APR 15 1982

MEMORANDUM FOR: H. L. Thompson, Jr., Acting Director, Division of Human Factors Safety

FROM: G. R. Mazetis, Chairman, Robinson PTS Task Force

SUBJECT: ROBINSON 2 SHORT TERM TASK FORCE ON PRESSURIZED THERMAL SHOCK (PTS)

Your memorandum dated March 16, 1982 appointed a Task Force to make a detailed review and prepare a report on the status of efforts on PTS at the H. B. Robinson Nuclear Plant. A site visit was arranged on April 5-7th, during which time audits were conducted on procedures and training, specifically with regard to PTS. The Task Force members consisted of the following individuals:

Gerald R. Mazetis - DSI/RSB - Chairman
H. Brent Clayton - DHFS/PTRB
Joseph J. Buzy - DHFS/LQB
Edward Throm - DSI/RSB
Raymond Klecker - DE/MTEB
Roy Woods - DST/GIB (ex-officio)

Our instructions were to provide a report in 30 days (April 15, 1982) characterizing the problem(s), methodology of resolution, bases for conclusions, and recommendations regarding the adequacy of in-place training programs and operating procedures. In addition, you requested that the report attempt to characterize the applicability of this effort to other like facilities and propose review schedules and triteria that can be used in reviewing the other facilities of special concern.

The enclosed evaluation provides the requested report. The site audit of training programs was conducted by Joseph J. Buzy. The site audit of procedures was conducted primarily by H. Brent Clayton. The evaluation of the overcooling history at Robinson 2 was performed by Edward Throm. Ray Klecker and Neil Randall, although not part of the on-site audits, contributed to the fracture mechanics assessment. Roy Woods, although also not part of the site visit, assisted with the report to ensure consistency with other ongoing PTS programs.

As indicated in Section 3.0, "Key Findings from the Robinson Audit," it is clear that the control room emergency procedures remain weighted toward core cooling and do not go far enough in addressing the PTS issue. Pending generic resolution of TMI Action Plan Item I.C.1, such a procedural shortcoming could have been tempered during our audit of plant personnel by a strong awareness and knowledge of the PTS issue; however, the audit produced a varied response from good to poor. The reason for the varied response is due in large part to the need for closer validation by CP&L of operators retention of the material covered in the classroom training sessions on PTS. A more complete discussion of the interviews is presented in Section 3.3, and our recommendations are addressed in Section 5.0.

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Based on your direction in the memorandum dated March 16, 1982, the enclosed report completes the Charter of the Robinson 2 PTS Task Group.

Original signed by:

G. Mazetis, Chairman
Robinson PTS Task Force

Enclosure:
As stated

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ENCLOSURE

NRC STAFF AUDIT OF ROBINSON 2
PROCEDURES AND TRAINING FOR
PRESSURIZED THERMAL SHOCK

Task Force Chairman: G. Mazetis

NRC Staff Audit of Robinson 2 Procedures and Training for Pressurized Thermal Shock

CONTENTS

1 INTRODUCTION

1.1 Short-Term Objectives and Scope of Review

On March 16, 1982, an interdisciplinary Task Force was established to evaluate certain aspects of the Pressurized Thermal Shock (PTS) issue for Robinson 2. The question that the Robinson Task Force focused on was:

> ARE CORRECTIVE ACTIONS REQUIRED THAT MUST BE INITIATED BEFORE THE LONGER TERM PTS PROGRAM PROVIDES GENERIC RESOLUTION AND ACCEPTANCE CRITERIA?

Emergency procedures and operator training were the only areas in which the Robinson Task Force applied the above general question. As noted in the NRR March 9, 1982 presentation to the Commission:

> "...we will undertake a program to verify that existing operating procedures contain the steps necessary to prevent and/or mitigate PTS events, and to verify that operator education/training programs regarding PTS are acceptably thorough."

Initial informal contacts were made with CP&L the week of March 15th and, during a conference call on March 19th, the details of our expected review areas were discussed. Also discussed was a planned visit to the site.

With the limitation of a 30-day response, the scope of review had to be narrowed so that meaningful conclusions and recommendations could be produced. Therefore, resolution to the varied technical questions on PTS (thermal-hydraulic analyses, fracture mechanics, probabilities) was not part of the Task Force charter. Also, implementation of any recommendations (see Section 5) is subject to coordination and consistency with the longer term generic program (USI A-49).

A visit to the Robinson 2 site took place on April 5-7, 1982, during which time the Task Group evaluated procedures and training. The key findings of the group are discussed in Section 3. In preparation for the Robinson 2 evaluation, the Task Force used the general criteria addressed in Section 2.

1.2 Current Status of the Generic PTS Issue

Efforts to pursue an integrated PTS program involving a variety of technical areas are continuing under USI A-49. The summer of 1983 is the current schedule for finalizing our generic regulatory requirements for PTS along with required corrective actions if the generic requirements are not met. Key issues are yet to be resolved and extensive programs exist to provide the foundation for the generic regulatory requirements.

Before the above effort resulting in regulatory requirements is completed, however, we have committed to the Commission to have developed an interim initial position for the summer of 1982 (June). The interim initial position will consist of NRC evaluation of the safety of continued plant operation (and initial corrective actions required) for the eight plants previously identified as representative of plants having the highest RT_{NDT}. Technical assistance is being provided by a PNL multi-disciplinary team. PNL has been contracted to work with the staff to provide recommendations regarding the June 1982 initial position on the safety of continued operation and to recommend any additional corrective actions that PNL believes should be initiated before the NRC generic resolution and acceptance criteria are adopted. The June recommendations by the NRC staff to the Commission will also consider the findings and recommendations addressed in Sections 3 and 5 of this report, as well as other Task Forces formed for related investigations (such as fluence reduction at the vessel wall).

1.3 Robinson 2 Configuration

Robinson 2 is a three-loop Westinghouse PWR rated at 2200 MWt (700 MWe). Normal pressurizer level is controlled by the chemical and volume control system which contains three positive displacement pumps. The safety injection system (SI) utilizes three high head pumps which will initially discharge the boron injection tank (BIT) into the cold legs of the reactor coolant system.

2 SHORT-TERM CRITERIA USED FOR ROBINSON AUDIT

2.1 Transient and Accident Analyses

2.1.1 Introduction

Overcooling events in PWRs may occur as a result of steam line breaks (excessive steam flow), feedwater system malfunctions, or loss-of-coolant accidents. Multiple failures and/or operator errors can result in more severe overcooling events. Of particular concern are those events in which repressurization of the primary system occurs following the severe overcooling. This section addresses an overview of Robinson 2 overcooling events which occurred since the plant was built. Aside from the primary mission of the Task Force to audit procedures and training, also provided (Section 2.1.4) is a summary of the thermal-hydraulic analyses available for evaluating pressurized thermal shock events.

