REV.# 1 DATE: 10/15/82
2464A

SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE NUCLEAR GENERATING STATION
NUCLEAR TRAINING DIVISION

COURSE OPERATOR REQUALIFICATION

TOPIC PRESSURIZED THERMAL SHOCK

LENGTH OF LESSON 2 HOURS

OT-1006

INSTRUCTIONAL MATERIAL

Revised By: M. J. Kirby

Reviewed By: P. R. Kuhner

Reviewed and Approved By: M. J. Kirby

REFERENCES

1. Letter from NRC to SCE dated April 20, 1981.

2. Letter from NRC to SCE dated August 21, 1981.

3. Letter from NRC to SCE dated December 18, 1981.

4. Letter from SCE to NRC dated November 4, 1981.

5. Letter from SCE to NRC dated January 26, 1982.

6. Letter from WOG to NRC, OG-58, dated May 14, 1981 with enclosure "An Assessment of Westinghouse PWR Vessel Integrity for Severe Thermal Shock Conditions."

7. Letter from WOG to NRC, OG-56, dated December 30, 1981 with enclosure WCAP-10019 "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants."

8. "An Overview of the Significance of Pressurized Overcooling Transients," EPRI Pressurized Thermal Shock Workshop, Ocotber 8-9, 1981.

9. Meeting Report, NRC/PWR Owners Group Meeting, April 29, 1981.

10. WOG Slides, NRC/PWR Owners Group Meeting, July 29, 1981.

11. NRC Staff Slides, Commission Briefing, September 15, 1981.

12. NRC Staff Slides, Commission Briefing, November 24, 1981.

13. Letter from SCE to NRC dated April 18, 1980 with enclosure "Analysis of Capsule F From the SCE San Onofre Reactor Vessel Surveillance Program."

14. "Brittle Fracture and Reactor Pressure Circuits," Nuclear Engineering, March 1961.

15. General Physics course book "Reactor Plant Materials".

OBJECTIVES

1. To familiarize the operator with the problem and history of Pressurized Thermal Shock (PTS).

2. To review the concepts of Metallurgy as applies to Reactor Pressure Vessels.

3. To discuss the causes of Pressurized Thermal Shock (PTS).

4. To train the operator in procedures and actions available to him to prevent and/or limit consequences of Pressurized Thermal Shock.

INTRODUCTION	INSTRUCTOR/TRAINEE ACTIVITY
A. Establish contact.	Introduce self; write name on board. Write topic on board; hand out attendance sheet, as applicable. Have student fill out. Explain coffee mess.
B. Create interest.	Create interest.
C. Overview	
1. Objectives	Ask questions as to student understanding of objectives.
2. Safety Precautions	

PRESENTATION	INSTRUCTOR/TRAINEE ACTIVITY

I. HISTORY OF PTS

A. Rancho Seco (excess feedwater transient)

1. On March 20, 1978, the Rancho Seco plant RCS was cooled from 582 degrees F. to about 285 degrees F. in slightly more than one hour (approximately 300 F./hr.), while RCS pressure was about 2000 psig. The transient was initiated by an inadvertent short in a DC power supply causing a loss of power to the plant's non-nuclear instrumentation (NNI). Loss of NNI power caused the loss of most control room instrumentation and the generation of erroneous signals to the plant's Integrated Control System (ICS). The ICS reduced main feedwater, causing the reactor to trip on high pressure. The cooldown was initiated when feedwater was readmitted to one steam generator by the ICS (auxiliary feedwater was automatically initiated) and by the operators (main feedwater was restored). The cooldown caused system pressure to drop to the setpoint (1600 psig) for the safety features actuation system which started the high pressure injection pumps and auxiliary feedwater to both steam generators. High pressure injection flow restored pressure to 2000 psig. With control room instrumentation either unavailable, or suspect for one hour and ten minutes (until NNI power was restored), operators continued auxiliary feedwater and main feedwater to the steam generators while maintaining RCS pressure with the high pressure injection pumps.

