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CORPORATION

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Mr. John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
NUREG-0737; Items I.A.2.1 and II.B.4
Upgrading SRO and RO Training and Training for Mitigating Core Damage

Dear Mr. Stolz:

By letter dated April 1, 1982, the Nuclear Regulatory Commission (NRC) requested additional information on NUREG-0737, Item I.A.2.1, "Upgrade of Senior Reactor Operator and Reactor Operator Training," and Item II.B.4, "Training for Mitigating Core Damage."

Florida Power Corporation (FPC) hereby provides responses to your eight questions as follows:

Item I.A.2.1 Upgrade Reactor Operator and Senior Reactor Operator Training

Question 1. Is the material on heat transfer, fluid flow, and thermodynamics which is taught in the training and requalification programs at a level of detail comparable to that identified in Enclosure 2 of Denton's March 28 letter?

Response 1. The material taught on heat transfer, fluid flow, and thermodynamics is of a comparable level of detail as that identified in Denton's March 28, 1980 letter.

Question 2. Are the training and requalification courses on mitigating the consequences of an accident involving a severely damaged core complete, and in effect at the present time? Do they follow the guidelines in Denton's March 28, 1980 letter?

Response 2. The training and requalification courses on mitigating core damage are complete and in effect at the present time. They follow the guidelines in Denton's March 28, 1980 letter.

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Question 3. Are the lectures and quizzes on the subject of accident mitigation given to shift technical advisors and operating personnel from the plant managers through the operations chain to the licensed operators? If they are, provide the titles of the people who are trained and an organization chart which illustrates their position in the operations chain.

Response 3. The lectures and quizzes on accident mitigation are given to the shift technical advisors and operating personnel from the plant managers down through the operations chain to the licensed operators. For the latest organizational chart, reference Technical Specification Change Request Number 67, Rev. 1 dated January 19, 1982. The titles of personnel who attend the training are as follows: (1) Nuclear Plant Manager; (2) ChemRad Manager; (3) Technical Services Superintendent; (4) Operations Superintendent; (5) Health Physics Supervisor; (6) Health Physics Technician; (7) Instrument and Controls Supervisor; (8) Instrument and Controls Technician; (9) All Licensed Operators; (10) Nuclear Operations Technical Advisor; and (11) Licensed Operator Candidates.

Question 4. Do the portions of the training program and requalification program which address heat transfer, fluid flow, thermodynamics, and accident mitigation involve 80 contact hours? (A contact hour 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 4. The training and requalification program does not involve 80 contact hours of training in heat transfer, fluid flow, thermodynamics, and accident mitigation. Each program does include sixteen (16) hours of accident mitigation and forty (40) hours of fluid flow, thermodynamics, and heat transfer training.

Question 5. Does the requalification program which the instructors take address current operating history, problems, and changes in procedures and administrative limits?

Response 5. The instructor requalification program does address current operating history, problems, and changes in procedures and administrative limits.

Question 6. As specified in Enclosure 1 of Denton's March 28, 1980 letter, does the operator requalification program call for accelerated requalification if the overall score is less than 80%, and the score in any category is less than 70%?

Response 6. The operator requalification program does call for accelerated requalification if the overall score is less than 80%, and if the score in any category is less than 70%.

Question 7. Does the operator requalification program call for control manipulations as specified in Enclosure 4 of Denton's letter of March 28, 1980, to all power reactor applicants and licensees?

Response 7. The operator requalification program does call for control manipulations as specified in Denton's March 28, 1980 letter.

Item II.B.2 Training for Mitigating Core Damage

Question 8. Provide an outline of the training program for mitigating core damage, including the number of training hours involved. Your outline can include any training program which relates to the training for mitigating core damage. Follow the guidelines given in the Enclosure 3 of Denton's letter dated March 28, 1980, and INPO Guidelines for Training to Recognize and Mitigate the Consequences of Core Damage (Document Number STG-01, Rev. 1, January 15, 1981). NRC requires minimum of 80 contact hours of training for mitigating core damage.

