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July 1, 1982
IPN-82-48

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
NUREG-0737 Items I.A.2.1 and II.B.4
Upgraded SRO and RO Training and
Training for Mitigating Core Damage

Reference: 1. Steven A. Varga (NRC) to Leroy W. Sinclair (PASNY) letter dated May 25, 1982, same as subject item.

Dear Sir:

In response to your request for additional information (Reference 1), regarding NUREG-0737 items I.A.2.1 and II.B.4, the Authority hereby submits the attached information. A copy of our submittal is being sent to Dr. R.T. Liner of Science Applications, Inc. as you requested.

If you or your staff have any further questions, please contact Mr. John Lamberski of my staff.

Very truly yours,

J.P. Bayne
Senior Vice President
Nuclear Generation

cc: attached

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PDR ADOCK 05000286
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A046

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POWER AUTHORITY OF THE STATE

OF NEW YORK

Response to NRC May 25, 1982

Request for Additional Information

Regarding NUREG-0737

Items I.A.2.1 and II.B.4

(Upgraded SRO and RO Training,

and Training for Mitigating

Core Damage)

Indian Point 3 Nuclear Power Plant

Docket No. 50-286

LIST OF ATTACHMENTS

I Response to NRC's May 25, 1982 letter

II Plant Staff Organization

III Heat Transfer, Fluid Flow and Thermodynamics Course Plan

IV Requalification Program - Heat Transfer, Fluid Flow and Thermodynamics Course Outline

V Requalification Program - Mitigating Reactor Core Damage Course Outline

ATTACHMENT I

A. Training Program

Question 1. Are the lectures and quizzes on the subject of accident mitigation given to shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators? If they are, would you please provide the titles of the people who are trained and an organization chart which illustrates their position in the operations chain?

Response: Attachment II provides the titles and positions in the operations chain of those personnel who receive lectures and quizzes on the subject of accident mitigation.

Question 2. U.S. NRC has taken a position that a minimum of 80 contact hours is required for the training subjects which involve heat transfer, fluid flow, thermodynamics and accident mitigation. Do your training and requalification programs meet this requirement? (A contact hour of instruction is a one-hour period in which the course instructor is present or available for instructing or assisting students; lectures, seminars, discussions, problem-solving sessions, and examinations are considered contact periods under this definition.)

Response: The Authority's initial training program in the past has provided more than 150 contact hours for the training of subjects which involve heat transfer, fluid flow, thermodynamics and accident mitigation. After September 1982, the training program will provide more than 250 contact hours of instruction for these subjects.

The requalification program does not include 80 contact hours of training for the subjects in question. The Authority does not agree with establishing the same minimum time requirement for instruction in both the initial training and the requalification program. Requalification is intended to ensure that an already achieved level of knowledge and proficiency is maintained. It is the Authority's opinion that the present requalification program is satisfactory for the training of heat transfer, fluid flow, thermodynamics and accident mitigation.

Question 3. Is there an increased emphasis on reactor and plant transients in the reactor operator and senior reactor operator training program as required by Item A.2.C.3 of enclosure 1 of Denton's March 28, 1980 letter? If there is, does this training deal with both normal transients and abnormal (accident) transients?

Response: Increased emphasis is provided in the initial training program for Reactor Operator and Senior Reactor Operator through forty contact hours of instruction concerning normal and abnormal (accident) transients. In addition, the topic is reenforced and further developed during the systems, simulator, and on-the-job training phases.

Question 4. Does your initial training program for Reactor Operator and Senior Reactor Operator cover the subjects of heat transfer, fluid flow, and thermodynamics? If it does, is the coverage to the level of detail spelled out in enclosure 2 of the Denton letter? Please send a course outline if available.

Response: Heat transfer, fluid flow and thermodynamics are covered to the level of detail spelled out in the referenced letter during the initial training program. A course outline is provided in Attachment III.

Question 5. Does your initial training program for Reactor Operator and Senior Reactor Operator cover the subject of using installed plant systems to control or mitigate an accident in which the core is severely damaged? If it does, is the coverage to the level of detail spelled out in enclosure 3 of Denton's letter? Please send a course outline if available.

Response: The initial training program for Reactor Operator and Senior Reactor Operator does provide the level of detail spelled out in enclosure 3 of the referenced letter. Accident mi-itigation and control is covered in related subjects throughout the initial training program as follows:

a) Incore and Excore Instrumentation

   1) Systems - Nuclear Instrumentation and Incore Instrumentation
   2) Transient Analysis
   3) Fluid Status
   4) Simulator

b) Vital Instrumentation

   1) Systems - RCS, Pressurizer and Steam Generator
   2) Fluid Mechanics
   3) Simulator

c) Primary Chemistry and Gas Generation
   1) Systems - Chemistry
   2) Transient Analysis
   3) Chemistry

d) Radiation Monitoring

   1) Systems - Area and Process Monitors
   2) Health Physics
   3) IP3 Emergency Plan

B. Requalification Program

Question 1. The requalification program for reactor operators (Section 5.5.5.15) has lectures which appear to cover the subjects of heat transfer, fluid flow and thermodynamics as called out in enclosure 1 of Denton's March 28, 1980 letter. Do these lectures in fact cover this material and is the coverage to the level of detail spelled out in enclosure 2 of the Denton letter? Please send a course outline if it is different from that used in the training program.

