January 27, 1998

Roy A. Anderson
Senior Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (SA2A)
Crystal River Energy Complex
15760 W Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR GENERATING PLANT UNIT 3 - ISSUANCE OF LICENSE AMENDMENT RE: LICENSE CONDITION 2.C.(5) FOR FLOW INDICATION (TAC NO. M99120)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR3). The amendment is in response to your letter dated October 31, 1997 to revise License Condition 2.C.(5) to delete the requirement relating to installation and testing of flow indicators in the emergency core cooling system to provide indication of 40 gallons per minute flow for boron dilution. By letters dated December 1, and 13, 1997, and January 19, 1998, you provided supplemental information which did not affect the original no significant hazards consideration determination. The enclosed amendment also incorporates a new License Condition 2.C.(11) for License No. DPR-72. Your staff has reviewed this License Condition and in a telephone call on January 13, 1998, agreed to the requirement

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
L. Raghavan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

DF01

Docket No. 50-302
Enclosures:
1. Amendment No. 164 to DPR-72
2. Safety Evaluation
cc w/enclosures: See next page

Distribution
Docket File, Public, CR3 Reading, B. Boger
J. Zwolinski, G. Hill (2), ACRS, W. Beckner
W. Lyon, T. Harris(E-mail SE only TLH3)

DOCUMENT NAME: G:\CRYSTAL\99120.AMD

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#41

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

January 27, 1998

Mr. Roy A. Anderson
Senior Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (SA2A)
Crystal River Energy Complex
15760 W Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR GENERATING PLANT UNIT 3 - ISSUANCE OF LICENSE AMENDMENT RE: LICENSE CONDITION 2.C.(5) FOR FLOW INDICATION (TAC NO. M99120)

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

L. Raghavan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures: 1. Amendment No. 164 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

Mr. Roy A. Anderson
Florida Power Corporation

CRYSTAL RIVER UNIT NO. 3

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Corporate Counsel
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MAC-A5A
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Nuclear Plant Operations (NA2C)
Florida Power Corporation
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Mr. Bruce J. Hickle, Director
Director, Restart (NA2C)
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61 Forsyth Street, SW., Suite 23T85
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated October 31, 1997, as supplemented December 1, and 13, 1997, and January 19, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended as indicated in the attachment to this license amendment, to read as follows:

Paragraph 2.C.(5)

"Within six months of the date of issuance of this license, Florida Power Corporation shall complete modifications to the level indication of the borated water storage tank, ~~installation and testing of flow indicators in the emergency core cooling system to provide indication of 40 gallons per minute flow for boron dilution,~~ and installation of dual setpoint pilot-operated relief valve on the pressurizer."

Paragraph 2.C.(11)

A system of thermocouples added to the decay heat (DH) drop and Auxiliary Pressurizer Spray (APS) lines, capable of detecting flow initiation, shall be operable for Modes 4 through 1. Channel checks of the thermocouples shall be performed on a monthly basis to demonstrate operability. If either the DH or APS system thermocouples become inoperable, operability shall be restored within 30 days or the NRC shall be informed, in a Special Report within the following fourteen (14) days, of the inoperability and the plans to restore operability.

The license is hereby also amended to authorize revision of the Final Safety Analysis Report (FSAR) as set forth in the application for amendment by Florida Power Corporation dated December 1, and 13, 1997, and January 19, 1998, for monitoring sump boron concentration including concentration delay times and application of 25% safety factor as discussed in the attached safety evaluation. The licensee shall submit the revised FSAR authorized by this amendment describing procedure EM-225B with the next update of the FSAR in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Pages 4a, 5 and 5a of License DPR-72*

Date of Issuance: January 27, 1998

*Pages 4a, 5, and 5a are attached for convenience, for the composite license to reflect this change.

i) SR 3.8.4.5 shall be successfully demonstrated prior to entering MODE 4 on the first plant start-up following Refuel Outage 9.

j) SR 3.8.7.1 shall be successfully demonstrated no later than 7 days following the implementation date of the ITS.

k) SR 3.8.8.1 shall be successfully demonstrated no later than 7 days following the implementation date of the ITS.

2.C.(3) Florida Power Corporation shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation until safety analyses for less than three pump operation have been submitted by the licensees and approval has been granted by the Commission by amendment to this license.

2.C.(4) DELETED per Amendment No. 20 dated 7-3-79.

2.C.(5) Within six months of the date of issuance of this license, Florida Power Corporation shall complete modifications to the level indication of the borated water storage tank, and installation of dual setpoint pilot-operated relief valve on the pressurizer.

Deleted Per Amdt. #21, 7-3-79

~~2.C.(6) Prior to startup following the first regularly scheduled refueling outage, Florida Power Corporation shall install, to the satisfaction of the Commission, a long-term means of protection against reactor coolant system overpressurization.~~

2.C.(7) Prior to startup following the first regularly scheduled refueling outage, Florida Power Corporation shall modify to the satisfaction of the Commisison, the reactor coolant system flow indication to meet the single failure criterion with regard to pressure sensing lines to the flow differential pressure transmitters.