2. Crystal River 3 (small break LOCA transient). On February 26, 1980, the Crystal River 3 plant experienced a small break LOCA transient when a power operated relief valve (PORV) was inadvertently opened. The resulting transient caused a decrease in RCS temperature of about 90 degrees F. in 30 minutes (approximately 200 F./hr.) with a system pressure of about 2400 psig. The transient was initiated when an electrical short in a DC power supply for the plant's NNI caused a pressurizer PORV to open, a loss of most control room instrumentation, and the generation of erroneous signals to the plant's ICS.

The ICS caused a reduction in feedwater flow and a withdrawal of control rods. RCS pressure initially increased, tripping the reactor on high pressure, and then decreased as coolant discharged through the open PORV. The high pressure injection pumps started at 1500 psig and repressurized the RCS to about 2400 psig. The PORV block valve was closed, but flow out of the RCS continued throgh the pressurizer safety valves. After approximately 30 minutes, the high pressure injection pumps are throttled back, but RCS pressure was maintained at about 2300 psig for the next one and a half hours. The RCS temperature decreased by about 90 degrees F. in the first 30 minutes and was thereafter brought to cold shutdown conditions by normal operating procedures since NNI power had been restored.

3. Borssele (steam line break transient). On March 2, 1981, an inadvertent opening of a main steam safety valve at the Borssele plant (located in the Netherlands) caused one steam generator to boil dry and resulted in a primary system temperature decrease from 446 degrees F. to 284 degrees F. in about 20 minutes with primary system pressure above 2000 psig. Opening of the steam safety valve was caused by a maintenance error when the cable connectors of two solenoid pilot valves controlling the steam safety valve were interchanged. When power to the solenoids was switched on during plant startup, the safety valve opened. Operators disconnected the power supply to the pilot valves, but the safety valve remained open because the piston of one of the pilot valves stuck in its open position. For about 25 minutes, steam was released through a main steam safety valve into the atmosphere causing a blowdown of one steam generator until it boiled dry. The primary system temperature decreased at a rate of about 500 F./hr. for 20 minutes. The primary system pressure dropped from 2234 psig to 2060 psig.

4. San Onofre Unit 1

a. There have been 3 potential PTS events at SONGS 1. A brief description follows -

b. April 30, 1972

On April 30, 1972, with the unit at 55 MWe during startup failure of the "C" feedwater controller resulted in a reactor trip on high "C" steam generator level. Overfilling this generator caused average RCS temperature to fall from 550°F. to about 460°F. in 18 minutes and pressure fell from 2035 to 1550 psig. The event was terminated when the safety injection setpoint of 1685 psig was reached and safety injection initiated. No actual flow from the safety injection system was added to the reactor coolant system. Nine minutes after actuation the safety injection system was secured.

c. October 21, 1973

On October 21, 1973, unit load was being gradually reduced from 450 MWe to perform plant plant maintenance when a turbine trip and resulting reactor trip occurred. At that time, the feedwater regulating system was programmed open the regulating valves on 80% open on any trip. This resulted in a rapid filling of the steam generators and a cooldown of the RCS from 548°F. to 470°F. in about eight (8) minutes. Initiation of Safety Injection at 1685 psig terminated the event. As a result of this event, the feedwater regulating system was reprogrammed to provide 5% flow on a reactor trip, thereby preventing a recurrence of this event.

d. September 3, 1981

On September 3, 1981, with the unit operating at 390 MWe, a failure in the #1 Regulated Power Supply caused several alarms and the loss of several plant parameter indications.

As a result, the operator manually tripped the plant, but feedwater flow continued, resulting in an overfilling of the steam generator. This resulted in RCS temperature falling from 550°F. to 480°F. and pressure falling from 2077 to 1700 psig in about five (5) minutes. Safety Injection which terminated the event was automatically initiated at the new setpoint of 1735 psig. As a result of this event, Westinghouse was consulted about the cooldown of the vessel. They confirmed that vessel shock was not a concern in this event.