Response: Lectures given during the requalification program do include the subjects of heat transfer, fluid flow and thermodynamics and are to the level of detail spelled out in enclosure 2 of the referenced letter. A course outline is provided in Attachment IV.

Question 2. The reactor operator and senior reactor operator qualification and requalification programs (Section 5.5.4.16) have lectures which appear address the subject of using installed plant systems to control or mitigate an accident in which the core is severely damaged. Do these lectures address the topic and do they cover the subject to the level of detail spelled out in enclosure 3 of Denton's letter? Please send a course outline if it is different from that used in the training program.

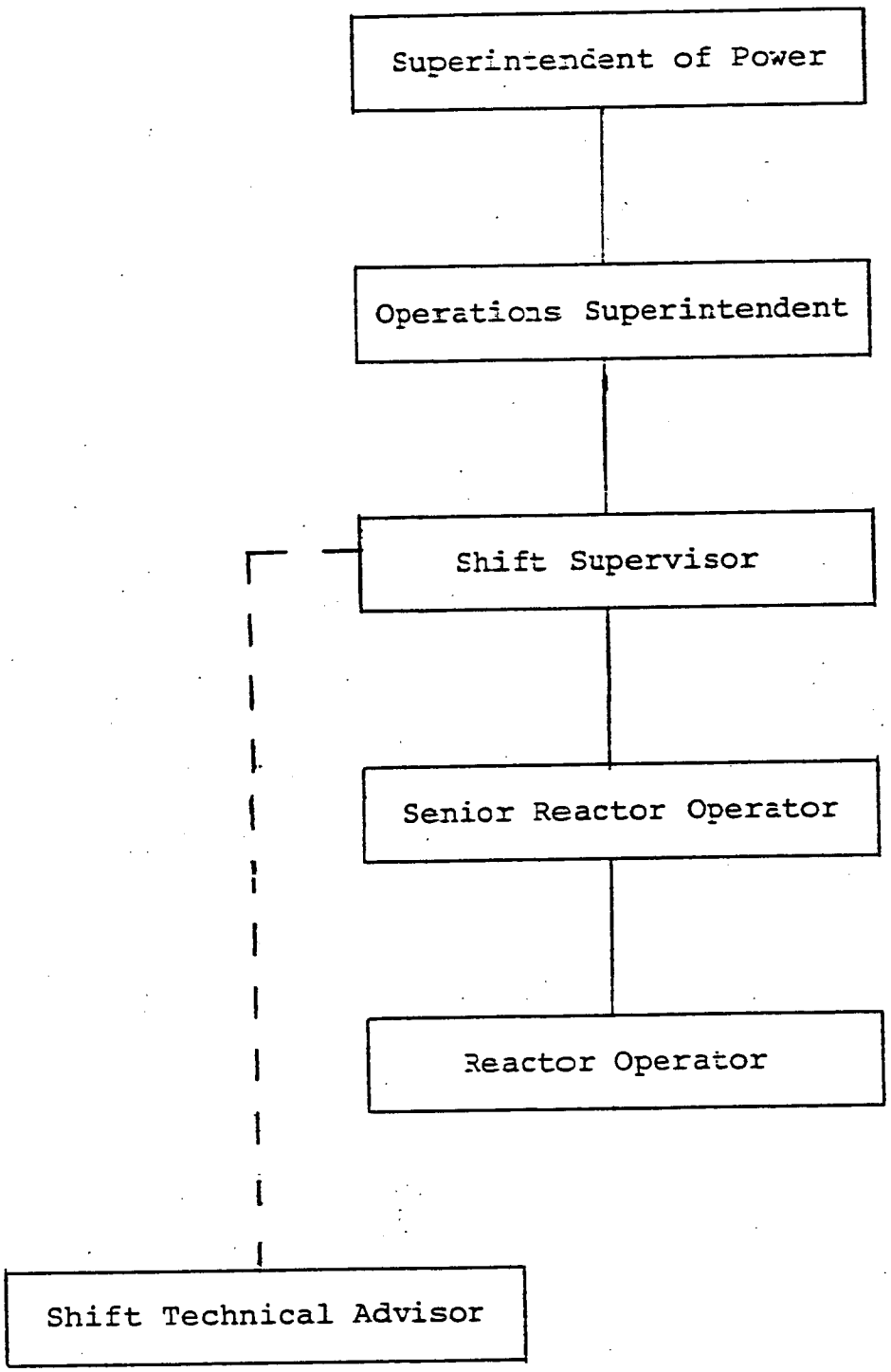
Response: During the requalification programs separate lectures dealing with the use of installed plant system to control or mitigate an accident are used. The content of these lectures provides the level of detail spelled out in enclosure 3 of the referenced letter. A course outline is provided in Attachment V.