2.C.(8) Within three months of issuance of this license, Florida Power Corporation shall submit to the Commission a proposed surveillance program for monitoring the containment for the purpose of determining any future delamination of the dome.

2.C.(9) Fire Protection

Amdt. #147, 1-22-93

Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985 and March 16, 1988, subject to the following provisions:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Added per Amdt. 89, 5/23/86

2.C.(10) The design of the reactor coolant pump supports need not include consideration of the effects of postulated ruptures of the primary reactor coolant loop piping and may be revised in accordance with Florida Power Corporation's amendment request of April 24, 1986.

2.C.(11) A system of thermocouples added to the decay heat (DH) drop and Auxiliary Pressurizer Spray (APS) lines, capable of detecting flow initiation, shall be operable for Modes 4 through 1. Channel checks of the thermocouples shall be performed on a monthly basis to demonstrate operability. If either the DH or APS system thermocouples become inoperable, operability shall be restored within 30 days or the NRC shall be informed, in a Special Report within the following fourteen (14) days, of the inoperability and the plans to restore operability.

Amendment 107 - 8/1/88

2.D Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Crystal River Nuclear Power Plant, Unit 3, Modified Amended Security Plan," with revisions submitted through April 14, 1988; "Crystal River Nuclear Power Plant, Unit 3, Security Training and Qualification Plan," with revisions submitted through July 29, 1987; and "Crystal River Nuclear Power Plant, Unit 3, Safeguards Contingency Plan," with revisions submitted through July 23, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

2.D.(2) "Crystal River Nuclear Plant Unit 3 Security Training and Qualification Plan", Revision 3, dated December 30, 1981, submitted by letter dated March 19, 1982, and consisting of all previous revisions. This plan shall be followed in accordance with 10 CFR 73.55(b)(4), 60 days after approval by the Commission. All security personnel, as required in the above plans, shall be qualified within two years of this approval. The licensee may make changes to this plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of and submit reports concerning such changes in the same manner as required for changes made to the Security Plan and Safeguards Contingency Plan pursuant to 10 CFR 50.54(p).

Added per Amdt. #62 3-4-83

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

Operating License No. DPR-72 for the Crystal River 3 Nuclear generating Unit (CR3), includes a License Condition 2.C.(5) which states: "Within six months of issuance of this license, Florida Power Corporation shall complete modifications to the level indication of the borated water storage tank, installation and testing of flow indicators in the emergency core cooling system [ECCS] to provide indication of 40 gallons per minute flow for boron dilution, and installation of dual setpoint pilot-operated relief valve on the pressurizer." By letter dated October 31, 1997, as supplemented December 1, and 13, 1997, and January 19, 1998, Florida Power Corporation (FPC or the licensee), requested to modify the license condition and delete the requirement for flow indicators, DH-45-FI and DH-46-FI, in the ECCS to provide flow indication of 40 gallons per minute flow for boron dilution. The FPC letters dated December 1, and December 13, 1997, and January 19, 1998, provided supplemental information which did not affect the original no significant hazards consideration determination.

2.0 BACKGROUND

In 1976, FPC, Babcock&Wilcox, and the NRC determined that measurement of a flow rate of at least 40 gpm in the decay heat (DH) drop line and in the auxiliary pressurizer spray (APS) line would ensure prevention of boron precipitation for the dump-to-sump (DTS) and APS methods, respectively. This resulted in the requirement for DH-45-FI and DH-46-FI in License Condition 2.C.(5). FPC has determined that these flow instruments are no longer needed and accordingly, proposed to revise the License Condition and delete the requirement for the flow instruments.

DH-45-FI is an ultrasonic flow rate instrument that is located on the pipe that connects between a reactor coolant system (RCS) hot leg and the DH removal system. This pipe is normally used during DH removal operation to provide RCS water to the DH pumps. It will be referred to as the DH drop line. DH-46-FI is an ultrasonic flow rate instrument that is located on a pipe that connects between downstream of DH removal heat exchanger 3A and the pressurizer spray pipe to provide APS flow from the low pressure safety injection (LPSI) system to the pressurizer. (Flow can be provided from downstream of DHR heat exchanger 3B via cross-connects.) These instruments and associated data were used in the past for the sole purpose of controlling or mitigating boron precipitation in a post loss-of-coolant-accident (LOCA) plant condition.

3.0 EVALUATION

The issue addressed in this safety evaluation (SE) is whether the flow instruments DH-45-FI and DH-46-FI are required for boron control purposes.

The purpose of DH-45-FI and DH-46-FI was to enable determination of DTS and APS line flow rates, respectively, of at least 40 gpm when expectations were that these flow rates would achieve boron control under conditions where the original DTS and APS methods, or modifications to those methods, would be applied. Consequently, the question to be addressed is:

> Is information that would be provided by either of these instruments necessary for boron mitigation that involves flow past either of the flow instrument locations?