B. NRC Concern-TMI Interfered.

1. Political background.

a. NRC dissident wrote letter to Rep. M. Udall (D.-Ariz.) about PTS. Udall harassed NRC which lead to increased NRC concern at commissioner level.

C. NRC Correspondance

1. Generic letter April 20, 1981.

a. Defined area of concern.

b. Define Pressurized Thermal Shock.

1) Reactor Vessel cracking due to repressurizing vessel following a severe overcooling transient.

2) Really just a new term for concerns over vessel integrity. Recent studies show that repressurization is not entirely necessary temperature alone can be sufficient.

3) 3 factors basically affect PTS.

a) Vessel material properties with irradiation effects.

b) Severity of thermal shock.

c) Magnitude of pressure transient during repressurization.

2. August 21, 1981 letter to R. Dietch.

   a. Listed SONGS 1 as one of eight reactors that was of concern to NRC because of embrittlement that could lead to more severe PTS consequences. Requested SCE response in 2 steps - 60 day response and 150 day response.

3. SCE November 4, 1981 - 60 day response. This letter responded to all items except a limiting valve for RT_{NDT} which was delayed pending WOG Generic Guidelines.

4. NRC evaluated 60 day response in letter dated December 18, 1981 which generated additional questions to be answered in final 150 day response.

5. SCE 150 day response final SCE position on PTS.

   "The enclosure provides the results of plant specific analyses that demonstrate the reactor vessel integrity will be maintained for the San Onofre Unit 1 vessel beyond its design lifetime. The analyses indicate that a vessel integrity safety concern does not exist through the end of plant life. Consequently, a plan defining remedial actions and schedules for resolution of this issue, as requested by your August 21, 1981 letter, is not warranted. However, as indicated in the enclosure, we will institute formal training on this issue as part of the operator requalification training program beginning in February, 1982".

6. NRC responded with a series of questions on 150 day response and sent an audit team to evaluate our training and procedures. The audit was conducted June 3-4, 1982.

7. Audit results were forwarded on Sept. 20, 1982 and 5 recommendations were made of a minor nature: 2 training items and 3 minor procedure recommendations.

PRESENTATION	INSTRUCTOR/TRAINEE ACTIVITY
II. METALLUGICAL BACKGROUND FOR PTS	
A. Brittle Fracture	
1. History: 1919 2 x 10^6 gal molassess tank ruptured killing 12 in WWII. Research was conducted because 19 ships literally broke in half. Common factor was temperature low temp. 40°F. in all cases.	Brittle Fracture Overhead.
2. Brittle fracture is catastrophic propagation of a crack through a material. 3 factors affect brittle fracture.	Brittle Fracture Overhead.
a. Defect.	
b. Stress.	
1) Tensile.	
2) Compressive.	
c. Temperature.	
3. Temperature effects on Brittle Fracture.	
a. Nil Ductility Transition Temp: NDTT Temp. above which brittle fracture will not occur. Determined by 2 tests. Drop weight test - NDTT Charpy V Notch test - PUDTT & 60. These 2 combine to yield a most conservative NDTT. The term Reference Temperature for NIL Ductility RT_{NDT} is used instead of NDTT. Same thing only more conservative (i.e., NDTT + 60°).	
4. Radiation effects on materials.	
a. Radiation damage is due to high energy neutrons 1 MEV. Damage is accomplished 2 ways.	
1) Displacing atoms from crystal structure creating vacancies and interstitial atoms.	
2) Thermal spikes crystal deformation.	
b. Neutron fluence.	

