOCA limits were calculated; therefore, the restriction does apply.

7. *The limiting ECCS configuration, including minimum versus maximum ECCS must be determined for each plant or class of plants using this methodology.*

Response

This restriction applies to analyses of large break LOCAs. As mentioned above, the letdown line break is considered an isolatable SBLOCA, and the use of minimum ECCS flow rates is *per se* the limiting case for SBLOCA analyses. However, in order to maximize releases and dose consequences, normal makeup is maximized, and the engineered safeguards pressure setpoint for high pressure injection actuation is minimized such that ECCS is not automatically actuated. Therefore, this restriction does not apply.

8. *For the small break model, the hot channel radial peaking factor to be used should correspond to that of the hottest rod in the core, and not to the radial peaking factor of the 12 hottest bundles.*

Response

The hot channel radial peaking factor used corresponded to the hottest rod in the core. Since this analysis is primarily for a radiological consequence, no core heatup calculations were performed, or required to be performed, and therefore, this restriction is not relevant.

9. *The constant discharge coefficient model (discharge coefficient = 1.0) referred to as the "High or Low Break Voiding Normalized Value," should be used for all small break analyses. The model which changes the discharge coefficient as a function of void fraction, i.e., the "Intermediate Break Voiding Normalized Value" should not be used unless the transient is analyzed with both discharge models and the intermediate void method produces the more conservative results.*

Response

The constant discharge coefficient model (discharge coefficient = 1.0) referred in the BAW-10192-P analysis as the "High or Low Break Voiding Normalized Value," was used for the makeup system letdown line failure accident analysis.

10. *For a specific application of the FTI small break LOCA methodology, the break size which yields the local maximum PCT must be identified. In light of the different possible behaviors of the local maximum, FTI should justify its choice of break sizes in each application to assure that either there is no local maximum or the size yielding the maximum local PCT has been found. Break sizes down to 0.01 $ft^2$ should be considered.*

Response

The analyses done to support the evaluations reported in this document were not done to predict limiting PCT consequences for SBLOCA, therefore the requirement does not apply.

11. *B&W-designed plants have internal reactor vessel vent valves (RVVVs) that provide a path for core steam venting directly to the cold legs. The BWNT LOCA evaluation model credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction seal to clear during blowdown and then reform during reflood before the evaluation model analyses predict average core quench. Since the REFLOOD3B code cannot predict this reformation of the loop seal, FTI is required to run the RELAP5/MOD2-B&W system model until the whole core quench, to confirm that the loop seal does not reform. This documentation should be performed at least once for each plant type (raised loop and lowered loop) and be judged applicable for all LBLOCA break sizes.*

Response

This restriction applies to post-blowdown RCS behavior following a large break LOCA. None of the cases analyzed to support the evaluations in this document were for break sizes sufficient to produce formation of a cold leg pump suction seal. Therefore, the restriction does not apply.

## IV. Interface of RELAP5 Use and CRAFT/THETA (FSAR Chapter 14)

The use of RELAP5 for the letdown line failure accident analyses has been evaluated to determine if it is compatible with the use of CRAFT2 in the CR-3 FSAR Chapter 14 accident analyses. This accident consists of an isolatable SBLOCA that is performed to determine the offsite dose consequences of a failure in a line carrying primary coolant outside of the reactor building. RELAP5 was used because the code allows for more detailed modeling of the system but still produces a conservative prediction of the RCS pressure response. The higher RCS pressure (as predicated with RELAP5), coupled with a conservative critical flow calculation, results in a greater inventory loss through the broken letdown line and maximizes the dose consequences. It is concluded that no interface problems exist as a result of the use of the two codes because (1) the RELAP5 code has been reviewed and accepted by the NRC for application to SBLOCA transients, and (2) the analysis performed is more conservative and more reflective of the plant response to the letdown line failure accident.

## V. Licensing Basis

Based on the above discussions, the NRC's criteria for reliance on BAW-10192-P have been satisfied. Accordingly, for purposes of the letdown line failure accident, the licensing basis code is RELAP5 (BAW-10192-P) as approved by the NRC in its letter dated February 18, 1997.