FPC has added thermocouples to the DH drop and APS injection lines with control room indication of both temperature indications. FPC has developed and proposes to implement an operational procedure to monitor sump boron concentration. These operational requirements are fundamental to the staff's resolution of the issue. The licensee confirmed in a letter dated January 19, 1998, that the requirements had been imposed. Each issue is addressed below, followed by a discussion of the need for each of the flow instruments.

3.1 Thermocouple Indications

FPC has installed strap-on thermocouples on the DH drop and APS injection lines outside containment with indication in the control room. DH drop line temperature indication prior to initiation of flow will be approximately the local auxiliary building temperature. Any flow from the RCS hot leg toward the sump will cause an indicated temperature increase because temperature upstream of the thermocouple location will be higher prior to the initiation. A similar rationale applies to initiation of flow in the APS pipe that presently contains DH-46-FI. Since these thermocouples will provide indication of flow initiation for boron control methods that involve flow past these locations, a license condition has been included which requires the thermocouples to be operational for Modes 4 through 1.

3.2 Sump Boron Concentration Monitoring

The licensee has proposed to monitor the reactor building sump concentration as a more direct indicator of boron concentration in the core, and believes that it is a better indicator of boron concentration in the core than flow rate. The licensee has an on-line post accident sampling system (PASS) designed to sample and evaluate various sample streams during an accident, including the reactor building sump.[1] A PASS sub-assembly would be used for automated gamma isotopic and boron analyses of reactor building sump water. Alternatively, samples representative of sump water may be obtained from vent valves located on DH system piping that is circulating sump water, which addresses failure of the PASS system or plugging of sample piping and associated valves. The PASS boronometer will be used to provide feedback

[1]The PASS system was approved by reference 3 and the boronometer was added later as an improvement under 50.59.

to the control room and Technical Support Center (TSC) to ensure adequate core boron dilution is taking place.[2]

The licensee developed a correlation of sump boron concentration by the following steps:

(1) Calculation of acceptable boron concentration in the core as a function of incore water temperature indication using calculation methods the staff has previously accepted.[3,4]

(2) Calculation of sump boron concentration that would exist for the core boron concentration calculated in step 1 if water in the bottom of the reactor building, including the sump, is well mixed.

(3) Calculation of sump boron concentration that would exist if no boron concentrated in the core.

(4) Plotting Item 3 minus Item 2 (ΔC) as a function of incore temperature indication. This correlation is provided as Attachment 1 to this SE.

An observed value of ΔC below the plotted line, with acceptable allowance for concentration nonuniformity and delay times, indicates an acceptable core boron concentration and is a safe operating region. A value above the line is an indication that more boron has accumulated in the core than permitted, and is unacceptable.

FPC will apply a 25% safety factor (multiply ΔC by 0.75) to Curve 1 discussed above. The correlation would be applied following a LOCA unless (1) the time has been too short for core boron concentration to reach the saturation curve (Item 1, above) or (2) if incore temperature indication shows that an adequate subcooling margin has been established for a sufficient time to ensure that boron precipitation cannot occur. By letter dated January 19, 1998, the licensee stated that boron sump concentration monitoring is in a controlled procedure. In its January 19, 1998 letter, the licensee stated that "[A] 25 % factor of safety has been applied to the Reactor Building (RB) Emergency Sump boron concentration difference curve. This factor of safety has been required for use except when either of the following conditions apply:

a. [T]he time following the loss of coolant accident (LOCA) is less than the minimum time required to reach the core boron solubility limit curve; or

b. [A]dequate subcooling has been established for sufficient time following the LOCA to ensure that boron precipitation cannot occur.

This requirement has been incorporated in EM-225B, "Post-Accident Boron Concentration Management."

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[2]Emergency Plan Implementing Procedure CH-632D, "Post Accident Sampling and Analysis of Reactor Building Sump," is included as Enclosure 3 of FPC 12/4/97. In addition to direct sampling considerations, this also identifies such topics as radiation monitoring and ALARA pre-planning, and specifies allowable dose rates.

[3]Guidance for these calculations is provided in references 1 and 2. These are included as Attachment 2 to this SE since they may not be readily available.

[4]The licensee has also included addition of cool water that has the potential to cause precipitation.

Based on the above, the staff finds this correlation with application of the 25% safety factor acceptable. By this amendment, the staff requires that this procedure be described in the next update of the FSAR.

The correlation is independent of the boron mitigation methods that are to be used. The aspects of nonuniformity and delay times are specific to the boron mitigation method that would be used and are not addressed in this SE. These aspects will be addressed in the staff's SE associated with FPC's amendment request relating to boron mitigation methodology which is currently under staff review.