PRESENTATION	INSTRUCTOR/TRAINEE ACTIVITY
Integrated neutron dose results in embrittlement of vessel. This makes material stronger but more brittle. This is reflected in RT_{NDT} (i.e., a shift in NDTT).	
c. Determination of RT_{NDT}	
1) Calculated in FSAR.	
2) Used as basis for heatup and cooldown curves.	
3) Actually determined with surveillance capsules removed to date. Capsule F removed Dec. 1978. Most recent predicted valves very close to actual valves from capsule.	PTS-2 Overhead.
4) 60 days response values of RT_{NDT} for SONGS 1.	PTS-2A
d. Neutron Fluence as a function of thickness and azimuthal angle.	PTS-3
1) Azimuthal angle also note variables with 1/4 T and 3/4 T. These are most limiting points (i.e., they generally have more limiting RT_{NDT}).	PTS-4
2) Fluence also varies as a function of distance through the material.	PTS-5
3) Fluence as a function of height of the core.	PTS-6
e. SONGS 1 Vessel Construction	PTS-7
Welds - Vertical 3 courses.	
- Horizontal 3 welds.	
Point our most restrictive weld.	
Weld #7-860A	
Cu & Ni make embrittlement more severe - our welds are known to be low in Ni but Cu is unknown, net result is lower RT_{NDT}.	

PRESENTATION	INSTRUCTOR/TRAINEE ACTIVITY
f. Stress's and stress profiles.	
Total stresses allowed on vessel stresses are either tensile or compressive.	
1) Pressure stress - tensile function of pressure.	
2) Embittlement stress - tensile function of neutron fluence.	
3) Temperature stresses - tensile and compressive. Most severe during transients.	
2 conditions:	
a) Heatup.	PTS-8
3/4 T most limiting stresses are addative also more surface area at 3/4 T greater possibility of a flaw.	
b) Cooldown.	PTS-9
(1) 1/4 T most restrictive.	
(2) Higher fluence.	
(3) Stresses are addative.	
If heatup and cooldown rates were the same the total stress at 3/4 T for heatup would approach total allowable stress closer that the total stress at 1/4 T for cooldown, hence heatup rates for SONGS 1 are more restrictive than cooldown.	

III. PTS EVENTS AND APPLICABILITY TO SONGS

A. What are PTS Events

1. Rancho Seco Event.

a. Overcooling due to overfeeding.

b. Sept. 3, 1981 transient presented to USNRC. PTS-10

c. Not significant transient due to large thermal inertia of steam generators.

2. Large Break LOCA.

a. Double ended guillotine rupture.

b. Rapid depressurization and cooldown.

c. No repressurization takes place.

3. SBLOCA.

a. Isolatable SBLOCA stuck open PORV.

b. Most limiting accident for SONGS 1.

c. Still will not exzceed overpressure condition to cause damage for rest of vessel life.

4. Large Steam Break.

a. Double ended rupture.

b. Repressurize but also reheat so not as severe.

c. Special termination criteria in this event.

5. Small Steam Break.

a. Stuck open safety.

b. Again no problem from PTS point of view.

c. Special termination also applicable here.

6. Results. With minimum operating lige of 18 years. PTS-11

IV. OPERATOR ACTIONS

A. No procedure revisions were identified as being required based on the analysis.

PRESENTATION	INSTRUCTOR/TRAINEE ACTIVITY
B. If operator follows current termination criteria in loss of coolant procedure we will not reach a PTS condition. i.e., there will not be a challenge to the Reactor Pressure Vessel.	
C. So reinforce the termination criteria of SO1-1.2.1 Loss of Coolant.	Distribute handout "An Overview of the Significance of Pressurized Overcooling Transients".
1. Basic termination for LOCA.	
a. Pzr Press 1200#, Pzr Level 50%, 40% subcooling heat sink.	
b. If termination is in accordance with procedures analysis shows that although PTS may have occured but no challenge to reactor vessel.	
2. Termination for SLB.	
a. Below 350°F. PTS concern results in termination @ 700# vs. 2000#.	
	PTS-11
3. Net result no remedial action is required.	
V. ADDITIONAL MATERIAL COVERED	
A. Training in Maintenance Orders and Maintenance Tags in response to INPO Audit.	
B. Discussion of downscale failures of Radiation Monitors.	Distribute handout "An Overview of the Significance of Pressurized Overcooling Transients".
C. Discussion of interpretation to Tech. Spec. Change #58.	
D. Presentation of new Chem. procedure SO1-III-2.1 Steam Generator Chemistry Control.	