Section 3.1 provides our comments and conclusions on these events and analyses.

2.1.2 H. B. Robinson Overcooling Events Summary

2.1.2.1 Steam Safety Valve Line Break, April 28, 1970

On April 28, 1970, during hot functional testing (no fuel loaded), one of the steam generator safety valve connections failed due to overloading. A 360° circumferential break allowed the safety valve to blow off the main steam line. The plant conditions were:

- 533°F, 2225 psi primary
- 900 psi secondary
- 3 RCPs running
- 45 gpm charging/letdown
- no feedwater to the steam generators

As a result of the 6-in. schedule 80 pipe break, and with no decay heat, the plant cooled down 213°F in 1 hour to a 320°F cold leg temperature. The operator immediately tripped the RCPs (30 seconds) and started the remaining two coolant charging pumps (70 seconds). The minimum primary system pressure was 1880 psi; with the safety injection (SI) setpoint at 1715 psi, no safety injection occurred. The plant was recovered to a normal no-load condition of 2050 psig and charging/letdown reestablished prior to shutdown.

A post-event review of the data indicated that the pressurizer surge line did not empty. A base case analysis was performed for the event. In addition, a sensitivity analysis was performed without RCP trip, with only one charging pump, and with a primary heat source. The analysis showed that the pressurizer would drain and the primary system pressure would fall below the SI setpoint in about 3 minutes. The cooldown was less and the pressures were lower than the base case analysis. It is expected that the operator actions, based on current procedures, would be similar to this sensitivity analysis. The safety valve stand-off piping was redesigned to prevent any similar occurrences.

2.1.2.2 Reactor Coolant Pump (RCP) Seal Failure Event, May 1, 1975

During full-power operation, RCP "C" seal 1 leakage exceeded the technical specification limit of 6 gpm. A load reduction was commenced at a rate of 10% per minute to 36% power and pump "C" was deenergized. Reactor trip occurred due to a turbine trip resulting from the load reduction. The decision was made to restart pump "C" when seal injection could not be restored to pumps "A" and "B." Shortly after restarting the pump, while at 1700 psig and 480°F, seals 2 and 3 failed on pump "C" and the pressurizer level began to decrease.

The following chronology is provided:

2300 - RC system at 1700 psig, 480°F
RCP "C" running

0015 - Stop RCP "C," on high standpipe level alarm
Pressurizer level falling rapidly due to seal 2 and seal 3 failure

0016 - SI pump "A" manually started to supplement charging flow (injection to hot leg)

0018 - SI pumps "B" and "C" manually started, pressurizer level stops falling

0036 - Divert charging flow from "B" loop to auxiliary pressurizer spray to reduce pressure (1150 psig at this time, coolant temperature below 400°F)

0039 - Stop SI pump "C" due to rising pressurizer level

0048 - SI accumulators partially inject prior to isolation (500 psig at this time)

The cooldown for this event was from 450°F to approximately 310°F in one-half hour, with the pressure decreasing from 1700 psig to about 1150 psig over the period of interest. The use of the auxiliary pressurizer spray rapidly reduced the pressure to 500 psig.

The operator used SI to stabilize pressurizer level and pressure while using the main condenser to cool down the plant for RHR entry.

There is no indication that SI was used to repressurize the plant.

2.1.2.3 Stuck Open Steam Generator Relief Valve Event, November 5, 1972

While at nominal full-power operating conditions, the operator was using steam generator relief valves to provide RCS temperature control. One valve would not reclose, resulting in the equivalent of a small steam line break. The secondary side blowdown resulted in a reactor trip and safety injection. The overall cooldown rate was 157°F over a 2-hour period, to 389°F, during the course of the event. Insufficient information is currently available to address operator actions taken during this event.

2.1.3 H. B. Robinson Termination Criteria

2.1.3.1 Reactor Coolant Pumps (RCPs)

The RCPs are tripped when the primary system pressure falls to 1300 psig. In addition, the RCPs are tripped if seal cooling is lost, if excessive seal leakage occurs, or if excessive vibration occurs.

2.1.3.2 Auxiliary Feedwater

Auxiliary feedwater is isolated to the steam generator identified as faulted for steam line breaks or steam generator tube rupture. The flow rate is limited to 400 gpm to any steam generator.

2.1.3.3 SI Termination During LOCA

The termination criteria for safety injection during a LOCA addresses core cooling. No reference to pressurized thermal shock is provided. The termination criteria include a 2000 psig (and increasing) requirement.

2.1.3.4 SI Termination During Steam Line Break

The termination criteria for safety injection during a steam line break are:

- One RCS T_{HOT} less than 460°F,
- RCS pressure greater than 700 psig (stable or increasing),
- Pressurizer level greater than 20% (heaters covered),
- RCS subcooling greater than 40°F, and
- Heat sink available (U-tubes covered).

As shown, one of the criteria for terminating SI during a steam line break is one wide-range T_{HOT} reading less than 460°F, with wide-range primary coolant system pressure greater than 700 psig and stable or increasing. The Westinghouse guideline value is 350°F, T_{HOT}. This value includes all uncertainties and does imply reference to the downcomer temperature.

The uncertainties include core heatup during natural circulation, ECC mixing and instrument errors. Westinghouse has reviewed their fracture data for a wide range of transients and, for the most limiting vessel at end of life, they conclude that the 350°F T_{HOT} would not result in vessel failure. The 700 psig, stable or increasing, pressure assures that a primary side LOCA does not exist coincident with the steam line break. Robinson 2 has increased the 350°F value to 460°F to provide a combined assurance that 40°F subcooling exists at a pressure of 700 psig, concurrent with a sufficiently high temperature to accommodate brittle fracture concerns. Also, it is noted that the Westinghouse 350°F/700 psig values would violate the Robinson 2 NDT limit for 100°F/hr cooldown events.

2.1.4 Thermal-Hydraulic Analyses

2.1.4.1 FSAR Analyses

FSAR analyses assumptions are developed to demonstrate compliance with current NRC regulations concerning fuel design limits, pressure boundary protection (overpressure protection), and radiological releases. These assumptions do not necessarily result in the most severe overcooling. The analyses are typically carried out for only a few minutes and do not provide enough data to perform vessel integrity fracture analyses.

2.1.4.2 WCAP-10019 Vessel Integrity Analyses

The analyses provided in WCAP-10019 are typical of FSAR-type design bases events. However, the boundary conditions have been selected to enhance the overcooling. Maximum safety injection and feedwater flows are assumed, minimum water temperatures are used, and heat sources are either omitted or are conservatively underestimated. Large and small LOCAs have been addressed, as well as large and small steam line breaks. In addition, the Ranch Seco overcooling event was included. Westinghouse indicates that the dynamics of this event would be similar to a low probability small steam line break (including additional failures). Operator action is identified for two events presented in WCAP-10019. For the isolatable LOCA (a stuck open PORV), it is assumed that the operator isolated the break in 30 minutes. For the large steam line break, it is assumed that auxiliary feedwater to the faulted steam generator and makeup injection flow to the RCS is terminated within 10 minutes.