Question 3. Please answer question 1 of section A with respect to your requalification program.

Response: Mitigating core damage lectures and quizzes have been given to all licensed personnel during the course of the requalification program. Shift technical advisors receive similator training and review transient analyses and industry-wide events. Core mitigation training is included in the simulator phase of this training. This requalification training program is consistent with INPO recommendations. Refer to the organization chart provided in response A.1.

Question 4. Please answer question 2 of Section A with respect to your requalification program.

Response: Refer to response A.2


Superintendent of Power
Operations Superintendent
Shift Supervisor
Senior Reactor Operator
Reactor Operator
Shift Technical Advisor

ATTACHMENT III

Program ID _____
June 5, 1981/DRC

TOPIC: THERMODYNAMICS

LESSON: THERMODYNAMICS, HEAT TRANSFER, AND FLUID MECHANICS

LESSON OVERVIEW

This lesson provides for a three week, advanced level, program on Heat Transfer, Fluid Flow and Theromodynamic. It is designed to meet the intent of H. Denton's NRC letter of March 28, 1980, for upgrading of the License Training Programs.

RESOURCES AND REFERENCES

1. "Elements of Applied Thermodynamics" - Johnston - Brocket Bock Keating - Fourth Edition.
2. "Nuclear Power Reactor Safety" - E.E. Lewis
3. "Nuclear Power Engineering" - E. L. Wakil
4. "Introduction to Power Plant Technology"
5. Keenan and Keyes - Steam Tables

TRAINING AIDS

1. Chalk Board
2. Transparencies
3. Hot Lesson Plan Text

OBJECTIVES

1. The student will be able to define and explain the thermodynamic properties of substances, the laws of thermodynamics, and the ideal gas laws; use the steam tables and mollier diagrams to determine properties of substances and the conversion of heat energy into mechanical energy; use T-S&P-V diagrams to describe the application of the general energy equation to a basic rankine cycle; perform a heat balance on a steam - turbine cycle.

2. The student will be able to define and explain the Fluid Mechanical Properties, the theory and use of pressure, level and flow measuring devices; use the laws of fluid statics to solve hydraulic problems; solve ideal fluid flow problems using the continuity equation; use reynolds number to describe actual fluid flow phenomena; discuss in detail the establishment, maintenance and calculation of natural circulation.

3. The student will be able to solve basic heat exchanger problems; describe the modes of heat transfer; reproduce the boiling water curve and discuss in detail the various boiling regimes; define DNBR and describe the effects on DNBR of all process variables; define each hot channel factor and how each is determined.

THERMODYNAMICS
WEEK 1

Day 1

I. INTRODUCTION

A. Objectives

B. Power Plant Applications

C. Basic Thermodynamics

D. License Exam Section

II. DEFINITIONS

A. Thermodynamics

B. Energy

C. Work

D. Power

E. System

1. Boundaries

2. Definition

3. Working Fluid

4. Application

5. Closed

6. Open

7. Properties

8. State

9. Phase

10. Relation of Phase to State

11. Process

F. Steady State

Day 1 (continued)