Response 8. FPC's training program includes sixteen (16) hours of mitigating core damage training and forty (40) hours of training in fluid flow, thermodynamics, and heat transfer. An outline of this training program is attached. From conversations with the NRC staff, FPC understands that the minimum of 80 contact hours of training for mitigating core damage (and related subjects) is not required by the NRC. FPC's training program does address and satisfy the concerns and intent of the INPO recommendations.

If you have any further questions, please contact this office.

Very truly yours,

David G. Mardis

David G. Mardis
Acting Manager
Nuclear Licensing

Attachment

RAW:mm

cc: Mr. J. P. O'Reilly, Regional Administrator
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Management Personnel (6)

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COURSE OUTLINE

DEGRADED CORE TRAINING

Part I: Preventing Degraded Core Conditions

Lesson 1 - Core Cooling Mechanics

Lesson 2 - Gas/Steam Binding Affects on Core Cooling

Lesson 3 - Boron Precipitation Concerns Following a LOCA

Lesson 4 - Equipment Failure Sequences that Could Lead to a Degraded Core

Lesson 5 - Avoiding Degraded Core Conditions

Part II: Recognizing a Degraded Core

Lesson 6 - Consequences of Inadequate Core Cooling and Likely Core Damage Effects

Lesson 7 - Use of SPND's in the Recognition of Degraded Core Conditions

Lesson 8 - Detection and Treatment of Inadequate Core Cooling Using Core Exit Thermocouples

Lesson 9 - Relationship of Out-of-Core (OCD) Source Range Detectors to Degraded Core Conditions

Lesson 10- Incore Thermocouples and Core Flow Blockage

Lesson 11- Release of Fission Products from Damaged Fuel

Lesson 12- Fission Product Transport Characteristics and Release Pathways

Lesson 13- Response of Gamma Radiation Monitors

Lesson 14- Chemical and Radiochemical Sampling Problems

CONTENTS

<u>CONTENTS</u>

Lesson 1 — CORE COOLING MECHANICS

Introduction

1. Lecturer —
2. Purpose — To describe the basic principles of heat transfer from the core, the effects of loss of this heat transfer, and the reasons for disruption of normal circulation patterns.

Objectives

The following subjects will be covered during this lesson:

1. Basic heat transfer from the core to the ultimate heat sink.
2. The factors affecting this heat transfer and the operator's control over them.
3. Heat transfer characteristics of the various states of water.
4. Natural circulation theory.
5. High-pressure injection (once-through) method of core cooling.

Key points to be retained are as follows:

1. The various methods available for cooling the core.
2. How to interpret primary system status from saturation curves.
3. Factors that affect natural circulation.
4. How the operator can effect heat transfer from the core to the ultimate heat sink.

Lesson 2 — GAS/STEAM BINDING

Introduction

1. Lecturer —

2. Purpose — To describe situations that could lead to introduction of large quantities of gases into the RCS. To list potential gas sources — condensible and non-condensible gases. To discuss potential gas effects on natural circulation and on OTSG heat removal. To give guidance for recognizing and dealing with those gases.

Objectives

The following subjects will be covered during this lesson:

1. Why gases might enter the RCS.
2. What gases might do once inside the RCS.
3. How an operator might recognize and deal with gas presence and effects.

Key points to be retained are as follows:

1. There are situations which could lead to gas volumes in the RCS.
2. Once there, gases can affect natural circulation and heat removal through the OTSGs.
3. These effects can be recognized and dealt with by the operator.
4. Actions are dependent on availability of certain equipment (e.g., OTSGs, HPIs, RC pumps).

Lesson 3 — BORON PRECIPITATION CONCERNS FOLLOWING A LOCA

Introduction

1. Lecturer —

2. Purposes —

   a. To explain mechanisms for obtaining high boron concentrations in the core following a LOCA.

   b. To discuss the dangers of high boron concentrations relative to long-term, post-LOCA core cooling.

   c. To discuss the interface between natural circulation and operator-induced circulation in preventing boron precipitation.

   d. To outline the long-term flow requirements to prevent boron precipitation.