2.1.4.3 Westinghouse Procedural Guideline Analyses

In response to Item I.C.1 of the TMI Action Plan, Westinghouse has performed a series of "best-estimate" analyses to support their current program for operator guidelines and procedure development. These analyses indicate that considerable conservatism exists in the WCAP-10019 vessel integrity analyses.

2.1.4.4 NRC Independent Audit Analyses

Independent audit analyses of a large steam line break have been performed by LANL with the TRAC-PD2 computer programs. These analyses are in agreement with the Westinghouse guideline analyses.

Independent audit analyses are also being performed at INEL with the RELAP5 computer program for small steam line breaks. The results of these analyses will be available at the end of April 1982.

2.2 Criteria for Procedural Reviews

The procedures to be reviewed were selected based on the perceived likelihood of conditions occurring that might subject the reactor vessel to pressurized thermal shock conditions and based on the potential consequences of less likely transients. Such procedures selected included normal heatup and cooldown, steam generator tube rupture, steam line breaks, and loss of coolant accidents.

The audit criteria for the content of procedures was somewhat flexible to account for the operator knowledge interface and to identify which procedures must be used to respond to a certain transient. In addition, detailed operator knowledge of actions for preventing or mitigating PTS could offset some weaknesses in procedures. With this in mind, the following criteria were established for the procedures audit:

(1) Procedures should not instruct operators to take actions that would violate NDT limits.

(2) Procedures should provide guidance on recovering from transient or accident conditions without violating NDT or saturation limits.

(3) Procedures should provide guidance on recovering from PTS conditions.

(4) PTS procedural guidance should have a supporting technical basis.

(5) High pressure injection and charging system operating instructions should reflect a consideration for PTS.

(6) Feedwater and/or auxiliary feedwater operating instructions should reflect PTS concerns.

(7) An NDT curve and saturation curve should be provided in the control room. (Appendix G limits for cooldowns not exceeding 100°F/hr).

2.3 In-Plant Training Program

The effort of the task force to determine the effectiveness of CP&L training in PTS began by developing training criteria which would be used in evaluating the training material, interviewing Robinson 2 shift personnel, and assessing the evaluation CP&L made after completion of the training. The criteria developed into three general areas:

(1) Training should include specific instruction on NDT vessel limits for NORMAL modes of operation.

(2) Training should include specific instruction on NDT vessel limits for transients and accidents.

(3) Training should particularly emphasize those events known to require operator response to mitigate PTS.

More specific criteria were also developed to aid in the review of the training program and in preparation of interviews with operating personnel.

CP&L was requested to furnish an outline of their training program on PTS and the lesson plan which was used in the training classes. They were also questioned on the method used to evaluate the effectiveness of the training sessions.

Preparation for review of the training program included a review of CP&L correspondence with the Commission, including a report on vessel integrity of Westinghouse operating plants (WCAP-10019), normal and emergency procedures furnished by Robinson 2, the Robinson 2 license, technical specifications, and the FSAR. An interview plan was developed which used the general training criteria and the specific subjects which were included in the CP&L training material.

Each interview was preceded by a discussion of the reason for the audit, acknowledgement that the individual could use all material available in the control room, particularly the followup or recovery steps in the emergency procedures, and a request that the individual not inform other operators of the questions asked in the interview. Several interview aids were prepared to provide the operators a point of reference for discussion and to allow them to predict responses or execute recovery strategies to mitigate PTS or challenges to other limits.

3 KEY FINDINGS FROM THE ROBINSON AUDIT

3.1 Transient and Accident Analyses

3.1.1 Introduction

This section presents our comments and conclusions based on the material provided in Section 2.1 of this report.

3.1.2 Robinson 2 Overcooling Events

CP&L reviewed the Robinson 2 operating history and presented three events where the cooldown rate exceeded 100°F per hour. The minimum cold leg temperature measured was approximately 310°F during the cooldown for the reactor coolant pump seal failure event of May 1, 1975. In each case reviewed where operator data was available, the operator actions were different than would be expected with current plant emergency procedures.

For example, for steam line break events, the cooldown transients would be less severe using the current reactor coolant pump trip criteria (continue to run until 1300 psig). Insufficient current procedural guidance exists to evaluate whether the operator would continue to run additional charging pumps during the small steam line break for an extended period. For a given overcooling event, particularly if the pressurizer does not empty, continued use of additional charging pumps could result in rapid repressurization.

For small-break LOCAs, repressurization to 2000 psig may not be advisable following a severe overcooling event. CP&L and Westinghouse believe that repressurization to 2000 psig will not compromise vessel integrity.

3.1.3 Robinson 2 Termination Criteria

3.1.3.1 SI Termination During LOCA

The termination criteria for safety injection during a LOCA are:

- RCS pressure greater than 2000 psig and increasing,
- Pressurizer level at no-load level and responding,
- Heat sink available (U-tubes covered), and
- RCS subcooled at least 40°F.

These criteria are weighted to core cooling concerns, and do not explicitly address the pressurized thermal shock issue. The licensee has indicated that, based on the Westinghouse analyses under review by the staff, no PTS concerns exist during a LOCA.

One of the criteria for termination of SI during a LOCA is that the primary coolant system pressure is 2000 psig and increasing. This value provides the following information:

(1) The break has been isolated, or the SI flow is equal to or greater than the break flow.

(2) Some margin exists to terminate SI before the PORV would be challenged.

(3) Repressurization to 2000 psig further assures a 40°F subcooling margin, including uncertainties.

At the time the emergency procedure was developed, Robinson did not have the subcooling meter installed, and core cooling was the dominating issue. To verify subcooling, and include uncertainties in instrument readings and flow conditions, a primary system pressure of 2000 psig was adopted. (It is noted that the Robinson high head safety injection pump cut-off head is 1500 psia.)

3.1.3.2 SI Termination Criteria During Steam Line Breaks

The termination criteria for safety injection during a steam line break, as presented in Section 2.1.3.4, address the pressurized thermal shock issue by a change to the LOCA criteria discussed in the preceding Section 3.1.3.1. The criteria reduces the pressure at which SI termination is allowed. Therefore, we conclude that these criteria provide a reasonable balance between core cooling and PTS concerns.

3.1.4 Thermal-Hydraulic Analyses

FSAR design bases analyses are not suited to the evaluation of vessel integrity. Insufficient carryout in time exists to perform fracture analyses. The events presented in WCAP-10019 are bounding overcooling events, and are representative of design bases events (single failure). These analyses are suitable for vessel integrity studies. Analyses performed by Westinghouse, using "best-estimate" assumptions, indicate that considerable conservatism exists in the WCAP-10019 calculations. These best-estimate analyses indicate that the cooldown would not be less than 350°F for the steam line break spectrum. While some uncertainties exist with regard to mixing for small-break LOCAs, these loss of RCS inventory events appear to be bounded by the steam line break spectrum.

