NED Proposed Change

Safety Evaluation

No.: 1959 Rev 1

SAFETY EVALUATION
PILGRIM NUCLEAR POWER STATION

Rev. No.

Initiator:	Dept:	Group:	~~PDC PCN~~ No.:	System Name:	Calc. No.:	Date:
DBHeard	NED	FSIMK	TM-86-20	RHR	M-269 Rev.0	6/13/86

Description of Proposed change, test or experiment: Establishment of a controlled leak in the RHR system per Temporary Modification 86-20.

SAFETY EVALUATION CONCLUSIONS:

The proposed change, test or experiment:

1. (✓) Does Not ( ) Does increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

2. (✓) Does Not ( ) Does increase the possibility for accident or malfunction of a different type than any evaluated previously in the FSAR.

3. (✓) Does Not ( ) Does decrease the margin of safety as defined in the basis for any technical specification.

BASIS FOR SAFETY EVALUATION CONCLUSIONS:
Prior to the current outage high pressure alarms for the RHR
system piping were being received at frequent intervals. This was an
undesirable situation as the MO 1001-36 A/B valves were being used to relieve
system pressure. In order to relieve operations personnel from taking
these actions and to provide a positive means for confirming containment
integrity, a controlled leak-off in the RHR system will be established.
RHR system leakoff will be controlled by throttling a bypass valve around
the RHR pump discharge check valve. Containment integrity is not affected
by this change since the controlled leakoff is small compared to the

Change
(✓) Recommended

Change
( ) Not Recommended

SE Performed by David Heard

Date 6/13/86

Exhibit 3.07-A
Sheet 1 of 3

Rev. 2

8608040299 860616
PDR ADOCK 05000293
Q PDR

SE 1959
Rev. 1

MO1001-28B and 29B leakage rate allowed by 10CFR50 Appendix J. The leakoff rate will be set at ≤ 1 GPM. This leak rate will allow stabilization of RHR system pressure yet will allow the operator to be alerted (by an alarm in the control room) to an increase in leakage past the MO1001-28B and 29B valves. The 1 GPM leak is less than the equivalent air leakage allowed by 10CFR50 Appendix J by a factor of approximately 3.6. Calculation M-269 Rev. 0 proves that the ≤1.0 GPM leak rate falls well within the calculated water leakage acceptance criteria based on the data obtained during testing of the B RHR loop. In fact, this modification provides an improved means of assessing containment integrity. In addition, the containment isolation functions of 1001-28B and 29B have not been affected by this change.

Finally, the leak off line will have no impact on RHR pump operation as this modification only provides a leak path around the pump discharge check valve but not the pump itself. All pump flow is still directed to the RHR system.

Safety Evaluation
No.: ~~H~~ 1959 Rev 1

SAFETY EVALUATION
PILGRIM NUCLEAR POWER STATION

Rev. No. (2)

A. APPROVAL

(14) (✓) This proposed change does not involve a change in the Technical Specifications Ref. 10CFR50.59(c).

(14) (✓) This proposed change, test or experiment does ( ) does not (✓) involve an unreviewed safety question as defined in 10CFR, Part 50.59(a)(2).

(15) (✓) This proposed change involves a change to the FSAR per 10CFR 50.71(e) and is reportable under 10CFR50.59(b).

(15) ( ) Comments: ______________________________

(16) The safety evaluation basis and conclusion is:

(✓) Approved ( ) Not Approved

[signature] for LDSC (16) per Telcon 6/13/86
Discipline Group Leader/Date

(17)
Supporting Discipline Group Leader/Date

B. REVIEW APPROVAL

(18) ( ) Comments: ______________________________

(19) JL Rogers for REG 6/13/86
S&SA Group Leader/Date

C. ORC REVIEW

(20) ( ) This proposed change involves an unreviewed safety question and a requests for authorization of this change must be filed with the Directorate of Licensing, NRC prior to implementation.

(20) (✓) This proposed change does not involve an unreviewed safety question.