III. UNITS AND CONVERSIONS

A. Conversion Relationship

B. Pressure

C. Temperature

D. Weight and Volume

E. Specific Volume, Specific Weight and Density

F. Specific Gravity

IV. ORAL QUIZ

THERMODYNAMICS
WEEK 1

Day 2

I. ENERGY

A. Stored Energy

1. Internal Energy

2. Kenetic Energy

3. Potential Energy

4. Flow Energy

B. Transient Energy

1. Mechanical Work

2. Heat

a. Temperature

b. Sensible Heat

c. Specific Heat Capacity

C. Summerize Energies

1. First Law of Thermodynamics

2. Stored vs. Transient Energy

3. Comparison of Heat and Work

D. Steady Flow Energy Equation

1. Continuity Equation

2. Elementary Steady Flow Process

a. General Energy Equation

b. Nozzles

c. Turbine

d. Pump

e. Heat Exchanger

II. ORAL QUIZ

Day 3

I. STEAM AND VAPOR PROPERTIES

A. Introduction to Steam Tables

B. Saturation Temperature and Pressure

1. Table I

2. Table II

C. Specfic Volume

D. Quality

E. % Moisture

F. Wet Steam

G. Enthalpy

H. Enthalpy of High Pressure Water

I. Entropy

J. Internal Energy

K. Super Heated Steam Tables - Table III

L. Compressed Liquid Tables - Table IV

M. T-S Diagram

N. Definitions

O. P-V Diagram

P. Mollier Diagram

1. Isenthalpic Processes

2. Isentropic Processes

Q. Gas Relationships

1. Boyles Law

2. Charles Law

3. Combined Gas Law

4. Ideal Gas Law

R. Problem Set

II. ORAL QUIZ

THERMODYNAMICS
Week 1

Day 4

I. WORK AND HEAT

A. General Energy Equation

B. Work and System Boundary

C. Heat and Heat Units

D. Heat of Fusion and Heat of Vaporization

E. Specific Heat

II. FIRST LAW OF THERMODYNAMICS

A. Flow Energy

B. General Energy Equation

C. Application of General Energy Equation

1. Steam Generators

2. Steam Turbine

3. Feed Pump, Condensate Pump

4. Friction

III. SECOND LAW OF THERMODYNAMICS

A. Thermodynamics of a Basic Cycle

B. P-V Diagram

C. Thermal Efficiency

D. Entropy

E. Cycle Diagrams

IV. ORAL QUIZ

THERMODYNAMICS
Week 1

Day 5

I. REVIEW

II. EXAMINATION

II. WEEK 2 (THERMODYNAMICS AND FLUID MECHANICS)

A. Day 1

1. Basic Fluid Flow Relationships
   a. Units of Fluid Flow

2. Laminar and Turbulent Flow

3. Continuity of Flow

4. Bernoulli's Equation

5. Fluid Friction
   a. Evaluating Head Loss Due to Fluid Friction
   b. Evaluating Head Added by Pumps

6. Measuring Fluid Flow Rate
   a. Manometer
   b. Pilot Tube
   c. Venturi Meter
   d. Orifice Meter

7. Two Phase Flow

THERMODYNAMICS AND FLUID MECHANICS

Day 2

I. PUMPS

A. Types

1. Centrifical

a. NPSH
b. Cavitation
c. Pump Runout
d. Pump Laws
e. Pump Efficiency

2. Positive Displacement

3. Jet Pump

II. VALVES

A. Introduction

B. Classifications

C. Uses of Simple Valves

D. Head-Loss Relationship to Valves

E. Valve Selection

F. Valve Actuators

C. DAY 3 (THERMODYNAMICS AND FLUID MECHANICS)

1. Fluid Statics

A. Pressure
   1. Definition
   2. Absolute Pressure
   3. Atmospheric Pressure
   4. Pressure Head
      a. Concept
      b. Pascal's Law

B. Pressure Measuring Devices
   1. Manometers
   2. Barometer
   3. Bordon Tube

C. Level Measuring Devices
   1. Diffen~~dal~~ TIAL Press
      a. Ref. Leg
      b. Var. Leg
      c. Failure Indication
      d. Hostile Environment

D. Problem Set

THERMODYNAMICS AND FLUID MECHANICS

Week 2

Day 4

1. Thermodynamics Cycles

   A. Turbine Work & Power
   B. Condenser Heat Removal
   C. Pump Work
   D. Reactor Heat Input
   E. Thermal Efficiency
   F. Feedwater Heating
   G. Cycle Analysis
   H. Effects of Feedwater Heaters