The NRC independent audit thermal-hydraulic calculations for the large steam line break addressed in Section 2.1.4.4 support the above observation on the Westinghouse analyses. Additional audit calculations to be performed during April are expected to provide further confirmation of the Westinghouse thermal-hydraulic analyses.

3.2 Procedures

3.2.1 Description of the Audit

Our audit included a review of procedures selected as discussed in Section 2.2, discussions with licensee and Westinghouse representatives on the instructions relating to PTS and their bases, and an audit of the control room copy of the procedures to determine their legibility and currency. Our audit included the following Emergency Instructions (EIs), Abnormal Procedures (APs), and General Procedures (GPs):

EI-1	Incident Involving Reactor Coolant System Depressurization
EI-6	Loss of Feedwater
EI-7	Station Blackout Operation
EI-14	Reactor Trip (Part A) Turbine and Generator Trip (Part B)
AP-19	Malfunction of RCS Pressure Control System
AP-24	Loss of Instrument Bus
AP-25	Spurious Safeguards Actuation
GP-2	Heatup (Cold Solid to Hot Subcritical at No-Load T_{AVG})
GP-3B	Reactor Trip Recovery
GP-5	Shutdown (Normal Plant Shutdown From Power Operations to Hot Shutdown Conditions)
GP-5A	Plant Temperature and Pressure Control Using Natural Circulation
GP-6	Cooldown (Plant Cooldown From Hot Shutdown to Cold Shutdown Conditions)

3.2.2 Comparison of Procedures With the Audit Criteria

(1) Procedures should not instruct operators to take actions that would violate NDT limits. The procedures audited generally did not appear to contain instructions which would cause an operator to violate NDT limits; most of the procedures referred to, or included cautions to stay within, the limits of the NDT curves. These curves are consistent with the technical specification heatup and cooldown limits. The only area where the procedural instructions may violate these limits (even though cautions exist) is the safety injection termination criteria and charging pump operating instructions in the loss-of-coolant accident procedures. The termination criteria require RCS pressure greater than 2000 psig and increasing prior to terminating high head safety injection (shutoff head approximately 1500 psig). There are no explicit instructions for pressure control or operation of the charging pumps until a controlled cooldown/depressurization is begun using GP-6. Discussions with Westinghouse representatives indicated that the SI termination criteria are under review as part of the generic procedural guideline development and it is anticipated that they will be changed to a lower pressure, at least for the plants having intermediate head SI pumps like Robinson 2.

(2) Procedures shall provide guidance on recovering from transient or accident conditions without violating NDT or saturation limits. See item (1) above for discussion on NDT limits. The procedure for depressurization events (EI-1) refers the operator to Curve 3.5 and provides instructionto maintain at least 40°F subcooling. If reactor coolant pumps are tripped, the procedure for natural circulation instructs the operator to maintain at least 50°F subcooling. The procedures do not provide a maximum subcooling limit. Curve 3.5 is a pressure-temperature plot showing a saturation curve and a 40°F subcocled curve. The recovery instructions for a secondary coolant rupture instruct the operator to establish steam dump from the "good" steam generators to stablize temperatures when temperature and pressure start to increase following dryout of the faulted steam generator.

(3) Procedures should provide guidance on recovering from PTS conditions. While the procedures provide instructions for maintaining the RCS within conditions allowed by the NDT curves, it is not apparent that the procedures recognize that some transients or accidents may result in PTS conditions at the time that the operator can begin to control plant conditions. There are no explicit instructions to the operator on how to recover from PTS conditions. However, terminating feedwater flow to the faulted steam generator and the SI termination criteria help to limit PTS following a steam line break.

(4) PTS procedural guidance should have a supporting technical basis The procedural guidance is generally consistent with that provided by the Westinghouse Owners' Group emergency procedure guidelines. These guidelines are based on best-estimate analyses of transients. The actions specified in the guidelines which would impact PTS are also consistent with the bounding analyses presented in WCAP-10019. Westinghouse representatives stated that the guidelines are also being reviewed against best-estimate fracture mechanics analyses and that this effort will be completed in May 1982. See Sections 2.1 and 3.1 for a discussion of the safety injection termination criteria.

(5) High pressure injection and charging system operating instructions should reflect a consideration for PTS. The 700 psig SI termination criteria for steam line breaks reflect PTS concerns. The SI termination criteria for loss-of-coolant accidents would allow repressurization to above 2000 psig with a cool vessel. There are no specific instructions for operation of the charging pumps following the depressurization transients.

(6) Feedwater (FW) and/or auxiliary feedwater (AFW) operating instructions should reflect PTS concerns. Instructions are provided in the steam generator tube rupture and the loss-of-coolant accident procedures to terminate FW/AFW flow to the faulted steam generator. These and other procedures provide instructions to maintain steam generator levels in the good steam generators within a defined band.

(7) An NDT curve and a saturation curve should be provided in the control room. These curves are provided in the Curve Book located in the control room and are referenced in the applicable procedures. Each of these curves is on a pressure-temperature plot. Curves 3.3 and 3.4 show the technical specification heatup and cooldown limits. Curve 3.5 shows the saturation curve and a 40°F subcooled curve.

The control room copy of the procedures and curves that we audited was legible and current.

3.3 Training

3.3.1 Introduction

The site audit of CP&L's PTS training program consisted of a review of the lesson plan used for classroom training and personnel interviews with five Senior Operators (two of these SOs were Shift Foremen), and two Shift Technical Advisors.

3.3.2 Comparison of training with the Audit Criteria

(1) Training should include specific instruction on NDT vessel limits for NORMAL modes of operation. All senior operators (SOs) and Shift Technical Advisors (STAs) were aware of NDT vessel limits and the bases for normal plant heatup and cooldown restrictions. The SOs exhibited a good knowledge in the use of plant procedures, control board indications and controls, and vessel limit curves. Recent classroom training had re-emphasized the reason for these limits. Both STAs lacked a familiarity with control board indications and controls.

(2) Training should include specific instructions on NDT vessel limits for transients and accidents. Training was conducted to emphasize concerns of vessel limits during transients and accidents, however, the training was limited to classroom instruction. The training included discussions of the termination criteria for LOCA and steamline break accidents. Four of five SOs and one of the two STAs were familiar with PTS concerns during accidents. One of the STAs had not attended the classroom training.