Determination of sump boron concentration as summarized is fundamental to the effectiveness of boron control and the licensee has proposed using this method in place of DH-45-FI and DH-46-FI. A license condition has been included which requires that this method of sump boron concentration monitoring be provided. FPC's procedure requires operability of the thermocouples during these modes and therefore, is acceptable.

3.3 <u>DH-45-FI - DH drop line flow rate instrument.</u>

The original function of DH-45-FI was to provide flow rate from the RCS hot leg to the sump following initiation of a method involving use of this line to transfer water from the RCS hot leg to the sump. This would provide operators and TSC personnel with information to assess whether the anticipated flow rate was achieved and to monitor flow rate during operation.

The function of DH-45-FI will be met by the combination of thermocouple response to initiation of flow and reactor building sump monitoring. The former fulfills the function of short-term feedback that flow started in response to operator action; the latter fulfills the function of long term feedback that the action is continuing since, should the action not continue, sump boron will not respond as anticipated. Sump boron monitoring additionally provides feedback regarding the effectiveness of the action, feedback that was not provided by DH-45-FI. Consequently, the licensee's request to remove requirements applicable to DH-45-FI is acceptable..

3.4 <u>DH-46-FI - APS flow rate.</u>

The original function of DH-46-FI was to provide flow rate to the pressurizer during use of the APS line between the LPSI system and the pressurizer to control boron concentration. This would provide operators and TSC personnel with information to assess whether the anticipated flow rate was achieved and to monitor flow rate during operation.

The pressurizer will contain steam, with perhaps a small quantity of non-condensable gas, for LOCAs of concern with respect to boron. Initiation of flow through the APS line will be indicated by a thermocouple response followed by an increasing pressurizer level as cold APS water condenses steam, a level change that is reflective of the initiation of APS flow. This flow will reduce pressurizer pressure, causing an inflow from the hot leg to the pressurizer. Water will not flow from the pressurizer into the RCS until this pressure condition changes, which will require that sufficient steam is condensed such that pressurizer pressure becomes greater than pressure at the hot leg junction with the pressurizer surge line. Consequently, it will be one to three hours, depending upon APS flow rate, before the APS system can provide water to the reactor vessel. Level monitoring during this time allows assessment of the adequacy of the APS flow rate. Once APS water is flowing into the reactor vessel, the effectiveness can be monitored via reactor building sump boron concentration monitoring, as discussed above.

The function of DH-46-FI will be met by the combination of thermocouple response to initiation of APS flow, pressurizer level monitoring, and reactor building sump monitoring. Consequently, DH-46-FI is no longer needed, and the licensee's request to remove requirements applicable to DH-46-FI is acceptable.

4.0 STATE CONSULTATION

Based upon the written notice of the proposed amendments, the Florida State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 60733). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments:
1. Core Boron Concentration Control Limits Chart
2. Letter to PWR SEP Licensees from D. Crutchfield, 6/23/81

Principal Contributor: W. Lyon

Dated: January 27, 1998

REFERENCES

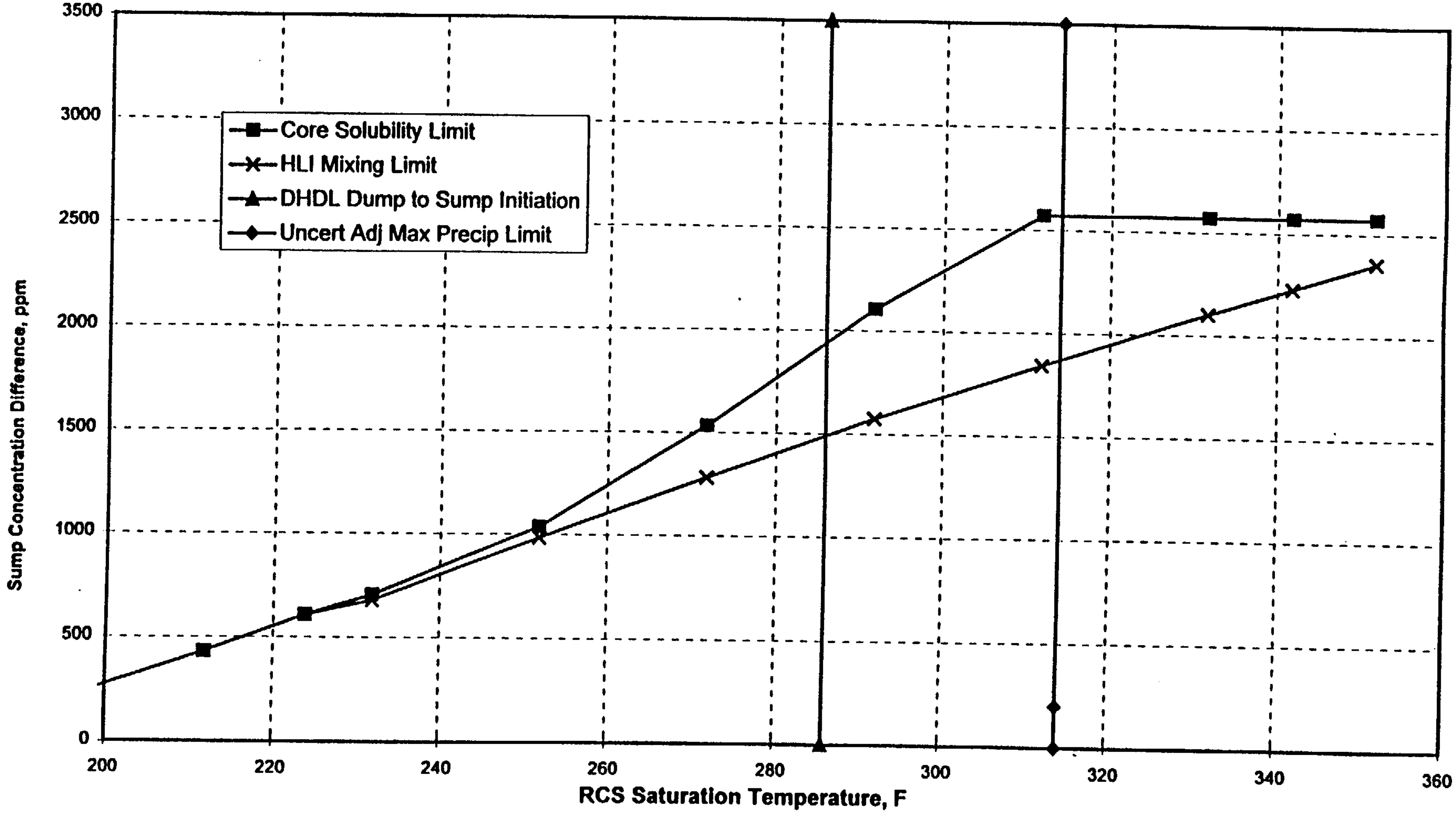
1. NRC letter to all PWR SEP Licensees dated June 23, 1981, "SEP Topic IX-4, Boron Addition System,"