Objectives

The following material will be presented during this lesson:

1. How the boron precipitation problem can arise.

2. The dangers of boron precipitation and how these could lead to a degraded core situation.

3. Natural circulation and subsequent operator-initiated circulation to prevent boron precipitation.

4. Specific results of operator actions.

The key points to be retained are as follows:

1. How and why boron precipitation can become a problem following a LOCA.

2. How long the operator can wait to take action before the danger point is reached.

3. In general, what actions (including NRC requirements) must be taken to manage the boron precipitation problem on a long-term basis.

4. There is nothing to indicate to the operator that boron precipitation is causing a problem. The best approach is to prevent rather than correct.

Lesson 4 — EQUIPMENT FAILURE SEQUENCES THAT COULD LEAD TO A DEGRADED CORE

1. Functional Failures

A degraded core could evolve from several vulnerable plant conditions. Such conditions include small LOCAs (including a stuck-open pressurizer safety valve and RC pump seal rupture), loss of normal heat sink, and loss of offsite power. In all cases, additional failures are required to cause inadequate core cooling. Any event path leading to a degraded core has a low probability of occurrence. Before addressing combinations of equipment failure and time frames, it is important to note failures in system functions — for example, a small LOCA with a loss of the HPI function could lead to a degraded core condition. Similarly, loss of the secondary side cooling function with loss of the HPI function or tripping of RC pumps during a small break while the RC void fraction is 70% or greater could also lead to a degraded core condition.

Loss of any system function can result from combinations of independent equipment failures, operator errors, or common-cause failures. Combinations of independent equipment failure in different systems, e.g., MFW pumps, auxiliary feed pumps, and HPI pumps, are a remote possibility. It is the intent of these lesson plans and ATOG to lessen the probability of operator error that would propogate a degraded core condition (e.g., turning off HPI and auxiliary FW), leaving common-cause failures as the most probable (but still very small) path to inadequate core cooling conditions. These include hardware common causes and the operator common cause (which is the result of conflicting information being displayed to the operator). Lesson 5 describes recognition and procedures to employ for these conditions.

2. Scenarios

2.1. Natural Catastrophic, Sabotage, and ATWS Scenarios

These sequences are beyond the immediate scope of these lesson plans although operator corrective actions can be taken to mitigate their effects. The initiating event itself is outside the control of the operator, but depending on the resulting effects of the initiating event, some portions of the ATOG guidelines are applicable. For example, if the initiating event caused a loss of secondary heat removal, operator actions would coincide with guidelines for loss of all feedwater. It should be noted that ATWS sequences are manageable if secondary heat removal is available and if peak ATWS pressure does not result in primary system rupture (if it ruptures, it is a LOCA condition). B&W plants also have a manual trip and runback capability in the event of failure of the RPS breakers. HPI boration should also be attempted as soon as possible (the initial primary system pressure spike may be too large for the HPI pumps to deliver water, but they should be flowing as pressure drops). HPI initiation will be by manual operator action. To summarize, if the RPS fails to drop the rods, try to run the rods in manually while initiating HPI flow. It should be noted that this course of action is consistent with ATOG procedures in that HPI flow should be established on loss of subcooling margin.

2.2. Scenarios Terminated by Procedural Action

Most of the corrective actions for the INPO vulnerable plant conditions have been accounted for in the first lesson plan. This section addresses only those items not discussed in Lesson 1. Vulnerable plant conditions identified by INPO include the following:

1. Loss of offsite power while one onsite power source is out of service. Loss of offsite power results in loss of main feedwater. If the diesel supplying the motor-driven emergency feed pump is the one that is out of service, secondary side heat removal is dependent on the turbine-driven pump (unless condensate/fire pumps, etc. can be aligned and are supplied by the good diesel). If the HPI that is aligned to the good diesel is out of service (e.g., down for maintenance), the HPI function is lost unless the standby HPI pump

is aligned (or can be through a swing arrangement*) to the good diesel. Unless these multiple equipment failures (feedwater and HPI pumps) occur, normal procedural action terminates this event sequence. If the multiple failures do occur, this event would be classified under section 3 (common cause and causal failures).