(3) Training should particularly emphasize those events known to require operator response to mitigate PTS. Classroom training included actions required by the operators to mitigate PTS events; however, no training was conducted in the control room, nor were past events at Robinson 2 reviewed in detail. In addition, training did not include discussions of events in which a steam bubble could develop in the RCS (other than the pressurizer), nor the potential for competing concerns in the steamline break procedure between attempting to control RCS temperature and pressure while not worsening the cooldown.

Three of the five SOs had recent simulator training and recalled that they could adequately control RCS pressure and temperature during a steam line break. The other two SOs did not recall the details of previous steamline break simulator exercises. It was recognized that there was limited instrumentation (wide range pressure recorder) to alert the operator of rates of pressure rise during the steam line break recovery.

3.3.2 Personnel Interviews

The initial interviews with two Senior Operators (SO) indicated an excellent background of vessel pressure/temperature NDT (P/T) limits and basis for curves, in addition to a good knowledge of PTS concerns and how plant conditions could lead to PTS events. They exhibited an excellent knowledge of control room instruments and equipment controls. During the PTS event discussion, which included single- and two-phase flow in addition to a reactor vessel steam bubble, they were able to follow procedures and predict portion of the recovery procedure which would challenge P/T limits.

One of these two SOs was concerned with the operator's ability to anticipat rapid rate of pressure change using meters. He recognized that the wide-ra recorder was the only instrument which could display the past and present transient, and adequately depict any rapid rate of increase. The other operator had recently trained at the Shearon Harris Simulator. He remember the team's concern on core subcooling limits during steam line break (SLB) events and that they could adequately control safety injection (SI) and ra RCS temperature and pressure rise by use of steam dumps. The other SO did recall specific details of the last time he witnessed an SLB at the simula Both were concerned that a bubble in the reactor vessel head could negate control of pressure after termination of SI; however, they believed they c control secondary plant steaming to negate a rapid rate of primary system temperature or pressure increase.

With regard to the interviews with two Shift Technical Advisors (STA), on had attended training in PTS and had a good understanding of reactor vess limits during normal operation. He was also aware of PTS concerns during accidents and events leading to PTS. He had difficulty identifying which temperatures to monitor for PTS (procedures identify T_{HOT} in SI terminat but concluded after discussions that T_{COLD} is more of interest than T_{HOT} did have some problems identifying meters on the console, but knew their general location. He did not consider possibility of steam bubble forma the reactor vessel head and the possibility of two-phase conditions after RC pumps are tripped. He did not know the manual actions required for any reactor trip nor did he find the procedural manual actions to terminate auxiliary feedwater in the affected loop for a steam line break. (Procedure step is not explicit.)

He did not appreciate that two steps (2.9 and 2.12) in the SLB procedures concerning control of RCS temperature and pressure using steam dumps could involve another cooling transient on the vessel, and could compete with the SI termination criteria. Some difficulty locating SI flow and pump controls was demonstrated. He feels that his duty is to warn the Shift Foreman (SF) that he may be violating procedure steps or exceeding limits, but does not believe he is ready to contribute to any discussion of deviations or changes in strategies when conditions do not match procedures. He is in training for an RO license and may apply in January 1983. He did not recall simulator exercises which approached vessel P/T limits nor recall significant Robinson 2 events which may have challenged P/T limits. The other STA did not attend the PTS lectures; however, he has reviewed the Summary Report on Vessel Integrity (WCAP-10019). He indicated a basic understanding of P/T limits; however, he is aware that he needs more knowledge in PTS background and possible events. He had consider_ able difficulty in locating equipment, specific controls, and meters on the control board. He also had difficulty with interpreting the RCS wide-range loop temperature indications, and in determining degrees subcooled or pressure-to-saturation on the saturation curve. He had to ask the licensed operator for SI pump head/flow values and also needed assistance in locating steam dumps and auxiliary feedwater controls. He also had no appreciation of possible competing steps of termination of SI and controlling RCS temperature and pressure increase during an SLB, nor how to control the secondary system to achieve these goals. He did not recall any simulator training on SLBs which could help him in PTS events, nor previous Robinson 2 events that challenged reactor vessel P/T limits.

One SO was interviewed who has not been on shift for almost two months. Although he had received PTS training, he believed that the PTS concerns were an increase in RCS temperature after a decrease in RCS temperature and pressure. He stated that the pressurizer surge line is on the cold leg and had to be led (with some difficulty) to reevaluate his statement. During discussions on the steam line break, he attempted to use the steam generator tube rupture (SGTR) procedure in lieu of the SLB recovery procedure. He took almost 2 minutes to determine his error. He did not appreciate possible competing steps concerning control of RCS temperature and pressure increase coupled with termination criteria for SI. It was obvious that he has not "walked thru" the procedure for some time. In addition, he did not recall specifics of the SLB when he last had simulator training. He did recall two Robinson 2 events (safety valve failure and large leak in an RC pump) that challenged reactor vessel P/T limits. He believes they could have been helpful in reviewing PTS history.

The final two SOs were recently licensed and had received additional simulator training in February 1982. Both were very knowledgable about reactor vessel P/T limits and the PTS issue; however, both stated that the PTS training was conducted after the simulator training. They had worked as a team with other SO candidates and did consider reactor vesselP/T limits in many of their exercises. Although they considered that PTS classroom training was good, they did not receive prepared training material. (They apparently were not aware of the PTS reference material which had been recently placed in the control room.)

Both SOs were exceptionally knowledgeable in predicting SLB responses and aware of possible repressurization with and without steam bubbles in the vessel head. They recognized that the SLB model at the Shearon Harris Simulator may not respond to the same event at Robinson 2.

The Robinson 2 PTS training outline was reviewed prior to the site visit on April 5-7, 1982 and found to be acceptable with the general criteria as well as most of the specific criteria. The CP&L training was conducted over a 2-month period and consisted of six classroom sessions. All licensed personnel were required to attend the training sessions; however, STA attendance was not mandatory. No formal evaluation of the effectiveness of the training was conducted; however, the instructor did question individuals during the classroom sessions.

3.4 Summary

On the positive side, it was clear that operator training, specifically on the PTS issue, had been conducted by CP&L. A general awareness of brittle fracture concerns existed, and some personnel interviewed were very good on procedural walk-thru's and control board knowledge (indications, controls, etc). The procedures used in the control room frequently reference curves of NDT limits, particularly those procedures used for normal heatup or cooldown evolutions. Some accident procedures address the PTS issue, specifically the modified SI termination criteria in EI-1, Appendix B, "Loss of Secondary Coolant."