2. Parczewski 76 Parczewski, K. I., "Concentration of Boric Acid in Reactor Vessel During Long Term Cooling - Method for Reviewing Appendix K Submittals," NRC Memorandum to Thomas M. Novak, Chief, Reactor Systems Branch thru Robert L. Baer, Chief, Reactor Safety Branch, January 21, 1976.

3. NRC letter dated November 12, 1985, to FPC

4. NRC letter dated March 3, 1993 from Ashok C. Thadani to P. S. Walsh, B&W Owners Group from Director, Division of Systems Safety and Analysis, "Post-LOCA Reactor Vessel Recirculation to Avoid Boron Precipitation,"

Figure 12. Core Boron Concentration Control Limits.



ATTACHMENT 1

ATTACHMENT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
June 23, 1981

LS05-81-06-079

LETTER TO ALL PWR SEP LICENSEES

Gentlemen:

SUBJECT: SEP TOPIC IX-4, BORON ADDITION SYSTEM

In our Safety Evaluation (SE) of Topic IX-4 for the San Onofre Plant, Section IV, "Review Guidelines", references a memo, dated January 21, 1976 entitled, "Concentration of Boric Acid in Reactor Vessel during Long Term Cooling - Method for Reviewing Appendix K Submittals", which was not included in the SE and is not available from the Public Document Room. Attached is this memo for your information. We are also filing this memo in the Public Document Room.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

~~8106290351~~ 1p.

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

JAN 21 1978

Thomas M. Novak, Chief, Reactor Systems Branch, SS
THRU: Robert L. Baer, Chief, Reactor Safety Branch, OR RLB

CONCENTRATION OF BORIC ACID IN REACTOR VESSEL DURING LONG TERM COOLING - METHOD FOR REVIEWING APPENDIX K SUBMITTALS

Attached is a memorandum entitled: "Concentration of Boric Acid in Reactor Vessel during Long Term Cooling - Method for Reviewing Appendix K Submittals". The memorandum is intended for the reviewers of Appendix K submittals. It describes the methods used in reviewing the calculations of boric acid buildup during a post-LOCA long term cooling.

K. I. Parczewski

K. I. Parczewski
Reactor Safety Branch
Division of Operating Reactors

Attachment:
As stated

cc: D. Ross
D. Eisenhut
Reactor Systems Branch Members
Reactor Safety Branch Members

~~8107020118~~ 12 pp.

