Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

April 19, 1996
3F0496-09

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Subject: Information for Reactor Coolant Pump Seal Area Cooler Assessment

Dear Sir:

Florida Power Corporation (FPC) is submitting the attached material which contains internal FPC correspondence dealing with FPC's evaluation of the event described in Information Notice 89-54, "Potential Overpressurization of the Component Cooling Water System." This submittal is requested by Reactor Systems Branch personnel to allow them to complete their assessment of FPC's position regarding the event described in IN 89-54. This assessment by NRR is an outgrowth of Inspector Followup Item (IFI) 50-302/95-15-03.

Sincerely,

Larry Kelley

L. C. Kelley, Director
Nuclear Operations Site Support

LCK/JWT:ff

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

A001
1/1

Florida Power
CORPORATION

INTEROFFICE CORRESPONDENCE

Nuclear Operations Engineering — C2I — 231-4440
OFFICE — MAC — TELEPHONE

SUBJECT: **Crystal River Unit 3**
NOTES Track 18264A
File: SP 90-037

02839

TO: **E. E. Froats**

DATE: **September 14, 1992**
NEA92-1420

Nuclear Operations Engineering (NOE) has completed the Engineering Study to evaluate the alternatives available (including doing nothing) to address the event described in NRC Information Notice 89-054. The event is a break in an RCP seal cooler which results in high pressure RCS fluid discharging into the lower pressure Nuclear Services Closed Cycle Cooling Water (SW) system. Enclosure 2 to AI-404B and copies of the following pages from the study are attached to this IOC as evidence of completion of the corrective action.

Data Transmittal Sheet
Study Cover Page
Study Pages 3, 4, 5, 21, and 22.

The study recommends installation of relief valves in the SW lines upstream and downstream of the RCP coolers to accommodate the pressure surge that would be experienced after such an event. It is important to note that the postulation of the event is not within the licensing basis of CR-3. Should FPC choose to take the regulatory position that the event is not credible, of course the "Do Nothing" alternative would be the most cost-beneficial. For these reasons, the modification will be presented to senior management as a discretionary project via a Request for Project Approval. That RPA will be issued by October 16, 1992. You will be notified of the disposition of the RPA by senior mangement when the decision is made.

Brian Gutherman
Nuclear Engineering Supervisor

cc: J. W. Tunstill (w/attach.)
J. R. Maseda (w/attach.)
File (w/attach)
Records Management (w/attach.)

ENCLOSURE 2

ACTION ADDRESSEE
INDUSTRY OPERATING EXPERIENCE REVIEW FORM

Document Type/Number: IEN 89-054

Additional Related Documents: Engineering Study SP 90-037

Item Number: 18264A

Actions Completed or Planned:	Completion Date or Completion Due Date:
Completed Engineering Study to evaluate alternatives to modify the SW system to mitigate an RCP cooler tube rupture	9/9/92
Issue RPA to senior management for their concurrence to proceede with the recommended modification	10/16/92

Attach documentation for all actions listed as complete!

Justification: This study recommends the installation of relief valves up and downstream at the SW RCP coolers (see attached). However, this project is discretionary since it is beyond CR-3 design basis to postulate such an event. Senior management will have the final call on whether to proceed.

Action Addressee: [signature] Nuclear Engrg Supervisor 9/14/92
Signature Title Date

NOTA:
Signature Title Date

ATTACHMENT

Florida Power

DATA TRANSMITTAL SHEET

PREPARED BY	DATE	FILE CODE
DEH	8/20/92	SP-90-037

PROJECT: Overpressurization of SW System

PLANT: Crystal River Unit #3

TYPE OF DATA
- [ ] Tracing / Sepia / Mylar
- [ ] Prints
- [ ] Aperture Cards
- [ ] Specifications or R.O.'s
- [ ] Bills of Materials
- [ ] Manufacturers Prints
- [ ] Instruction Manuals
- [x] Other