82-7
6/1
~~May 24~~, 1982

REACTOR VESSEL PRESSURIZED THERMAL SHOCK

REFERENCES:

UNIT:	RANCHO SECO	CRYSTAL RIVER 3	BORSSELE
DOC NO/LER NO:	50-312/78-001	50-302/80-010	NA
DATE:	3/20/78	2/26/80	3/2/81
NSSS/AE:	B&W/BECHTEL	B&W/GILBERT	KRAFTWERK UNION/KWU-BREDERO

NUCLEAR NOTEPAD Significant Event Reports (SER): 93-81, 5-82

INPO/NSAC SOERs: 81-1, 81-2 & 81-3

SMUD letter to NRC Operations Office, Region IV, March 31, 1978, followup report to Rancho Seco LER 78-001

INPO-1/NSAC-3 Report, March 1980, "Analysis & Evaluation of Crystal River Unit 3 Incident"

NRC IE Bulletin 79-27

Borssele Plant Abnormal Occurrence Report, October 10, 1981

AIF Letter, February 19, 1982, "Reactor Pressure Vessel Shock Concern"

DESCRIPTION:

This SOER presents a review of pressurized water reactor (PWR) events that, due to a rapid cooldown of reactor coolant system (RCS) fluid, produced trends that could lead to a reactor vessel pressurized thermal shock (PTS) condition. It has been recognized that appropriate operator actions can limit the approach to such a condition. Recommendations are made for reviewing and/or modifying operating procedures, training operators to understand and deal effectively with PTS issues, and modifying certain design features and operations practices.

RED	IMMEDIATE ATTENTION
*YELLOW	PROMPT ATTENTION
GREEN	NORMAL ATTENTION

A number of overcooling events have occurred, wherein the RCS fluid temperature decreased, while the RCS pressure remained near normal levels. Other events produced both a temperature and pressure decrease, followed by RCS repressurization at reduced temperature. Both types of events cause conditions to exist that degrade the margin to the reactor pressure vessel (RPV) reference temperature for nil ductility transition (RT_{NDT}). This is the temperature below which, the RPV welds or plate metal become embrittled. If such a temperature was actually attained at the inner surface of the RPV (thermal shock), as the result of a severe overcooling transient, and adequate pressure existed, preexisting flaws may propagate into the vessel wall. Concerns have been raised that such conditions may require extensive plant shutdown for requalification of the RPV or, in the extreme case of a leak caused by a through-wall crack, inhibit core cooling.

Reactor vessel PTS is a safety concern primarily in older plants that have a large degree of radiation induced embrittlement. Newer plants that have lower copper and phosphorus content in the RPV welds or plates are less susceptible to such embrittlement. Work on this issue has been ongoing for many years in the nuclear industry by the utilities, PWR owners groups, the nuclear steam supply system (NSSS) vendors, the Electric Power Research Institute (EPRI), and others, but has received increasing attention as the number of overcooling events and years of vessel operation accumulate.

For a preexisting flaw to propagate into the vessel wall and result in failure of the RPV, several factors must be present simultaneously and to a sufficient degree. These include the following:

- an initial vessel flaw of sufficient size to propagate, at a location where it will be subject to significant temperature reduction
- a region of the vessel that has had a large shift in RT_{NDT} due to a high level of neutron irradiation and a relatively high content of the impurities copper and, to a lesser extent, phosphorous
- a severe overcooling transient while maintaining a high system pressure, or a severe overcooling transient with an initial depressurization followed by a system repressurization

One should note that all of these factors are necessary conditions for a crack to propagate into the vessel wall, but are not necessarily sufficient to cause vessel failure. The interaction of these factors and their degree determine the potential for vessel failure.