Enclosure 1

CONCENTRATION OF BORIC ACID IN REACTOR VESSEL DURING LONG TERM COOLING - METHOD FOR REVIEWING APPENDIX K SUBMITTALS

I. Description of the Problem

Following a LOCA, boric acid solution is introduced into the reactor vessel by two modes of injection. In the initial injection mode, borated water is provided from the accumulators from the refueling water storage tank and from the boron injection tank (Westinghouse plants only). After this initial period, which may last somewhere between 20-60 minutes, the ECC system is realigned for the recirculation mode. In this mode borated water is provided from the containment sump. It is recirculated from the sump to the reactor vessel and back to the sump through the break. A portion of the water introduced into the reactor vessel is converted into steam by the decay heat generated in the core. Since the steam contains virtually no impurities, the boric acid content in the water that was vaporized remains in the vessel. The concentration of boric acid in the core region will therefore continuously increase, unless a dilution flow is provided through the core. Without the dilution flow the concentration of boric acid will eventually reach the saturation limit and any further increase in boric acid inventory will cause its precipitation. Boric acid deposited in the core may clog flow passages and seriously compromise the performance of the ECC system. To prevent this from happening, the ECCS should be designed and operated in such a manner that a sufficient throughflow is provided before the concentration of boric acid will reach its saturation limit. The general performance criteria for the boron dilution systems are given in Appendix I to this memo.

There are two situations when ECC system cannot provide sufficient diluting flow through the core. It occurs during:

(a) cold leg injection with cold leg break

(b) hot leg injection with hot leg break

In both these cases the injected fluid does not have enough head to pass through the core. Since it is not possible for an operator to distinguish between cold and hot leg breaks, the only way of assuring dilution flow through the core is to provide one of the following modes of operation for the ECC system:

(a) alternate injection to cold and hot legs.

(b) simultaneous injection to cold and hot legs.

(c) simultaneous cold leg injection and hot leg suction.

II. <u>Methods Proposed by the Vendors for Preventing Boric Acid Concentration</u>

1. Babcock and Wilcox

The B&W plants are unique in that they possess internal vent valves between the upper plenum and the downcomer. The existence of these vent valves allows natural circulation to take place in the reactor vessel as long as the level of the two-phase mixture is high enough to reach the vent valves and keep them open. B&W has performed an analysis indicating that regardless of hot or cold leg break, the natural circulation flow through the core could be maintained for over 30 days after a LOCA and no operator action is needed during that period of time (References 1, 2 and 3). After 30 days B&W proposes three different modes of operation, aiming at establishing diluting flow through the core. They are as follows:

(a) injection into the downcomer and suction or drainage from the hot leg through the decay heat line.

(b) simultaneous injection into the downcomer through the injection line and into the hot leg through the decay heat line.

(c) simultaneous injection into the downcomer through the injection line and into the hot leg through the auxiliary pressurizer spray line.

B&W claims that any of these methods, if initiated within 30 days, will prevent boric acid concentration from reaching the saturation limit.

The NRC staff has reviewed the claims made in the B&W analysis and found that, indeed, in most plants natural circulation can provide dilution flow through the core well in excess of 30 days. The modes of operation, proposed by B&W for maintaining dilution flow beyond this time have to be reviewed individually for each plant because their performance depends on the parameters of each individual plant.

2. Combustion Engineering

In the Combustion Engineering plants the initial injection of borated water is through the cold legs. After some specified time, cold leg injection is replaced either by simultaneous hot and cold leg injection, or by simultaneous hot leg suction and cold leg injection. The time when the switchover must take place is determined by the concentration of boric acid in the reactor vessel.

In the analysis (Reference 4) Combustion Engineering makes two assumptions. It assumes ideal mixing of boric acid solution in the reactor vessel which produces uniform concentration buildup in the whole vessel and it assumes that in all cases there is a residual pressure of at least 20 psia in the reactor vessel. This pressure causes higher boiling liquid temperatures with the resulting increase of 4 1/2 weight percent in boric acid solubility. Both these assumptions are non-conservative and Combustion Engineering does not provide satisfactory justification for including them in the analysis.

In the procedure provided by Combustion Engineering, hot leg suction is accomplished by use of the shutdown cooling suction line and the hot leg injection can be made either through the shutdown cooling suction line or through the auxiliary pressurizer spray line. In most plants these three alternatives provide a system that can withstand a single active failure.

The procedure proposed by Combustion Engineering has certain limitations. If the steam produced in the vessel cannot be freely vented through the hot leg, it can depress the level of the liquid in the upper plenum below the bottom of hot leg nozzles and impede the suction through the shutdown cooling line. This case will occur when all loop seals, formed by the U bend in cold legs, are filled with water. On the other hand, if the steam leaving the vessel reaches velocities too high it may carry the injected fluid into the steam generator and prevent it from reaching the core.

Both these problems have been addressed by Combustion Engineering (References 5 and 6) and it was demonstrated that, for the plants considered, these effects did not seriously affect the performance of ECC system.

3. Westinghouse

After an initial period of cold leg injection, Westinghouse recommends hot leg injection as means for providing dilution flow through the core. The generic analysis performed by Westinghouse (References 7 and 8) assumes that the fluid injected through the hot leg to the upper plenum mixes with the fluid in the reactor vessel. Hot leg injection, therefore, provides a continuous dilution flow through the core for both hot and cold leg breaks. The Westinghouse analysis, which forms the basis for these assumptions, is not complete. Until more information is obtained, it is recommended that the initial cold leg injection should be replaced either with simultaneous hot and cold leg injection, or alternate periods of hot and cold leg injection, so that adequate dilution flow is provided for either a hot leg or cold leg break.