(21) ORC Chairman [signature] Date 6/16/86 (21)

(22) ORC Meeting Number 86-80

cc:

Exhibit 3.07-A Rev. 2
Sheet 2 of 3

PILGRIM STATION
FSAR REVIEW SHEET

References:

Safety Evaluation: 1959 Rev. No.: 1 Date: 6/13/86

Support a change

List FSAR test, diagrams, and indices affected by this change and corresponding FSAR revision.

Affected FSAR Section	Revision to affected FSAR Section is shown on:	Preliminary	Final
No changes except PID which will be updated separately ~~by NED.~~	Attachment 1		
	Attachment 2		
	Attachment 3		
	Attachment 4		
	Attachment 5		
	Attachment 6		

PRELIMINARY FSAR REVISION (to be completed at time of Safety Evaluation preparation).

Prepared by: JL Rogers /Date: 6/17/86 Reviewed by: N/A /Date: ______

Approved by: N/A ____/Date: ______

FINAL FSAR REVISION (Prepared following operational turnover of related systems structures of components for use at PNPS). (1)

Prepared by: ________/Date: ______ Reviewed by: ________/Date: ______

(1) Attach completed FSAR Change Request Form (Refer to NOP).

Exhibit 3.07-A
Sheet 3 of 3

Rev. 2

Safety Evaluation No.: 1959

SAFETY EVALUATION WORK SHEET

Rev. No. 1

A. System Structure Component Failure and Consequence Analyses.

	System Structure Component	Failure Modes	Effects of Failure	Comments
1.	—— See	attached	——	
2.				
3.				

General Reference Material Review

FSAR SECTION	PNPS TECHNICAL SPECS.	CALCULATIONS DESIGN SPECS PROCEDURES	REGULATORY GUIDES STANDARDS CODES
Ch. 3, 4	T.S. 3.5.A	M-269 Rev. 0	10CFR50 App. J
	3.7.A	8.7.1.3 / 8.7.1.5	ASME B+PV XI

B. For the proposed hardware change, identify the failure modes that are likely for the components consistent with FSAR assumptions. For each failure mode, show the consequences to the system, structures or related components. Especially show how the failure(s) affects the assigned safety basis (FSAR Text for each system) or plant safety functions FSAR Chapter 14 and Appendix G).

Prepared by JLRogers Date 6/13/86

NOTE: It is a requirement to include this work sheet with the Safety Evaluation.

Exhibit 3.07-C

Rev. 2

SYSTEM/ STRUCTURE/ COMPONENT	FAILURE MODES	EFFECTS OF FAILURE	COMMENTS
LOSS OF INVENTORY FROM THE Rx. VESSEL	CONTROLLED LEAK DECREASING Rx. VESSEL WATER LEVEL	THE MAGNITUDE OF THE LEAK IS SMALL COMPARED TO THE LEAKAGE EXPECTED IF THE VALVES (MO.1001-28 & 29 B) WERE AT THE APPENDIX J ACCEPTANCE CRITERIA	SEE CALCULATION M-269 Rev. 0
INCREASE IN TORUS WATER LEVEL	INCREASING TORUS WATER LEVEL DUE TO CONTROLLED LEAKAGE	TORUS WATER LEVEL WILL INCREASE AS A RESULT OF THE LEAK ESTABLISHED AROUND THE RHR PUMP DISCH. CK VALVE.	THE WATER ADDITION WILL BE LESS THAN THE WATER ADDED TO THE TORUS UTILIZING THE MO 1001-36A/B VALVES FOR LETTING DOWN THE RHR SYSTEM PRESSURE. TORUS H_2O level will continue to be contained within its limits as required B existing Administra[tive] controls & Technic[al] Specifications
CONTAINMENT INTEGRITY	THE ESTABLISHMENT OF A CONTROLLED LEAK MAY MASK THE DEGREDATION OF CONTAINMENT (IE; LEAKAGE PAST MO 1001-28B OR 29B)	LEAKAGE GREATER THAN THAT ALLOWE[D] BY 10CFR 50 APPENDIX J NOTE: THE 1.0 GPM LEAK IS LESS THAN THE EQUIVALENT AIR LEAKAGE ALLOWE[D] BY APPENDIX J BY A FACTOR OF 3.63.	A LEAKAGE RATE OF ≤ 1.0 GPM WILL ALLOW STABILIZATION OF RHR SYSTEM PRESSURE, YET PERMIT THE OPERATOR TO BE ALLERTED (VIA THE ALARM IN THE CONTROL ROOM) TO AN INCREASE IN LEAKAGE PAST MO 1001-28B OR MO 1001-29B. SEE CALC. M-269 Rev. 0