On the negative side, our audit of seven plant personnel in the control room produced a varied response from very good to poor. Knowledge of the PTS issue, location of key control room indicators and controls, and procedural walk-thru's were particularly weak with three of the seven individuals. With regard to the control room emergency procedures, there is no explicit mention of potential brittle fracture concerns in the LOCA instructions, and a relatively high pressure (2000 psig) remains as one of the four SI termination criteria. We also noted that no emergency procedures addressed strategies on what to do once the operator found himself in a severe PTS condition (specifically, trying to reduce pressure or minimize repressurization). In addition, step 2.9 of EI-1, Appendix B, provides minimal guidance to the operator on using steam dump valves to stabilize temperatures following a steam line or feedwater line break. Excessive dumping of steam could extend the cooldown transient. With regard to the PTS classroom training, STAs were not required to attend the sessions and the absence of CP&L validation of the learning process were large reasons for the variation in PTS knowledge. The previous overcooling history of Robinson 2 provides a particularly valuable training tool which was not emphasized sufficiently.

The existing procedures remain weighted toward core cooling concerns. While calculations performed conservatively to bound PTS concerns (WCAP-10019) have merit (analogous to Appendix K core cooling calculations), the use of only conservative analyses is not necessarily a sound approach in writing operator guidelines. As has been endorsed by the industry since the TMI-2 accident in 1979, more rigorous "better estimate" analyses are needed to supplement and support such procedural guidance. Such an objective (currently underway as part of TMI Action Item I.C.1) is intended to provide a better balance to safety functions needed to migitate the consequences of transients and accidents.

Based on the expectation that current procedural inadequacies will be corrected within approximately one year under TMI Action Item I.C.1 (both from a technical and a human factors standpoint), we conclude that with two exceptions, procedural changes should await completion of this program. Those exceptions are reducing the 2000 psig SI termination criterion, and providing additional guidance for stabilizing temperatures following a steam line or feedwater line break. Also, additional operator training should be conducted prior to restart to address the key procedure weaknesses discussed in Section 3.2 (see Section 5.0, "Recommendations").

4 FRACTURE MECHANICS

4.1 General

Aside from the primary mission to audit procedures and training, the Task Force also included in the following sections a discussion of an overview of fracture mechanics and a summary of Robinson 2 reactor vessel properties. Fracture mechanics analyses and thermal shock experiments have confirmed that relatively shallow pre-existing cracks can initiate, that is they can grow deeper into a cylindrical metal wall if the inner surface of the cylinder is subjected to a thermal shock by rapidly decreasing its temperature to the region of the metals nil-ductility transition temperature or lower. This transition region between ductile to more brittle material is referenced by the RT_{NDT} of the material, which increases in magnitude with neutron irradiation.

In addition to the thermal shock which could occur due to a rapid cooling of the beltline region of a reactor vessel, pressure stresses can also exist if the primary coolant pressure is maintained and/or the system is repressurized after an initial drop in pressure. For vessels with a relatively high RT_{NDT}, a particular cooldown transient is more likely to approach the transition temperature than if the same transient were to occur in a new vessel. Therefore, PTS considerations prescribe that repressurization should be avoided to minimize the potential for jeopardizing vessel integrity. This consideration translates to an overall objective of minimizing the RCS cooldown and subsequent repressurization while still ensuring that the core remains cool.

4.2 Robinson 2 Fracture Mechanics

In the fracture analyses of pressurized thermal shock, the fracture toughness of the material is obtained from curves given in the ASME Code as a function of temperature relative to the reference temperature, RT_{NDT}. It is the sum of two quantities, the initial RT_{NDT} measured according to the rules of the ASME Code, and the ΔRT_{NDT} caused by radiation damage and measured as required by Appendix G, 10 CFR Part 50.

For Robinson 2, the welds are the controlling material now and in the future because they are more sensitive to neutron radiation by virtue of their higher copper content. Although the longitudinal welds have low nickel content (less sensitivity to radiation), both longitudinal and circumferential welds must be considered since pressure stresses and the thermal stresses at deep cracks are higher for flaws in longitudinal welds.

Initial RT_{NDT} values were not measured for Robinson 2 because the vessel was fabricated before the ASME Code rules were in place. For the circumferential welds, there were three Charpy tests at +10°F. From these results, a conservative estimate of 0°F for their initial RT_{NDT} was obtained by using the methods given in SRP 5.3.2. From generic data on similar welds, welds made with Linde 1092 flux, a mean value of -56°F and an upper 2-sigma value of -20°F can be estimated; hence, the latter is used as a best estimate. For the longitudinal welds, there are no records available, except that they were made with ARCOS B-5 weld flux. From a limited amount of information obtained from other plants, the initial RT_{NDT} values were assumed by us to be the same as those for the circumferential welds--0°F for the conservative estimate and -20°F for the best estimate.

The only measurement of copper content for Robinson 2 welds is a value of 0.34% for the surveillance weld, which matched the circumferential weld near the top of the core, but not the weld where fluence was greatest. Consequently, for our prediction of RT_{NDT}, the copper content of the longitudinal welds was estimated to be 0.30% best estimate and 0.35% conservative estimate. For the analysis of the circumferential weld, 0.34% copper was used for the best estimate. For the conservative estimate, the calculated value of shift using 0.34% copper exceeded the upper limit of Regulatory Guide 1.99, Revision 1, which bounds all known surveillance and test data in this fluence region; hence, the Regulatory Guide prediction was followed, as given below. Nickel content was taken to be 0.1% and 0.75%, respectively, for the longitudinal and circumferential welds (best estimate values) and 0.2% and 1.2% for the conservative estimates.

Fluence values for the various weld locations are given in the "150 day" report to D. G. Eisenhut from CP&L dated January 25, 1982 (7.2 EFPY). For the longitudinal weld, the fluence as of December 31, 1981 was estimated to be 1.30×10^{19} n/cm² (E > 1 MeV) at the inside surface of the weld. For the circumferential weld the value was 1.24×10^{19} n/cm² (E > 1 MeV). (The critical weld is below the peak axial fluence location.)

The trend curve used by us to calculate ΔRT_{NDT} was developed from analysis of 136 PWR surveillance data points by G. Guthrie of HEDL. His mean curve formula, which has terms for percent copper, "Cu," nickel, "Ni," and fluence, "f" is:

$$\Delta RT_{NDT} = [-5 + 480\ Cu + 270\ CuNi]\ (f/10^{19})^{0.22}$$

The standard deviation was 22°F. The mean curve was used by us to complete the "best estimates" and the mean plus 2-sigma was calculated for the "conservative estimates."

Substituting the appropriate values in the Guthrie formula, our current values of RT_{NDT} for the Robinson 2 welds are:

	Best Estimate	Conservative Estimate
Longitudinal	140°F	240°F
Circumferential	220°F	290°F

These values were reported by us in a Commission meeting on March 9, 1982 and were compared with the licensee's conservative estimates for the longitudinal and circumferential welds of 183°F and 290°F, respectively.