STATUS OF DATA
- [ ] Preliminary
- [ ] For approval
- [x] Approved / Accepted
- [ ] Accepted "as noted"
- [ ] Not accepted
- [ ] "As built" / Final Issue
- [ ] Revised
- [ ] Other

PURPOSE OF DATA
- [ ] Review & Comment
- [ ] Approval
- [ ] Bidding Purposes
- [ ] Construction
- [ ] Fabrication
- [ ] Licensing
- [x] Information Only
- [ ] Other

ACTION TO BE TAKEN
- [ ] Acknowledge
- [ ] Approve & Return
- [ ] Revise & Resubmit
- [ ] See "Special instructions"
- [ ] Certify
- [x] New data / Destroy Old
- [ ] Other
- [ ] Review & submit comments (see below)

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OR	RER	S

No. Cop's	DISTRIBUTION	Purpose
OR	FILE	
1c	PDB	
1c	RER	
1c	FVF	
*	SEE BELOW	

RETURN COMMENTS TO

BY DATE

SPECIAL INSTRUCTIONS

ENGINEERING STUDY

DOCUMENT NUMBER	REV./DATE	TITLE OR DESCRIPTION
SP-90-037	0	ENGINEERING STUDY Overpressurization of SW System.

ADDITIONAL COMMENTS OR INSTRUCTIONS

* MAD (DTS and cover sheet only)

ATTACHMENT

ENGINEER'S APPROVAL	DATE	VERIFICATION APPROVAL	DATE	SUPERVISOR'S APPROVAL	DATE
R.E. Robbins	8/21/92	N/A		[signature]	9/9/92

EVALUATION OF SW SYSTEM OVERPRESSURIZATION

Prepared for:

Florida Power Corporation
3201 Thirty-Fourth Street South
P.O. Box 14042
St. Petersburg, Florida 33733

Prepared by:

ABB Impell Corporation
333 Research Court
Technology Park/Atlanta
Norcross, Georgia 30092

FLORIDA POWER CORPORATION
NUCLEAR ENGINEERING DEPARTMENT
CRYSTAL RIVER - UNIT 3
REVIEWED AND ACCEPTED BY:
Engineer R.E. Robbins Date 9/12/92
Supervisor [signature] Date 9/1/92

July 30, 1992

Job No. 0920-208

ABB Impell Report No. 0920-208-01

Revision 0

Timothy E. Bohlen — Submitted By

J. McCarthy — Reviewed By

John P. Buchanan — Approved By

ATTACHMENT

I. PROJECT DESCRIPTION

FPC contracted ABB Impell to conduct an engineering study to evaluate three alternatives relative to a potential overpressurization of the Crystal River Unit 3 Nuclear Services Closed Cycle Cooling (SW) System due to a failure of the RCP thermal barrier (see Figure #1 for a schematic diagram of the three alternatives). The benefits and costs associated with each alternative have been estimated, with the results tabulated in this report. An industry survey was conducted of seven similarly designed units to determine what actions, if any, have been taken to address the problem. A matrix which summarizes the results is attached as Attachment 1.

II. EXISTING CONDITION/BACKGROUND

NRC Information Notice 89-54 addresses concerns for potential overpressurization of cooling water systems resulting from failure of the tubing in a Reactor Coolant Pump (RCP) seal cooler heat exchanger (thermal barrier). FPC has reviewed IN 89-54 for applicability and has determined that the described event is not within the specific licensing/design basis for Crystal River Unit 3 (CR3). However as a prudency measure, FPC has decided to evaluate possible engineering actions to address the potential overpressurization of the SW System due to the IN 89-54 described event.

The current configuration of the SW system to the RCP thermal barrier cooler is that there are no upstream check valves to isolate any backleakage from the RCP into the supply side of the SW system. Also, there are no downstream isolation valves which have been designed to isolate/contain a leak from the RCS into the SW system through the RCP thermal barrier cooler. The existing SW system piping to and from the RCP thermal barrier is Schedule 40 carbon steel ASTM A106 GR. B piping designed with a 150 psig rating, which is well below the RCS normal operating pressure of 2250 psig (design pressure of 2500 psig).