ATTACHMENT 7

Plan and Schedule Details Regarding

Long Term Actions

Additional Details

Item C (Spurious Isolation) Reference (C), Page 1

According to the most current Long Term Program, the EPIC computer project implementation is scheduled for completion 3/31/87. That completion date is based on a September 1986 refueling outage. Approximately four months after return to power from the outage are required to complete system acceptance tests allowing for contingency.

Refueling Outage 7 is being rescheduled to commence in January, 1987. However, a firm start date and duration have not yet been established. EPIC completion will be scheduled (4) months after return to power from RFO #7.

Item C.6 Trend Surveillance History of 400 psig
Valve interlock for reliability. Reference (B), Attachment (4), Page 6 of 6

The results of Surveillance Test 8.M.2-2.1.8 of Pressure Switches 263-52A and 52B for the RHR injection valve opening permissive have been compiled for the five year period ending in April, 1986. The switch has always actuated at a 100% rate. The incidence where recalibration was needed to restore the setpoint to within Technical Specification limit is 3 occurrences out of 40 (20 tests per switch) or a 92.7% calibration reliability rate. The present calibration frequency is sufficient to assure proper setpoint; therefore, an increase in test frequency is not warranted.

The recommendation of the RHR Task Force Item C.6 has, therefore, been completed by this compilation and analysis.

Special Training Plan for Union and Management Operations Personnel Prior to Station Startup

Prior to the Union Operations Personnel resuming watch standing duties they will receive training as outlined in the following schedule.

In addition, all Management Operations Personnel, including STA's, will receive the following training prior to station startup..

SPECIAL REQUALIFICATION TRAINING SESSION 11A SCHEDULE

TIME:

8:00AM - 8:30AM	Revised Training Schedule	H. Balfour T. Sullivan
8:30AM - 9:00AM	Management Changes/Current Plant Status	P. Mastrangelo
9:00AM - 12:00PM	Plant Modification Update	
	- Complete review of "BECO Response to NRC Cal 86-10" (including all related Temporary Procedures) MSIV RHR Mode Switch	R. Woodard G. Sherman
	- Temporary Modifications 86-14, Change feedwater heater 105B outlet valve, MO3480, from seal-in to jog 86-19, Diesel Generator "A" Relaying Modification	D. Hughes
12:00AM-12:30PM	Lunch	
12:30PM - 4:30PM	Significant Industry Events	J. Klein
	- SER 37-85 - Premature Criticality Due to Control Rods Being Improperly Withdrawn	
	- SER 13-86 - Control Rod Misoperation	J. Klein
	- SER 18-84 - Diesel Generator Differential Relays Non Seismically Qualified	D. Hughes

Miscellaneous Events — J. Klein, R. Woodard, G. Sherman

- Technical Specifications Amendment #94
- Current Memos

Procedure Review

- 1.3.34 Conduct of Operations
- 2.1.1 Startup from Shutdown
- 2.1.16 N.P.O. Tour
- 2.2.22 R.C.I.C.
- 2.2.84 Reactor Recirculation System
- 2.3.2.1 Panel 903 Left
- 2.4.21 Double ended break of 3" instrument air/nitrogen line in drywell
- 2.4.31 Reactor basin/spent fuel pool drain down