2. Extended station blackout. If the emergency turbine feed pump is available, secondary side cooling is initiated and the event is within the scope of the guidelines. However, see section 3 for a discussion of vulnerability while operating in this mode.

3. Stuck-open pressurizer safety valve. This is a small LOCA, and guidelines are in place; however, depressurization to decay heat removal pressure is encouraged before the BWST is depleted.

4. BWR event. In its place we will consider a stuck-open steam safety valve or steam line break. The affected steam generator should be isolated by stopping feed flow, and cooldown should be initiated under the "one good OTSG" guideline.

5. Loss of normal heat sink following reactor/turbine trip. For loss of the secondary heat removal function, employ HPI cooling under Lesson 1 (part 5).

6. Loss of d-c control power to a 4160 V emergency safeguard bus. If this should occur, the diesel supplying the good 4160 V bus should be started and left running until the situation is rectified. If the diesel fails to start or fails to continue to run, a shutdown should begin. A loss of offsite power during this time is unlikely, but in case it should occur, a station blackout results as discussed in item 2 above.

7. Loss of automatic emergency turbine feedwater control. If the motor-driven emergency feedwater pump(s) are not available, then manual control of the turbine pump is necessary. Two methods of control

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*However, remember that if the third HPI pump is a swing pump, all supporting subsystems, such as cooling water, should also be aligned to some power service.

are possible: (1) manually start/stop pump based on steam generator level, or (2) continue to run the pump but regulate flow to the steam generator by varying the amount of recirculation to the condenser. The second method is preferable (as long as the bearing temperature on the turbine and pump are maintained in the safe region) due to the relatively large probability of failure of the turbine pump to start on demand (e.g., overspeed trips while coming up to load) and the fact that even small flow to the generators has a quenching (cooling) effect.

2.3. Scenarios Unaffected by Procedural Action

There are scenarios for which no guidelines are written. In short, these are the ones in which nothing operates properly, so the operator has little if anything to do concerning the primary system condition. No guidelines address the actions to be taken in the event of losses in the secondary heat transfer and high-pressure injection functions, or in fact for any combination of losses of functions that would prohibit eventual depressurization of the primary system. Under these conditions, the operator could adjust the containment function if it were operable. The reason guidelines are not written for these "everything fails" scenarios is that their probability is remote. The most likely of these improbable scenarios are the common-cause sequences, such as loss of all a-c power, and those in which the operator fails to act (analogous to "everything fails") or acts incorrectly due to inadequate feedback of plant conditions. The next section discusses the loss of all a-c power event sequence viewed from a common-cause/causal viewpoint. Some simplified event tree diagrams follow in section 4.

3. Common-Cause and Causal Failures

A loss of power is a common cause event that affects equipment in more than one system. There are three sources of high-voltage a-c power: (1) in-house, (2) offsite, and (3) emergency onsite. The pre-TMI setpoint (on PORV and high-pressure trip) for operation of B&W plants allowed for a possible runback of station power to house load in the event of load rejection. With the post-TMI setpoints, this capability is no longer available. Therefore, if offsite power is lost, only emergency onsite power is available. Most plants have two diesel generators for emergency onsite power. Safety systems, such as high-pressure injection, are loaded on their respective diesels (train A on diesel A, train B on diesel B), minimizing the consequences of failure of one diesel to start. If both diesels fail to start, only the steam-driven auxiliary feed pump is available as an active source for heat removal. In the event that flow blockages (e.g., closed valves, pump failure, or inadequate steam supply to drive the turbine pump) exist, then this source of heat removal is also lost. It should be noted that after a loss-of-feedwater event there is a limited amount of steam inventory (even if there are no stuck-open steam or atmospheric dump valves, which could be caused by loss of control power), so it is imperative that the auxiliary feed pump be started (if it has failed, for example, on overspeed trip) as soon as possible to supply feedwater not only for cooling purposes but also for steam generation to drive the pump. In the loss of all a-c power scenario, there is a limited amount of time before the d-c power source gives out. In addition, the loss of ventilation/air conditioning will eventually result in erroneous signals being displayed for operator information or used in automatic steam generator level control, even if d-c power is still available. This introduces additional event sequence paths that can lead to a degraded core condition.