III. PROJECT JUSTIFICATION

This project is discretionary since this project has a viable "do nothing" alternative and since the licensing basis at CR3 does not require postulation of the described event. In order to gain additional experience from the actions of other similarly designed plants, an industry survey has been conducted by ABB Impell:

INDUSTRY EXPERIENCE SURVEY

Attachment 1 provides a summary of the industry survey which includes design and other relevant data for PWR plants with RCP thermal barrier cooling design that is similar to CR3. Note: The Component Cooling Water (CCW) System or Nuclear Services Closed Cycle Cooling (NSCCC) System are hereafter referred to as the "cooling water system" for simplicity. Significant findings are as follows:

(1) Six (6) of the seven (7) plants have the isolable portion of the cooling water system rated to Reactor Coolant System (RCS) pressure. The one remaining plant (Palisades) is internally committed to modify the cooling water system, but has not finalized a specific action plan.

(2) All seven (7) plants have cooling water system check valve(s) upstream of the RCP thermal barrier cooling coil. Six (6) of the seven (7) plants have upstream check valves that are rated at RCS pressure to eliminate the likelihood of any upstream Loss of Coolant Accident (LOCA) for the described event. The other remaining plant (Palisades) recognizes that their upstream check valve is not rated for the described event but is committed to future modifications to the cooling water system to address that issue.

(3) Two (2) of the seven (7) plants are designed with automatically actuated motor-operated valve (MOV) closure - one uses a high pressure signal (Davis-Besse) and the other uses a high flow rate signal (Surry) to actuate the downstream MOV. Of the other five (5) plants which do not have automatic MOV closure, two (2) (Oconee & Summer) have MOV's downstream which can be manually actuated from the control room. Thus, four (4) of seven (7) plants surveyed have the capability to contain a LOCA within the Reactor Building originating from the described event. TMI 1 took the additional measure of overriding its MOV capability to allow an unobstructed flow path to the letdown cooler heat exchanger relief valves. Thus TMI 1 is not included in the four (4) plants which have the capability to contain a LOCA, rather TMI 1 is included as one of the two (2) plants which utilize pressure relief measures to address the described event.

(4) All seven (7) plants use relief valves (some have additional rupture plugs) to alleviate overpressure conditions in the cooling water system. All seven (7) plants are designed to relieve the high pressure fluid inside of the containment building.

(5) Not all plants have detailed analyses to ensure that the cooling water piping and components will be protected from overpressurization by the relief valves. Instead, these plants do not take credit for the relief valve function during the described event. Thus, shutting down the plant and repairing the leak is the official licensing position taken for the described event.

(6) Two (2) of the seven (7) plants (ANO 1 and TMI 1) depend solely upon relief valves to protect the cooling water piping. At ANO 1, a calculation has been performed which demonstrates that the relief valve manifold will prevent piping rupture and protect the containment isolation valves so that the containment isolation valves will not become inoperative during the described event. At TMI 1, the MOV's are overridden to stay open to allow an unobstructed flow path to the letdown cooler heat exchanger relief valves.

(7) It should be noted that neither NRC Information Notice 89-54 nor the individual FSAR's of any of the plants surveyed mandate modifications to isolate a thermal barrier rupture. Thus, shutting down the plant and repairing the leak is the official licensing position taken by all of the plants surveyed for the described event.

(8) None of the plants utilize rupture plugs as the primary means of overpressure protection. The rupture plugs are only used as a backup means of overpressure protection in case the primary means of overpressure protection (relief valves) fails.