THERMODYNAMICS, FLUID MECHANICS AND HEAT TRANSFER

Week 3

Day 1

1. Heat Transfer

    A. Heat Transfer

    B. Introduction

    C. Connection

    D. Radiation

2. Heat Transfer in Heat Exchangers

3. Performance of Power Plant Heat Exchangers

THERMODYNAMICS, FLUID MECHANICS AND HEAT TRANSFER

Week 3

Day 2

1. Calculations of Reactor Power

   A. Typical Heat Balance

2. Boiling Heat Transfer

   A. Convection Heat Transfer

   B. Natural Circulation

      Establishment

      Maintainence

   C. Nucleate Boiling

   D. Departure from Nucleate Building

   E. Film Boiling

   F. Boiling Water Curve

   G. Review

      1. DNB-DNBR

      2. Boiling Regimes

   H. Reactor Thermal Power Safety Limits

C. DAY 3 (THERMODYNAMICS, FLUID MECHANICS, AND HEAT TRANSFER)

I. Hot Channel Factors

A. Design Criteria

B. $F_{\Delta h}$ - Enthalpy Rise HCF

C. $F_Q(Z)$ Height Dependent Heat Flux HCF

D. DAY 4 (THERMODYNAMICS, FLUID MECHANICS, AND HEAT TRANSFER)

I. Summary and Review

A. Equations

$$\dot{Q}_{RX} = UA(\Delta T)(T_{FUEL} - T_{AVE})$$

$$\dot{Q}_{RX} = \dot{M}c(\Delta T)(T_H - T_C)$$

$$\dot{Q}_{SG} = \dot{M}c(\Delta T)(T_H - T_C)$$

$$\dot{Q}_{SG} = UA(\Delta T)(T_{AVE} - T_{STM})$$

$$\dot{Q}_{SG} = \dot{M}(\Delta h)(h_{STM} - h_{FEED})$$

$$\dot{W}_{TURB} = (h_{STM} - h_{EXH})\dot{M}_{STM}$$

$$\dot{Q}_{COND} = \dot{M}(\Delta h)(h_{EXH} - h_{CONDENSATE})$$

$$\dot{Q}_{COND} = U_{COND} A_{COND} (T_{COND} - T_{AVE\ RAW\ WATER})$$

$$\dot{Q}_{RAW\ WATER} = \dot{M}_{RAW\ WATER}\, C(\Delta T)(T_{OUT} - T_{in})$$

$$\dot{W}_{PUMPS} = \dot{M}_{FW}\, \nu (P_{S/G} - P_{COND})$$

$$\dot{Q}_{RX} = \dot{W}_{TURB} + \dot{Q}_{COND}$$

(Neglecting Pump Power)

B. Question and Answer Seminar

E. DAY 5 (THERMODYNAMICS, FLUID MECHANICS, AND HEAT TRANSFER)

I. Examination

ATTACHMENT IV

REQUALIFICATION PROGRAM - HEAT TRANSFER, FLUID FLOW AND THERMODYNAMICS COURSE OUTLINE

1. Properties of Working Fluids.
    - A. Units

II. Forms of Energy.
    - A. Define Energy
    - B. Define Heat
    - C. Sign Convention
    - D. Energy and Power Equivalents
    - E. Define Power

III. First Law of Thermodynamics
    - A. General Energy Equation
    - B. Define Enthalpy
    - C. Energy balance equations for power plant machinery.
    - D. Mass flow.
    - E. Examples.

IV. Second Law of Thermodynamics
    - A. Define Entropy.

V. Phases of matter
    - A. Define Phases
    - B. Energy relationships to phase change.
    - C. Steam quality and moisture content.
    - D. Phase Diagrams.
        1. P-V diagrams
        2. P-h diagrams
        3. P-T diagrams
        4. T-s diagrams

VI. Steam Tables

A. Saturated Steam Tables

B. Superheated Stead Tables

C. Use of Steam Tables

D. Mollier Diagram

E. Example Problems

ATTACHMENT V

MITIGATING REACTOR CORE DAMAGE
3/81

1.0 TITLE AND SCOPE

1.1 Title: TMI Transient Indications

1.2 Scope: This lesson will cover the major areas in the TMI Transient where problem arose, and how these problems can be prevented at IP-3.

2.0 PROBLEMS PRIOR TO THE TRANSIENT

2.1 Reactor Coolant Leak

1. Safety or PORV leaking
2. Caused ambiguous temperature indication in drain piping -- later masked the stuck open relief.

2.2 Auxiliary Feedwater Isolation Valves Closed

1. Prevented Auxiliary Feed from feeding the S/Gs after the loss of main feedwater.
2. Valve position indicator lights were covered by a caution tag.

2.3 Blocked Resin Transfer Line

1. Ultimately caused trip of condensate pump which started the transient.

3.0 TMI TRANSIENT

3.1 Main feed pumps trip

1. Main turbine trips
2. Auxiliary feed pumps start
3. Control rods inserted to runback reactor power
4. RCS pressure increases to relief setpoint, reactor scrams

3.2 Auxiliary Feed was isolated -- indicators covered with caution tag -- eight (8) minutes to determine this.

3.3 Pressurizer level "turns" and RCS pressure is decreasing.

1. Rate of Lp increase was high for the type of scram from operating experience.

2. NRC believes this indicates a reduction of coolant density (distributed voids).

3. Lp increasing with RCS pressure decreasing was unexpected -- not normal. (Voids in loop and/or vessel.)

3.4 Inability of pressurizer to maintain RCS pressure was believed to be caused by:

1. Pressurizer energy was diluted by relatively cool Th insurge.

2. Electromatic relief opened -- energy escaping

3. Pressurizer hot water boiling off due to pressure decreasing.

4. Heaters tripped.

3.5 Drain tank temperature & pressure increase -- relief stuck open.

1. Caused by discharge of coolant through relief.

2. Coolant continued to discharge (relief stuck open). Operators thought the drain tank temperature & pressure rise was due to the relief lifting and, when pressure returned to normal, assumed the relief valve closed.

NOTE: Valve position indication showed that the electromatic relief had closed, but it was only solenoid indication -- the valve was open.