The basic mechanisms for rapidly cooling the inner surface of the reactor vessel of a PWR include (1) depressurization of the primary system, (2) injection of cold fluid, and (3) rapid removal of energy through the steam generators. Five general classes of transients encompass one or more of these cooling mechanisms. They are as follows:

- large break loss of coolant accident (LBLOCA)
- small break loss of coolant accident (SBLOCA)
- main steam line break (MSLB)
- excess feedwater transient (EFWT)
- steam generator tube rupture (SGTR)

LBLOCA events produce the greatest thermal shock but also very low system pressures. Due to the size of the break, repressurization is precluded. The potential for a flaw to penetrate through the vessel wall for LBLOCA events is therefore extremely low. SBLOCA events, including stuck-open pressurizer relief and safety valves as well as small breaks in the primary coolant system boundary, cool the system less severely than LBLOCA events, but maintain a higher system pressure. SBLOCA transients may thermally shock the reactor vessel by primary system depressurization and injection of cold fluid from the high pressure safety injection (HPSI) pumps. The severity of the thermal shock is dependent on the degree of mixing that is attained with warmer fluids. Following an initial system depressurization and thermal shock, the HPSI or charging pumps can repressurize the reactor coolant system; the degree of repressurization is dependent on the SBLOCA break size and whether it can be isolated.

For SGTR events that have a leak rate in excess of the normal charging pump capacity, the RCS behavior is similar to that of a SBLOCA. System depressurization leads to injection of cold fluid from the HPSI pumps. HPSI or charging pumps can then repressurize the RCS.

MSLB events, including stuck-open secondary side steam valves, generally produce RCS cooldown rates that lie between those of LBLOCA and SBLOCA events. These transients result from a rapid primary-to-secondary energy removal rate caused by vaporization of fluid on the secondary side of the steam generators. The resulting primary system cooldown and depressurization usually initiates the high pressure safety injection pumps. Pressurized thermal shock of the reactor vessel may result from the rapid removal of energy through the steam generators and repressurization of the primary system due to continued use of either the HPSI or charging pumps. Core decay heat and pressurizer heaters can also cause repressurization of the primary system following a MSLB, if the secondary side heat sink remains isolated. However, core decay heat may also increase the RCS temperature.

EFWT events increase the temperature difference, and therefore the heat transfer rate, between the primary and secondary fluid. These events cool down the primary system while maintaining a relatively high system pressure. They may also initiate the safety injection system. While EFWT events may not be as severe as SBLOCA and MSLB events, they occur somewhat more frequently due to the numerous systems and components that can affect both the main and auxiliary feedwater systems.

To date, no known pressurized thermal shock event has caused preexisting flaws to propagate through a reactor pressure vessel. However, transients have occurred which demonstrate the potential for overcooling at pressure. A partial list of such events follows.

RANCHO SECO (excess feedwater transient)
On March 20, 1978, the Rancho Seco plant RCS was cooled from 582 degrees F to about 285 degrees F in slightly more than one hour (approximately 300 F/hr), while RCS pressure was about 2000 psig. The transient was initiated by an inadvertent short in a DC power supply causing a loss of power to the plant's non-nuclear instrumentation (NNI). Loss of NNI power caused the loss of most control room instrumentation and the generation of erroneous signals to the plant's Integrated Control System (ICS). The ICS reduced main feedwater, causing the reactor to trip on high pressure. The cooldown was initiated when feedwater was readmitted to one steam generator by the ICS (auxiliary feedwater was automatically initiated) and by the operators (main feedwater was restored). The cooldown caused system pressure to drop to the setpoint (1600 psig) for the safety features actuation system which started the high pressure injection pumps and auxiliary feedwater to both steam generators. High pressure injection flow restored pressure to 2000 psig. With control room instrumentation either unavailable, or suspect for one hour and ten minutes (until NNI power was restored), operators continued auxiliary feedwater and main feedwater to the steam generators while maintaining RCS pressure with the high pressure injection pumps.