Conclusions of Industry Survey

The results of the industry survey (see Attachment 1) have been factored into the conclusions/recommendations at the end of this report. The industry survey shows that two (2) of the seven (7) plants have designs similar to Alternative 2 (containment of loss of reactor coolant) to address NRC IN 89-54, while four (4) of the seven (7) plants have designs similar to Alternative 3 (prevention of loss of reactor coolant) to address NRC IN 89-54. The remaining plant, Palisades, is currently reviewing this issue to determine their future action.

VIII. CONCLUSIONS/RECOMMENDATIONS

The probability of the event described by NRC IN 89-54 cannot be reasonably determined as there is no relevant failure data. FPC has reviewed IN 89-54 for applicability and has determined that the described event is not within the specific licensing/design basis for Crystal River Unit 3 (CR3). If the probability of occurrence for the event can be determined, then the selection of the best alternative can be performed by selection of the alternative with either the highest benefit-to-cost (B/C) ratio or the highest benefit yield (B-C difference) in Tables 2a through 2e.

If it can be shown that the event has a greater than a 1-in-88 chance of occurrence up to a 1-in-1 (100%) chance of occurrence, then Alternative 3 (Prevention of Loss of Reactor Coolant by Local Isolation Valves at RCP) is the best alternative as shown by the highest benefit yield of $101,872,900 in Table 2a.

If it can be shown that the event has between a 1-in-88 and a 1-in-126 (e.g. 1 in 100, etc...) chance of occurrence, then Alternative 2 (Containment of Loss of Reactor Coolant by Local Relief Valves at RCP) is the best alternative since it is the only alternative with either a B/C ratio greater than one or a positive benefit yield (See Table 2b).

If it can be shown that the event has a probability of 1 chance in 126 or less (e.g. 1 in 1,000 or 1 in 10,000, etc...), then selection of Alternative 1 (Maintain Existing Configuration) is the best alternative, since no additional capital or operation and maintenance costs are incurred and there is the lowest Benefit Minus Cost Difference in Tables 2c through 2e.