3. Stuck relief caused loss of coolant for two hours.

a) Believed to cause core damage.

b) Closing the block valve would have stopped the loss of coolant, if it was recognized that the relief was stuck open.

c) Pressure increase in the drain tank continued after relief should have closed -- one indication that the relief was stuck open.

d) Drain tank relief lifted, and rupture disk "blew" -- further indication of trouble. Spewed coolant into containment sump -- pumped into auxiliary building. Tanks in auxiliary building overflowed.

e) Monitoring drain tank pressure after the relief valve opened -- and supposedly closed -- would have indicated that the relief was open. If block valve was shut, core damage and radioactive release through drain tank disk would have been prevented.

3.6 High Pressure Injection Initiates

1. Operator takes manual control.

2. Due to Lp rapid increase, the operator throttled back on HPI flow. (Prevent solid pressurizer)

   -- Lp increase was not due to excessive makeup, it was due to voiding. The HPI flow was needed.

3. Shutdown "C" HPI pump, "A" pump is running and "B" had previously tripped -- M/U valves closed.

4. Operator prepares for putting a second let-down cooler in service to allow for increased let-down flow to halt the pressurizer level increase.

5. Three hours, forty minutes later, high pressure injection was returned to service. Pumps came on due to high containment pressure. Operators turned off the pumps again. HPI was on and off until 4 1/2 hours into the accident -- then it was allowed to run at full flow -- too late.

3.7 Reactor Coolant Pump Shutdown

1. At 74 minutes, the operators shut down two reactor coolant pumps.

2. At 101 minutes, the operators shut down last two reactor coolant pumps.

3. The RCPs were circulating a steam/water mixture; this provides some cooling. After they are shut down though, the core becomes uncovered.

4. Reason pumps shutdown:

   -- Low pressure, low flow, high vibration (could feed vibration in control room)

5. Policy is to shutdown a reactor coolant pump if there is no indication of a loss of coolant -- to protect pumps -- controversial point.

6. Operators thought that natural circulation would keep reactor core cool; actually voids stopped natural circulation flow.

3.8 Reactor temperature rise

1. Two and one-half (2 1/2) hours -- temperature rise rapidly as core uncovers.

2. Four-five (4-5) hours -- digital voltmeter shows 2500°F core temps.

NOTE: Control Room wasn't informed of this.

3. Twelve and one-half (12 1/2) hours -- temperature indicators go off-scale 620°F on some core thermocouples.

3.9 Steam Environment in Core

1. Chemical reaction between zircaloy clad and steam released large amounts of H2.

2. H2 formed pockets in loops.

NOTE: H2 has been released to the coolant as early as one (1) minute into the accident. This H2 was released to the containment. Eventually ignited and caused a pressure spike.

3.10 Re-establishing Control of Core Conditions

1. Two hours, 18 minutes relief block valve closed, terminating the loss of coolant accident.

2. Attempts were made to establish stable conditions, but the hydrogen and steam coupled with the lack of coolant prevented cooling.

3. Thirteen (13) hours into accident, a coolant pump was started. This provided forced circulation and started core cooling. Fifteen hours, 50 minutes, a second pump was started. Eventually, the core cooled down.

4.0 SUMMARY OF SIGNIFICANT ERRORS

4.1 Failure to recognize that auxiliary feed flow was not going to the S/Gs.

1. IP-3 has AFW flow indication -- TMI didn't.

2. Covering valve position indication.

3. Leaving valves closed inadvertantly.

4.2 Failure to recognize that the relief was stuck open.

1. Monitor drain tank temperature and pressure for indication of discharge of coolant.

4.3 Throttling HPI flow

1. Lp was increasing too rapidly to be an indication of the reactor coolant volume. Also, pressure was falling while Lp increasing, another indication that the Lp increasing was not due to coolant expansion.

2. Didn't restart HPI pump in first 100 minutes.

4.4 Stopped RCS flow

1. Didn't recognize that the flow and pressure dropping along with high vibration indicated steam in coolant.

1.0 TITLE AND SCOPE

1.1 Title: Recognizing Core Damage

1.2 Scope: This lesson will describe the methods used in determining that core damage has taken place.

2.0 INSTRUMENTATION

2.1 Core Thermocouples

1. Description

   a) 65 chromel-alumel thermocouples

   b) Supported by guide tubes which maintain position above the core.

2. Components

   a) T/C
      -- Chromel-alumel

   b) Reference junction boxes
      -- Located in seal table room
      -- Insulated with heaters inside to maintain 160°F
      -- Will hold relatively constant temperature independent of containment temperature.

   c) Temperature Indicator
      -- Two scales 0-400°F and 400-700°F
      -- Only one T/C can be read at a time

3. Operations

   a) Prodac (See Enclosure #1)
      -- Provides Thermocouple Map (long and short form)
      -- Range of indication 0-999°F

   b) Direct Readout (See Enclosure #2)
      -- DVM connected at the Prodac Interface. Use T/C table to convert mV to °F.