Other common-cause initiating events include seismic events*, the effects of which are discussed in more detail in Lesson 5. Before the plant modifications on the separation of NNI-X and -Y, loss of power to either NNI would have been a

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*Note that seismic is not classified as natural catastrophic (section 2.1) because the more probable seismic events are the less severe or mild-amplitude occurrences which will not cause massive equipment damage but would generate conflicting information to the operator.

potentially serious initiating event for degraded core conditions due to the loss of indication or conflicting information to the operator. Now, however, with this separation — coupled with knowledge of where the instrumentation power is supplied from — the probability of this event sequence is minimized.

Lesson 6 — CONSEQUENCES OF INADEQUATE CORE COOLING AND LIKELY CORE DAMAGE EFFECTS

Introduction

1. Lecturer —

2. Purpose — To give the operator necessary background information on the likely consequences of sustained inadequate core cooling and the resulting progression of core damage.

Objectives

The following material will be presented during this lesson:

1. General physical and thermochemical description of the fuel and cladding.

2. Normal operating parameters, then accident (LOCA) conditions that effect the integrity of this system.

3. Direct consequences to the cladding and fuel of major decrease in cooling capacity at cladding surface and/or partial core uncovering.

4. Primary consequences of loss of fuel integrity.

Key points to be retained are as follows:

1. The general configuration of the fuel/fuel assembly.

2. Operational (i.e., pressure/temperature) limits of fuel assembly materials.

3. Mechanics of fuel failure under accident conditions.

4. The consequences of fuel failure to the primary system.

Lesson 6 Outline

1. Introduction

2. Fuel and Cladding System Description

    2.1. Fuel
    2.2. Cladding and Assembly Materials

3. Loss-of-Coolant Conditions

    3.1. Manual Operation
    3.2. LOCA

4. Direct Consequences of Core Uncovering to Cladding/Fuel

    4.1. Cladding Rupture
    4.2. Cladding Oxidation/Hydrogen Generation
        4.2.1. Zircaloy Oxidation
        4.2.2. Stainless Steel Oxidation
        4.2.3. Radiolytic Decomposition of Water
    4.3. Fuel/Cladding Eutectic Formation

5. Core LOCA Consequences and TMI-2 Estimates

    5.1. General Consequences
    5.2. TMI-2 Estimates

6. Summary

Lesson 7 - USE OF SPNDs IN RECOGNITION OF DEGRADED CORE CONDITIONS

Introduction

1. Lecturer —

2. Purpose — To investigate the use of self-powered neutron detectors (SPNDs) in the analysis of degrading or degraded core conditions.

Objectives

The following material will be presented during this lesson:

1. Description of the incore monitoring system (IMS).

2. Operation of the IMS under normal conditions: at power and during shutdown.

3. Response of SPNDs to high temperatures: thermionic current and release of space charge.

4. Interpretation of SPND alarm data following reactor trips.

5. Limitations of SPND post-trip alarms.

Key points to be retained are as follows:

1. Nuclear and high-temperature responses of the IMS.

2. Use of alarm printer in interpreting SPND alarms.

3. Limitations on SPND post-trip alarms.

Lesson 7 Outline

1. Introduction

2. Description of Self-Powered Neutron Detectors

3. Operation of SPNDs Under Normal Conditions

4. High-Temperature SPND Response

    4.1. Thermionic Emissions — 1979 LRC Experiment
    4.2. Space Charge Release Mechanism
    4.3. Other Phenomena
        4.3.1. Interaction of Thermionic Emissions and Space Charge Release
        4.3.2. Presence of Gamma Radiation and Electronic Circuitry