Current pressure-temperature Appendix G limits being used by Robinson 2 were submitted by letter of January 4, 1977 and were previously accepted by the NRC in a letter dated January 25, 1977. The curves are intended to apply for 20 EFPY, or about 13 EFPY beyond today. A recheck of these limits against the information available today regarding fluence accumulation and RT_{NDT} has confirmed our acceptance of the pressure-temperature limits. (An LER dated January 11, 1982 alerted the NRC to a possible 5°F error in the P/T limits, but resolution of this issue is not expected to change the general conclusion.) These limits do not apply to cooldown rates exceeding 100°F per hour. At that cooling rate, the thermal stresses produce values of K_I-thermal that are only a fraction of K_I-pressure, whereas in more severe (postulated) thermal shock transients the reverse is true.

Since definitive cooldown rate-dependent brittle fracture criteria beyond the Appendix G limits have yet to be decided, it is therefore of interest to minimize any severe RCS cooldown and subsequent repressurization, while still ensuring that the core remains cool. The preceding Section 3 addresses our audit of the operations staff at Robinson to determine their level of awareness of this concern, and the procedural guidance available in the control room. The procedures and training on PTS were evaluated against:

(1) Preventing or minimizing the potential for overcooling events.

(2) During an overcooling event, should one occur, limiting RCS pressure to minimize the probability of crack initiation.

(3) If (1) or (2), above, is not possible (severe, rapid overcooling accident), limiting RCS pressure to minimize the probability of through-wall crack propagation.

The licensee has indicated that for the conservative overcooling scenarios analyzed in WCAP-10019, at least 31 EFPY remain for the Robinson 2 reactor vessel. However, key technical questions on assumptions for these analyses are not yet resolved. An example is when to allow credit for warm pre-stress (WPS) which is dependent on defining the events which create PTS risk. Current experimental information suggests that the beneficial effects of WPS could be precluded after a cooldown and subsequent repressurization later in the transient. As addressed at the March 9 Commission meeting, the above question and uncertainties are being pursued intensively, but final resolution will not be available for the June 1982 reassessment.

Aside from the primary mission of the Robinson 2 Task Force to audit procedures and training, as discussed in previous sections of this report, the Task Force also discussed what parts of these unresolved questions are of most immediate interest for Robinson 2 pending resolution in 1983. While conservative worst-case PTS scenarios are being sought and analyzed, our attention focused on the more probable overcooling scenarios (anticipated operational occurrences). Previous staff evaluation has benchmarked the Rancho Seco 1978 event as historical reference to a severe overcooling scenario. Given that a similar event is postulated at Robinson 2, WCAP-10019 indicates that at least five additional years remain before their defined acceptance criteria for thermal shock transients are exceeded, even without credit for WPS. Ongoing staff fracture mechanics evaluations using conservative Robinson vessel properties support a period of at least one year and, using a best estimate RT_{NDT} (see page 4-3), support the five year value. As indicated in Sections 2.1 and 3.1, recent "better estimate" thermal-hydraulic analyses by Westinghouse to support proposed procedural guidelines indicate that the more likely scenarios (such as a stuck open PORV or steam dump) would be bounded by the analyzed Rancho Seco cooldown and repressurization scenario. These Westinghouse calculations are under review as part of TMI-2 Action Item I.C.1.

5 RECOMMENDATIONS

Based on the summary of findings in Section 3.4, which includes the key procedural and training shortcomings, the Robinson 2 PTS Task Force concludes that additional action by CP&L is warranted, particularly in the training area. The following recommendations are provided.

- Prior to restart, and pending longer term generic resolution of the PTS issue, all Robinson 2 operators and STAs should be retrained in the following areas:

  (1) Review of previous overcooling events at Robinson 2. This includes all available strip charts, event summaries, and review of operator response to mitigate the events.

  (2) Review the emergency and abnormal procedures which challenge core and P/T limits and sketch the typical progress of key parameters until recovery is achieved. This exercise should consider a RCS with and without a steam bubble at locations other than the pressurizer. As a team, each shift should review their sketches and operator response to mitigate the transient. This includes instrumentation and controls during the recovery phase, with a complete walk-thru until conditions stabilize. Emphasis should focus on discussing alternatives for recovering from a PTS condition, and alternatives for minimizing RCS overcooling and subsequent repressurization, while still ensuring that the core remains cool. The shift should provide feedback of any questions or comments arising from these drills to plant management. Resolution to these questions or comments should then follow, with revised procedures and additional training as necessary.

  (3) A CP&L audit of the shift's ability to cope with a PTS event should be made after the above is completed. This includes a short quiz and a drill or demonstration at the console.

- In the longer term, an independent audit of the ability to cope with PTS using the new I.C.1 procedures should be made to verify an acceptable level of training. Also, CP&L should review the Shearon Harris Simulator response for PTS events to verify that the models are reasonable and can demonstrate steam bubble(s) in the reactor coolant system (i.e., vessel head) during forced flow and natural circulation. Identified anolmalies between the simulator and Robsinson 2 responses should be discussed during the training process.

With regard to the current emergency procedures for safety injection termination:

(1) We recommend that prior to restart the SI termination criteria of 2000 psig be modified to lower the pressure at which the operator can secure SI, while still observing adequate subcooling, heat sink, and pressurizer level. Discussions with the licensee and Westinghouse indicate that this value could be the safety injection pump cut-off head, plus uncertainties (about 1600 psig).

(2) We recommend that prior to restart step 2.9 of EI-1, Appendix B, "Detailed Recovery Procedure-Steam Line or Feed Line Rupture," be revised to provide clear instructions for controlling temperature and pressure following dryout of the faulted steam generator. Such instructions should include recognition of the potential for extending the overcooling transient.

(3) In the longer term, we recommend more consideration be given to lowering the RCS pressure SI termination criterion further than (1) above. For example, an acceleration of the schedule for conversion of the subcooling meter to temperature indication would provide a direct subcooling indication. Such an indication, with a safety grade subcooling meter, should reduce the need to accommodate uncertainties with as high a pressure reference in the LOCA SI termination criteria. Criteria similar to the steam line break procedure (suitably weighted for both core cooling and PTS concerns) could then be adopted in the other accident procedures.

6 APPLICABILITY TO REMAINING SEVEN PWRS

The remaining seven PWRs which have been identified as representative of the plants having a relatively high RT_{NDT} are:

Ft. Calhoun (CE)
Oconee (B&W)
San Onofre (W)
Turkey Point (W)
Maine Yankee (CE)
Calvert Cliffs (CE)
TMI (B&W)

Since it is likely that San Onofre and Turkey Point emergency procedures are, like Robinson, based on similar initial Westinghouse guidelines, our procedural conclusions would probably equally apply. Portions of supporting Westinghouse analyses (WCAP-10019) may not apply to San Onofre due to the absence of main steam line isolation valves. This San Onofre design configuration would tend to increase the importance of adequate procedures and training to cope with secondary side breaks. Our findings on the Robinson training program and operations staff audits are plant specific and cannot be directly applied to the Turkey Point and San Onofre plants.