4. Use During Accident

   a) High temperatures can be used to locate areas where coolant flow channels are blocked.

   b) Large areas with high temperatures can indicate lack of heat removal such as coolant turning to steam.

5. Other PRODAC data:

a) Post Trip Review (See Enclosure #3).

b) Sequence of Events Record (See Enclosure #4).

c) Trend Blocks (digital & visual) (See Enclosure #5).

2.2 Movable Incore Neutron Flux Detectors

1. <u>Description</u>

a) Six (6) detectors, drive units and transfer devices.

b) Detectors are inserted into 50 guide thimbles.

c) Uses a U_3O_8 fission chambers which can see γ's or $_0n$'s.

2. <u>Operation</u>

a) Shutdown γ levels are $\sim$ 100,000 X less than operation.

b) Use Keithly Picoammeter and Battery Power Supply for shutdown readings.
-- Readings are ambiguous and will not help operator -- for use in later determining what happened.
-- Operating procedure for Keithly Picoammeter (See Enclosure #6).

2.3 Gross Failed Fuel Detector System

1. <u>Description</u>

a. Two (2) detectors
-- Neutron detector -- BF_3 proportional counter
-- Gamma detector -- GM tube

b. Samples of reactor coolant are routed past detectors.

2. <u>Operation</u>

a. Indicates the release of fuel particles into the coolant after the cladding has failed.

2.4 Radiation Monitoring

1. Containment Radiation Monitors (R11, R12 and Iodine)

   a) Isolate on an SI -- not available

   b) If attempt is made to use detector, the sample pump will trip.

2. ARM's

   a) Locations

      -- R2, Inside 80' airlock
      -- R7, seal table room, ~60' elev.
      -- R10, steam line penetration area
      -- R13, in fan house
      -- R14, in vent pipe

   b) Response during accident:

ACCIDENT	DETECTOR & INDICATION
LOCA (with no core damage)	R13 & R14 will increase only if ECIC doesn't hold. R2 & R7, R10 spike the rapidly decrease
LOCA (with Clad failure)	R13 & R14 will go off scale if releasing R2 & R7 will indicate high R10 ~ 65 mR/hr.
LOCA (with fuel meltdown)	R2 & R7 off scale R10 15 R/hr. R14 off scale due to shine
LOCA (with meltdown and loss of Ctmt integrity)	R10 > 1000 R/hr. (off scale)

2.5 Sources of Radionuclides (See Enclosure #7)

1. Primary Coolant Activation Products

   a) Due to makeup inpurities or activated corrosion products (i.e. CRUD)

   NOTE: See Table 1, pg. 5-5.

2. Normal Operations Fission Products

   a) Due to:

      1. Tramp Uranium

      2. Diffusion of noble gases through the clad.

      3. Design fuel element defects.

   NOTE: Table on pg. 5-8 & 5-9.

3. Fission Product Release During Accidents

   a) Gap Release -- release of fission gases that occupy the void space in fuel rods.
      -- Occurs when fuel cladding experiences initial rupture.
      -- Rapid depressurization of RCS provide motive force for escape.
      -- Coolant samples should contain the isotopes listed in Tables 2 & 3.

   b) Fuel Pellet Decomposition with Subsequent Release to Coolant.
      -- Fuel material is transported by coolant following initial clad damage.
      -- Primary chemistry should indicate prescents of transuranic elements as indicated on Table 4 (Cs, Sr, I).

2.6 Post Accident Critical Parameter Instrumentation.

1. New Instrument racks:

   a) V.C. Pressure

      1. Recorder will indicate trends and record hydrogen "burn".

b) Hydrogen Concentration

1. Will indicate adequate core flooding (Zirc - water reaction).

2. Anticipate explosive atmosphere.

c) Sump levels (Recirculation and Containment)

1. Adequate level for recirculation.

2. Back up existing "sugar cube" indicators

d) Containment Water Level

1. Assure water is going into core, not on floor.

2. Anticipate submergence of equipment and instruments ($\sim$52" above floor to inst. junction box).

2. Existing Instruments:

a) V.C. Pressure

b) Sump Levels

c) Incore Thermocouples

1. Indicate actual core temperatures and flow blockage.

d) Wide Range RTD's (T hot & T cold).
Wide Range Pressure
Saturation Pressure Margin

1. Ensure RCS at subcooled condition.

2. Ensure RCP NPSH requirements.

e) NIS Source Range

1. Ensure reactor is shut down.

f) Steam Generator Wide Range Level

1. Ensure adequate heat sink.

g) Steam Generator Pressure

1. Ensure decay heat removal.

2. Back-up temperature indication (steam tables).

h) S.I. Flows (High Hd & Low Hd)
S.I. Pressure

1. Ensure coolant being delivered to core.

2. Detect ruptured injection line, or line feeding break.

i) Spray flow (recirculation)

1. Ensure spray fluid delivery.

j) NaOH addition flow and NaOH tank level

1. Indicates Containment Spray Pump operation.

2. Assures NaOH addition to spray fluid for iodine removal. Also, raising pH will decrease stainless corrosion rate.

k) Auxiliary Feedwater Flow

1. Ensure adequate heat sink.

l) RWST level.