CRYSTAL RIVER 3 (small break LOCA transient)
On February 26, 1980, the Crystal River 3 plant experienced a small break LOCA transient when a power operated relief valve (PORV) was inadvertently opened. The resulting transient caused a decrease in RCS temperature of about 90 degrees F in 30 minutes (approximately 200 F/hr) with a system pressure of about 2400 psig. The transient was initiated when an electrical short in a DC power supply for the plant's NNI caused a pressurizer PORV to open, a loss of most control room instrumentation, and the generation of erroneous signals to the plant's ICS. The ICS caused a reduction in feedwater flow and a withdrawal of control rods. RCS pressure initially increased, tripping the reactor on high pressure, and then decreased as coolant discharged through the open PORV. The high pressure injection pumps started at 1500 psig and repressurized the RCS to about 2400 psig. The PORV

block valve was closed, but flow out of the RCS continued through the pressurizer safety valves. After approximately 30 minutes, the high pressure injection pumps were throttled back, but RCS pressure was maintained at about 2300 psig for the next one and a half hours. The RCS temperature decreased by about 90 degrees F in the first 30 minutes and was thereafter brought to cold shutdown conditions by normal operating procedures since NNI power had been restored.

BORSSELE (steam line break transient)
On March 2, 1981, an inadvertent opening of a main steam safety valve at the Borssele plant (located in the Netherlands) caused one steam generator to boil dry and resulted in a primary system temperature decrease from 446 degrees F to 284 degrees F in about 20 minutes with primary system pressure above 2000 psig. Opening of the steam safety valve was caused by a maintenance error when the cable connectors of two solenoid pilot valves controlling the steam safety valve were interchanged. When power to the solenoids was switched on during plant startup, the safety valve opened. Operators disconnected the power supply to the pilot valves, but the safety valve remained open because the piston of one of the pilot valves stuck in its open position. For about 25 minutes, steam was released through a main steam safety valve into the atmosphere causing a blowdown of one steam generator until it boiled dry. The primary system temperature decreased at a rate of about 500 F/hr for 20 minutes. The primary system pressure dropped from 2234 psig to 2060 psig.

SIGNIFICANCE:

The safety significance of pressurized thermal shock events is the potential for propagation of a flaw in, and potentially through the reactor vessel wall. Depending on the size and location of the penetration, vessel failure could inhibit core cooling. Even without a degradation in core cooling, vessel repair and re-qualification, following the development of even a partial penetration crack, would require an extensive shutdown of the plant. Furthermore, any event suspected of causing pressurized thermal shock may result in shutdown to demonstrate, by inspection, that no significant RPV flaws had been created.

The severity of a PTS event is dependent upon the temperature and pressure during an overcooling transient and the degree of embrittlement of the reactor vessel. The embrittlement rate of a reactor vessel varies depending upon the fast neutron fluence at the vessel wall and the presence of impurities, principally copper, in the base metal and welds of the vessel. Since design specifications for recent reactor vessels limit the copper content of weld and vessel material, pressurized thermal shock events are generally of greater concern to older PWR vessels which tend to have higher copper content and the greatest irradiation to date.

The severity of pressurized thermal shock events have been determined by the industry through a set of complex calculations that include (1) neutron fluence and linear elastic fracture mechanics calculations to determine the degree of vessel embrittlement and the potential for flaw propagation, and (2) thermal-hydraulic analyses of overcooling and pressurization transients to determine the vessel loading conditions. Conservatisms and uncertainties exist in these analyses that affect the determination of the severity of pressurized thermal shock events. Due to the potential seriousness of these events and the fact that operator actions can either increase or decrease the severity of such an event, consideration of remedial actions to prevent or limit the consequences of these events is appropriate at this time.

RECOMMENDATIONS:

OPERATOR TRAINING

Operator training and requalification training programs should provide course material to ensure that reactor operators are aware of the following:

- for certain RPV's, depending on type and age, overcooling transients at pressure could cause propagation of RPV flaws
- operators can make a significant difference in the severity of events which could lead to pressurized thermal shock by taking actions that will limit overcooling transients and that will limit system repressurization when overcooling transients occur.