III. NRC Review

1. General Description

The NRC review of the submitted procedures includes independent evaluation of basic parameters (switchover time, minimum flow rates, etc.). The independent evaluations have been performed for the following reasons:

(a) In some cases the information provided by the applicants were incomplete and it was not possible to check the soundness of their initial assumptions or their methods of analysis.

(b) In order to assure that under all operating conditions the procedures provided by the applicant will guarantee the maintenance of safe boric acid concentrations, the NRC staff proposed a series of conservative assumptions. It is necessary to determine how this additional conservatism would affect the procedures proposed by the applicants.

One of the most important assumptions introduced by the staff, was the four weigh percent margin in the saturation concentration of boric acid in the core region. This safety margin was introduced to take care of inaccuracies in predicting saturation concentrations in the core.

The NRC review consists of two parts:

(a) analysis (calculation) of boric acid concentrations

(b) hardware review

2. Analytical Methods

The following methods were used in calculating different performance parameters for the systems reviewed:

(a) Switchover time from cold leg to hot leg (or simultaneous hot and cold leg) injection.

In order to determine the switchover time the following conservative assumptions were made:

(1) During the initial cold leg injection, boric acid does not leave the core.

(2) The maximum concentration of boric acid in the core region (C_c) should not exceed 23.5 w/o (4 w/o below the saturation concentration at 212°F).

(3) The initial injection period lasts for 20 minutes after a LOCA. At this time the concentration of boric acid (C_o) is uniform in the whole plant.

(4) The effective volume in which boric acid buildup occurs consists of a volume of liquid in the core and in the upper plenum up to the height of the bottom of the cold leg nozzle.

The rate of boric acid buildup is expressed by the following differential equation:

$$\frac{dC}{dt} = A \cdot t^{-n} \cdot (C_\infty - C)$$

Where: C - boric acid concentration in core region

C_∞ - maximum boric acid concentration if all boric acid were concentrated in core region

t - time

A - group depending on several plant parameter

n - constant

Solving the equation for the following boundary conditions:

$t = t_o,\ C = C_o$

$t = t_s,\ C = C_c$

and rearranging the terms, the following expression for the switchover time (t_s) is obtained:

$$t_s = t_o \cdot \left[1 - \frac{1-n}{A \cdot t_o^{1-n}} \cdot \ln\left(\frac{C_\infty - C_c}{C_\infty - C_o}\right)\right]^{1/(1-n)}$$

Where: $A = \dfrac{K \cdot P}{(\Delta h_f + \Delta h_{fg}) \cdot M_s}$

for: $t_s < 4 \times 10^6$ sec (46 days): $K = 0.156$ and $n = 0.283$

M_s - mass of water in the containment sump, lb

P - reactor operating power, Btu/sec

C_c - 0.3077 boric acid/water

In the case of alternate hot and cold leg injections, the subsequent switchover times could be obtained by substituting t_{s1}, t_{s2}, t_{s3} etc for t_o in the above equation.

(b) Minimum cold leg injection

The minimum cold leg injection required during recirculation (prior to switchover to dilution mode) is determined by the rate of boil-off in the core at the very beginning of the long term cooling mode.

It is determined from the following expression:

$$Q_c = \frac{449\ H_o}{\rho_{in} \cdot (\Delta h_f + \Delta h_{fg})} \text{ gpm}$$

Where: H_o - decay heat generated at the beginning of recirculation period, Btu/sec

ρ_{in} - density of injected fluid, lb/ft^3

Δh_f - subcooling enthalpy, Btu/lb

Δh_{fg} - vaporization enthalpy, Btu/lb

(c) Minimum hot leg injection

The minimum injection rate, after the switchover from cold to hot leg or to simultaneous hot and cold leg injections, is determined by making conservative assumptions that the minimum injection should provide enough flow to replace the boiled-off liquid and to maintain sufficiently high dilution flow through the core. The boiloff rate is calculated using the decay heat generation rate at the switchover time from cold to hot leg injection. The dilution flow is determined for the maximum allowable concentration of boric acid.

Expression for the minimum injection rate is given below:

$$Q_h = \frac{449\ H_s}{\rho_{in}\left[\Delta h_f + (1 - 3.25\ C_o)\ \Delta h_{fg}\right]}$$

Where: H_s - decay heat generated at switchover time, Btu/sec. The decay heat is calculated using the methods given in Reference 9.

(d) Steam binding

When the switchover from cold to hot leg injection occurs shortly after a LOCA, large volume of steam leaving the reactor vessel may induce high steam velocities in outlet nozzles. These velocities impede the injected boric acid solution from reaching the core. In order to assure that hot leg injection is not compromised, steam velocities have to be estimated and the rate of entrainment of the injected liquid calculated. The rate of entrainment can be calculated using the methods given in Reference 10.