1. Ensure injection and spray flow.

2. Leave a reserve for later spray if needed.

2.7 Fuel/Clad Materials Behavior in Inadequate Core Cooling Case.

1. At 1585°F, Zircalloy exhibits allotropic characteristics, i.e. two crystalline structures. Poor mechanical properties at high temperatures.

2. At ~ 2200°F, Zirc-Water reaction takes place in the presence of water vapor.

a) Zirc-water reaction has been observed to occur at temperatures as low as 750°F in water. NOTE: FSAR puts peak clad temperature at 715°F during normal operations.

b) $Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2$ + heat.

c) Reaction proceeds autocatalytically due to heat produced up to ~ 3000°F, where actual melting occurs.

d) H_2 generation will be completed during first day following LOCA.

3. ZrO_2 - UO_2 eutectic (melting point).

a) ZrO_2 - UO_2 will not combine as they are already bound in their oxide forms.

1) UO melts at ~ 4600F (~ 2550°C).

2) ZrO_2 melts at ~ 4900°F (~ 2700°C).

b) However, Zr and U will combine. For a 45% Zr/55° U mix, the resultant melting temperature is 1125°F (~ 607°C).

4. Zircalloy embrittlement

a) Zircalloy contains nickel, which absorbs nitrogen. Absorption of N_2 results in break away of protective oxide film. The zirconium then absorbs hydrogen, causing embrittlement.

2.8 Emergency Reporting Requirements (Refer to Enclosure #8).

1.0 <u>Title</u>: Hydrogen Hazards During Accidents

2.0 Sources of H_2 in containment following LOCA:

1. Dissolved H_2 in the coolant for O_2 scavenging

2. Zirc-water reaction:

   a) $Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2 + \text{heat}$

3. Corrosion of plant materials

   a) Aluminum is only metal that will significantly add H_2 to the containment atmosphere, primarily due to NaOH addition during containment spray.

   b) $2Al + 2NaOH + 2H_2O \quad 2NaAlO_2 + 3H_2 \uparrow$
   - 20 scf H_2/lbm Al corroded.

   c) It is desirable to limit the amount of aluminum in the containment.

4. Radiolysis of water

   a) $2H_2O \rightleftarrows 2H_2 \uparrow + O2 \uparrow$

2.1 Sources of O_2 in Containment following LOCA

1. O_2 in containment atmosphere prior to accident

2. Radiolysis of water (see above)

3. WCCPP in leakage

3.0 Hydrogen Combustion

1. Limits of Combustion

   a) Near room temperature in air containing between 1ppm and 10% moisture, the lower hydrogen limits are as follows:

1. Upward flammability is 4% hydrogen

2. Downward flame propagation is 8% hydrogen

3. Detorability is 19% hydrogen

NOTE: Combustion of hydrogen mixtures between 4% and 9% is often incomplete, leaving above 4% unburned.

b) When the temperature and water content are high, combustibility will be limited by the O_2 concentration. Existing data indicate that between 300°F and 500°F with $>$ 50% steam, the lower limit for flammability occurs at $\sim$ 4% O_2, with detonability repressed at $<$ 9% O_2.

4.0 Hazardous Characteristics of H_2 Explosions

1. H_2 has 50,000 to 60,000 BTU/lbm (Propane has $\sim$ 20,000 BTU/lbm)

2. Theoretical calculations show that between 8% and 10% hydrogen, combustion yeilds an eight-fold increase in containment pressure.

3. When H_2 concentration reaches 19%, combustion transition to detonation occurs, causing a seventeen-fold increase in containment pressure.

NOTE: An actual spark is not required for detonation at these concentrations, but rather may be caused by a physical shock.

5.0 Hydrogen Measurement

1. Hydrogen will be continuously monitored by the new containment hydrogen monitoring instruments.

a) Sample is taken off R-11/12 sample line, and will be isolated on a Phase A Isolation.

b) NOTE: Fan cooler unit 32 and/or 34 must be in service to obtain a representative sample.

2. During a LOCA, containment atmosphere may be sampled using the PACASS at all times.

3. Post LOCA hydrogen reduction

a) If $>$ 4%, use installed H_2 recombiners

b) If $<$ 4%, use installed PACVS

6.0 Post LOCA RCS Venting

1. Refer to PEP-ES-7, Removal of a Non-Condensable Bubble in the Reactor Coolant System (Enclosure #9).