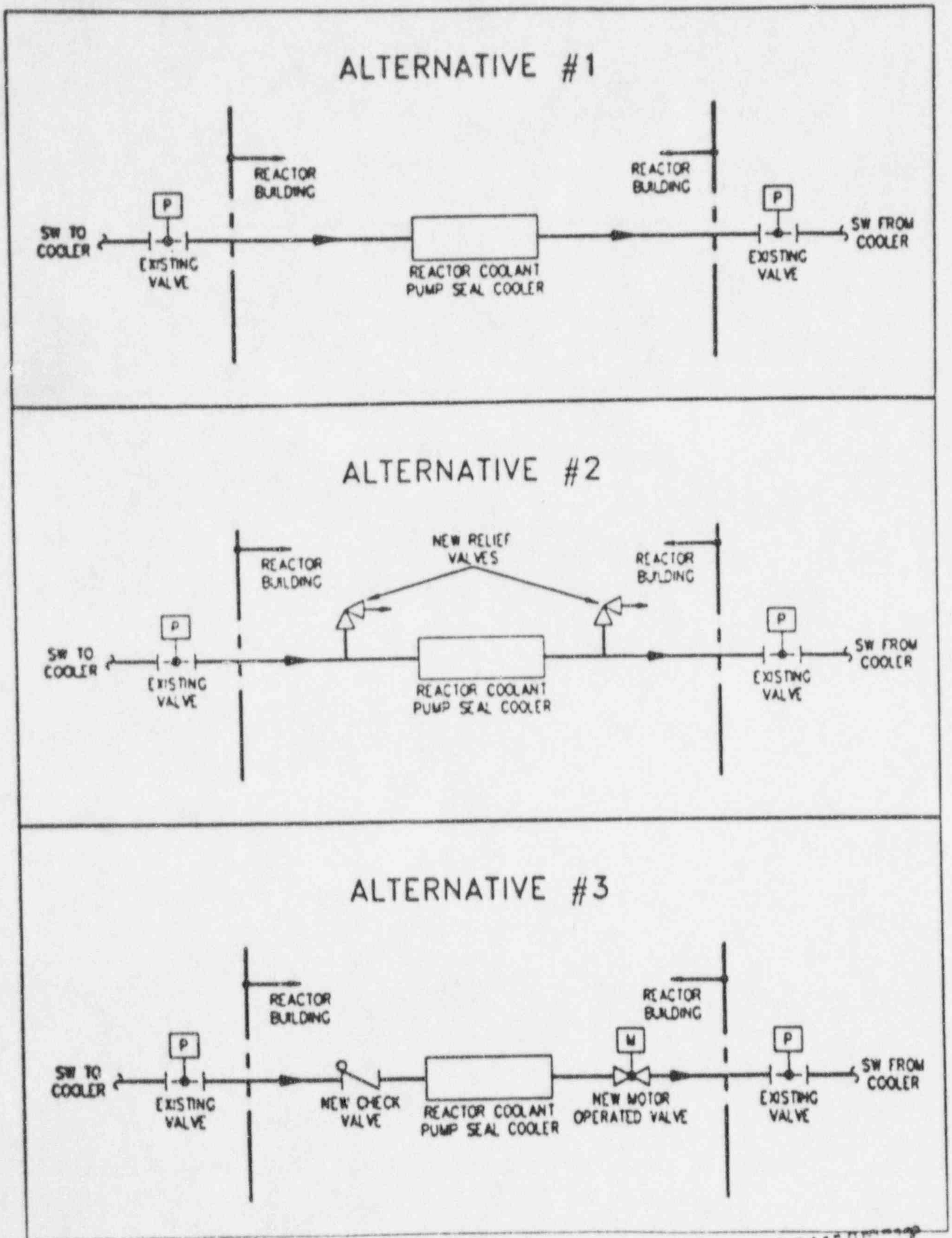
The number of Pressurized Water Reactor (PWR) operating hours in the United States is estimated at approximately 5.80×10^6 hours (662 years) as of July 1992 based upon the commercial operation date and a 70% industry capacity factor. During this period of operations no failures of a reactor coolant pump seal heat exchangers have been reported. Based on this operating experience, it can only be assumed that there is a very low probability of a catastrophic rupture of the RCP seal area cooler, however, the probability of occurrence will never be zero.

A survey of the industry, documented in Attachment #1, determined that all seven plants which were contacted had relief valves installed to prevent overpressurization of the cooling water system. In addition, two plants automatically isolated the cooling water line following a break and two other plants had the capability for remote-manual isolation of the lines from the control room.

In conclusion, ABB Impell recommends that Alternative 2 be considered for implementation. Although, the probability of a seal cooler rupture is low based upon operating experience, the majority of the nuclear industry is designed to mitigate the consequence of this event. ABB Impell feels that it is prudent to remain consistent with the nuclear industry. In addition, Alternative 2 provides a means of mitigating the consequences of this event at a cost of approximately 55% (~ $500,000 less) of Alternative 3 (see summary of Pre-Event Costs in Table 1).

ATTACHMENT

FIGURE #1


ALTERNATIVE #1
REACTOR BUILDING
REACTOR BUILDING
P
P
SW TO COOLER
EXISTING VALVE
REACTOR COOLANT PUMP SEAL COOLER
EXISTING VALVE
SW FROM COOLER
ALTERNATIVE #2
NEW RELIEF VALVES
REACTOR BUILDING
REACTOR BUILDING
P
P
SW TO COOLER
EXISTING VALVE
REACTOR COOLANT PUMP SEAL COOLER
EXISTING VALVE
SW FROM COOLER
ALTERNATIVE #3
REACTOR BUILDING
REACTOR BUILDING
P
M
P
SW TO COOLER
EXISTING VALVE
NEW CHECK VALVE
REACTOR COOLANT PUMP SEAL COOLER
NEW MOTOR OPERATED VALVE
EXISTING VALVE
SW FROM COOLER