3. Procedure

In reviewing the boric acid submittals, the following procedure was used:

Step 1 Review of the submittal and identification of any special features the applicant may be taking credit for.

In most cases the applicants follow the generic analyses developed by the vendors. However, occasionally the applicant may take credit for the phenomena which are not considered in the generic studies. For example, no consideration was given in the generic studies to the possible increase in the solubility of boric acid when sodium hydroxide is present. However, some applicant may use this phenomenon to increase the allowable solubility limits. In such cases the reviewer should identify and individually evaluate such features.

Step 2 Determination of the basis parameters identified in Section III-2 of this memo.

The information needed for the determination of these parameters are either provided in the original submittal or can be found in FSAR or PSAR.

Step 3 Hardware review

Using the P&I diagrams or special schematic diagrams provided by the applicant, the systems are reviewed to assure that they meet the requirements identified in the analytical part of the review, without violating the single failure criteria.

IV. Conclusions

The procedure for reviewing nuclear plant for boric acid precipitation is outlined in this memorandum. It should be mentioned that although this procedure may apply to the majority of plants reviewed, there may be some cases where the reviewer may be forced to use completely different approach. He should be, therefore, flexible in choosing his procedures, guided by the considerations described in Appendix I. Note the Appendix I is intended to apply fully only to CP applications.

V. References

1. Babcock and Wilcox, Topical Report BAW-10102, ECCS Evaluation of B&W's 205-FA NSS, June 1975.

2. Babcock and Wilcox, Topical Report BAW-10103, ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, June 1975.

3. Babcock and Wilcox, Topical Report BAW-10105, ECCS Evaluation of B&W's 177-FA Raised-Loop NSS, June 1975.

4. Switzer, D. C., (NNECO) letter to O. D. Parr (NRC), dated June 27, 1975, attachment A.

5. Switzer, D. C. (NNECO) letter to O. D. Parr (NRC), dated September 25, 1975, attachment 1.

6. Telecon, J. Longo (CE) and RSB personnel (NRC), October 30, 1975.

7. Caso, C. L. (Westinghouse) letter CLC-NS-309 to T. M. Novak (NRC), dated April 1, 1975.

8. Cermak, J. O. (Westinghouse) letter JOC-NS-369 to T. M. Novak (NRC), dated August 15, 1975.

9. ANS, Decay Energy Release Rates Following Shutdown of Uranium - Fueled Thermal Reactors, (Proposed ANS Standard), October 1971.

10. Wallis, G. B., One-dimensional Two-phase Flow, McGraw-Hill Book Company, 1969, Section 12.10.

Appendix I

PWR BORON DILUTION SYSTEMS FOR CP APPLICATIONS WHICH MUST MEET 10 CFR 50.46 CRITERIA FOR ACCEPTANCE

1. The boron dilution function shall not be vulnerable to a single failure. A single active or passive failure postulated to occur during the long term cooling period after a LOCA ~~can be assumed failure would then~~ shall be in lieu of a single active failure during the short term cooling period.

2. The spurious operation of any motor operated valve (open or closed) shall not compromise the boron dilution function nor shall it jeopardize the ability to remove decay heat from the primary system.

3. All components of the system which are within containment shall be designed to seismic Category 1 requirements and classified Quality Group B.

4. The primary mode for maintaining acceptable levels of boron in the vessel should be established. Should a single failure disable the primary mode, certain manual actions outside the control room would be allowed, depending on the nature of the action and the time available to establish back-up mode.

5. The average boric acid concentration in any region of the reactor vessel should not exceed the level of 4 weight percent below the solubility limits at the temperature of the solution.

6. During the post-LOCA long term cooling, the ECC system normally operates in two modes: the initial cold leg injection mode, followed by the dilution mode. The actual operating time in the cold leg injection mode will depend on plant design and steam binding considerations, but, in general, the switchover to the dilution mode should be made between 12 and 24 hours after LOCA.

7. The dilution mode can be accomplished by any of the following means:

   (a) Simultaneous cold leg injection and hot leg suction
   (b) Simultaneous hot and cold leg injections
   (c) Alternate hot and cold leg injections.

8. In the alternate hot and cold leg injection mode, the operating time at hot and cold leg injection should be sufficiently short to prevent excessive boric acid buildup.

9. The minimum ECCS flow rate delivered to the vessel during the dilution mode shall be sufficient to accommodate the boil-off due to fission product decay heat and possible liquid entrainment in the steam discharged to the containment and still provide sufficient liquid flow through the core to prevent further increases in boric acid concentration.

10. All dilution modes shall maintain testability comparable to other ECCS modes of operation (HPI-short term, LPI-short term, etc). The current criteria for levels of ECCS testability shall be used as guidelines (i.e., Regulatory Guides 1.68, 1.79, GDC 37).