The general procedural and training criteria identified in Section 2 can be applied to each of the plants to be audited. Review of referenced transient and accident analyses is warranted to verify applicability to plant configuration.

Based on the problems disclosed during the Robinson review, it appears necessary to audit six of the remaining seven plants with worst vessel properties prior to the Commission briefing in June (TMI-1 may be excluded as they are not operating). We recommend that a team or teams composed of Pacifi Northwest Laboratory (PNL) personnel audit the procedures and training for San Onofre 1, Ft. Calhoun, Turkey Point, Oconee, Calvert Cliffs, and Maine Yankee. The team(s) should consist of, as a minimum: procedures evaluator, plant operations specialist (preferably an operator licensing examiner), a reactor systems specialist for analysis evaluation, and a fracture mechanics specialist. The team members (as necessary) should visit each site to expedite the audits, to interview operations personnel, and to discuss questions with the licensees. It may not be necessary for all team members (e.g., the fracture mechanics specialist) to visit each site.

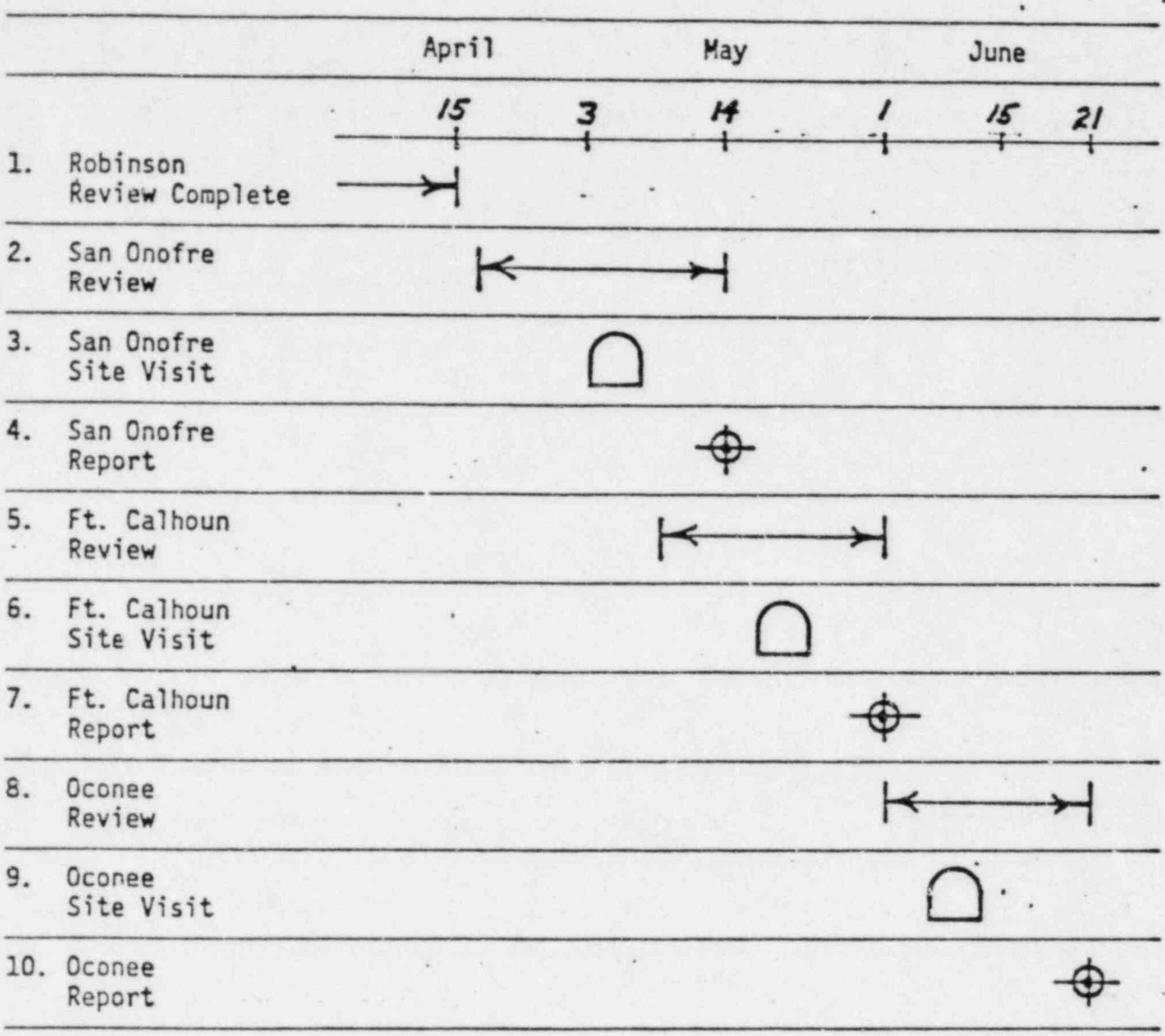
The team(s) will conduct an evaluation of each plant's training program for PTS, and conduct a technical and human engineering review of each plant's procedures used during possible PTS events. These reviews will use criteria developed from the Robinson 2 evaluation conducted April 5-7, 1982.

It is anticipated that the site visits will require 3-5 days each. Therefore, to complete the audits in early June, the site visits should be conducted at a rate of one a week, beginning April 19, 1982. A draft evaluation should be provided at the end of the week following each evaluation. It appears that two or more teams will be needed to meet this schedule. Because of questions raised during the SEP review of San Onofre 1, we recommend that it be the first plant to be audited. The OR project manager for each plant should attend the plant visits to provide liaison between the review team and the plant, since he is most familiar with any particular plant problems and with the Resident Inspector. The OR LPM's role will primarily be to ensure that the necessary documentation and personnel are available at the site, to ensure an efficient evaluation.

The reports will be submitted to the Generic Issues Task Manager, who may, depending on the findings, request additional evaluation by PTRB, LQB, RSB, or MTEB. The final evaluation will be summarized by the Generic Issues Task Manager for presentation to the Commissioners in June.

Should the above multi-team effort not be practical, an alternate option is limiting the site audits to three or four of the remaining six plants, with at least one per vendor complete by June. This would leave Ft. Calhoun, Oconee, and San Onofre as the next three candidates. Assuming a team effort is utilized (PNL), the enclosed schedule outline is proposed.

Prior to further site audits, however, copies of this Robinson 2 report should be made available to the six plants. Inquiry of the licensee should then be made as to whether the key negative findings on training (Section 3.3) at Robinson 2 would apply. A response that similar problems exist should dictate initiation of the training recommendations in Section 5 prior to any site visit. A positive response (no similar problems) would verify that a meaningful site audit could then be conducted.



Summary

-- About 3 weeks each plant (total)

-- 3 day site visit

-- About 1 week writing report