1. Licensed operator and shift technical advisor training programs should be reviewed and modified as necessary to provide the trainees with job-related knowledge and skills in the following areas:

   - material science fundamentals necessary to comprehend plant material limitations and industry concerns related to brittle fracture
   - transients or accidents during which RCS coolant conditions will approach RPV thermal shock limits
   - actions for preventing or mitigating the severity of overcooling and repressurization transients
   - plant system capabilities and operational practices related to ensuring adequate core cooling while avoiding pressurized thermal shock.

2. Licensed operator and shift technical advisor retraining programs should provide periodic review of these job-related areas. Relevant plant-specific and industry operating experiences and new developments related to overcooling and repressurization transients should also be covered.

OPERATING PROCEDURES

3. Review and modify operating procedures, as necessary, in order to ensure adequate core cooling while avoiding adverse system repressurization from high pressure safety injection (HPSI) or charging pump flow following overcooling transients such as small break LOCA, steam generator tube rupture, steam line break, and excess feedwater. HPSI pump termination and throttling criteria should be reviewed or developed and incorporated into operating procedures in such a way that the operator has clear instructions on what actions to take, and thus, avoid his having to make an arbitrary choice between cooling the core and avoiding pressurized thermal shock to the RPV. Several utilities currently throttle or terminate HPSI pump flow using criteria based on some combination of RCS pressure, subcooling, pressurizer water level, and steam generator water level. Cautions should be added to appropriate emergency procedures stating that continued operation of the high pressure injection or charging pumps may lead to pressurized thermal shock conditions, and ultimately, can cause the RCS to repressurize to the PORV or safety valve setpoint.

4. Review and modify operating procedures as necessary to ensure adequate heat removal while avoiding overcooling of the RCS by excessive addition of feedwater. Criteria for throttling auxiliary feedwater flow should be developed and cautions against causing excessive cooldown by overfeeding should be included in operating procedures.

DESIGN AND OPERATION

5. In order to reduce the probability of a loss of plant instrumentation and control power, which has the potential for initiating a pressurized thermal shock event, implement the recommendations of INPO SOER 81-1, 81-2, and 81-3, and NRC IE Bulletin No. 79-27, as applicable.

ADDITIONAL CONSIDERATIONS:

1. For those plants scheduled for upcoming inservice inspections, consideration should be given to obtaining additional information regarding vessel material properties and flaw sizes, locations, and orientations. Such data would assist individual utilities in performing plant-specific calculations to realistically assess the severity of pressurized thermal shock concerns for that vessel. Performing these inspections when convenient may preclude costly unscheduled

shutdowns in the future. Also, refilling empty surveillance capsule holders with RPV materials or foils to measure fluence, may prove useful for future analyses.

2. Evaluate increasing the temperature of the high pressure safety injection fluid by raising the refueling water storage tank (borated water storage tank) temperature. The warmer safety injection fluid may aid in reducing the thermal shock at the reactor vessel wall during safety injection flow depending on the degree of mixing which exists. Note: this recommendation does not significantly benefit steam line break transients and may cause an increase in peak clad temperature for LOCA calculations, and an increase in containment pressure and temperature for high energy line breaks inside containment.

3. It may be appropriate for certain plants with U-tube steam generators to lower the water level for hot zero power conditions when steam generator inventory is greatest. A reduction in steam generator inventory reduces the RCS cooldown caused by a main steam line break transient that boils dry a steam generator. This consideration must be evaluated against the impact on control and safety system setpoints.

4. Evaluate the use of low leakage fuel management options that reduce the fast neutron fluence at the reactor vessel wall by utilizing partially burned fuel assemblies on the core periphery. For some cores, this type of fuel management strategy could be economically advantageous as well as providing benefits with respect to reducing the potential for damaging the reactor vessel as the result of pressurized thermal shock events.

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