5. Interpretation of SPND Alarms After Reactor Trips

    5.1. Axial and Radial Distribution
    5.2. High Temperature Vs High Flux

6. Limitations of SPND Post-Trip Alarm Analysis

7. Conclusions

8. Summary

References

Lesson 8 — DETECTION AND TREATMENT OF INADEQUATE CORE COOLING USING CORE EXIT THERMOCOUPLES

Introduction

1. Lecturer —

2. Purposes —

   a. To discuss the changes in core cooling efficiency as the reactor coolant changes state and flow conditions.

   b. To discuss core exit thermocouple (T_c) use as a means of detecting dangerous conditions in the core.

   c. To describe analyses which resulted in the core exit T_c readings and cladding temperatures (T_{clad}).

   d. To provide instructions for converting T_c readings to T_{clad}.

   e. To discuss required operator actions during high T_{clad} (inadequate core cooling) conditions.

   f. To briefly discuss T_c limitations.

Objectives

The following material will be presented during this lesson:

1. What might cause the reactor coolant to change state and how will that affect core temperatures?

2. How might the operator be able to use T_c's to determine if dangerous conditions exist, and the degree of danger?

3. How are T_c readings and cladding temperature related, and how was relationship established?

4. What actions must operators take based on T_c readings?

5. How might these actions affect the core and the rest of the plant?

6. What are the limitations of the T_c's?

The following key points to be retained are:

1. What influence does the amount of available reactor coolant and the state of that coolant have on core temperatures?

2. What damage would be expected at high core temperatures?

3. How T_c's can measure existence and magnitude of danger to core cooling.

4. Some familiarity with analyses used to develop T_c/T_{clad} correlation.

5. What the operator should be doing for given T_c readings.

6. What the results of these actions might be for the core and the rest of the plant.

7. How T_c readings might be misleading.

8. What to expect for cladding temperatures above 1800F.

Lesson Outline

1. Introduction
2. Causes of Inadequate Core Cooling (ICC) and High Core Temperatures
3. Normal and Off-Normal Modes of Heat Transfer
4. Why and How Cladding Temperature is Measured
5. ICC Guidelines Defining Operator Actions Based on T_{clad}
6. Possible Results and Consequences of Operator Actions
7. Shortcomings of T_c's

Lesson 9 — RELATIONSHIP OF OCD SOURCE RANGE DETECTORS TO DEGRADED CORE CONDITIONS

Introduction

1. Lecturer —
2. Purpose — To provide interpretations of abnormal source range monitor readings after a reactor trip.

   Although the primary function of out-of-core flux detectors is to monitor reactor flux levels during approach to criticality and power operation, an analysis of post-accident TMI events has indicated that the source range monitors (SRMs) are sensitive to changes in primary coolant "apparent density" and changes in certain core conditions that affect neutron generation.

Objectives

The following material is covered in Lesson 9:

1. General description of out-of-core detector (OCD) system.
2. Brief discussion of physical processes involved — neutron generation in core, criticality, neutron transport.
3. Parametric effect of source generation and void fraction on SRM readings.
4. Discussion of normal and abnormal (TMI-2) SRM readings after a reactor trip.

The following key points are to be retained:

1. Above-normal SRM readings after trip can be caused by any one or a combination of the following: recriticality, fuel failure with fission product release, core voiding, downcomer voiding, and/or coolant temperature increase.
2. "Normal" SRM readings after trip can vary due to startup source strength, core multiplication, and/or coolant temperature.
3. SRM readings should decrease continually after trip, and the operator should be wary of any increase in count rate.

LESSON OUTLINE

1. Source Range Monitor Response After Reactor Trip
   - 1.1. Typical SRM Response (Normal Trip at Oconee 3)
   - 1.2. SRM Response After TMI-2 Accident
2. Out-of-Core Detector System
3. Neutron Transmission From Core to SRM
   - 3.1. Neutron Transport
   - 3.2. Homogeneously Distributed Voids
   - 3.3. Coolant Temperature
   - 3.4. Coolant Level
4. Neutron Generation After Reactor Trip
   - 4.1. Incore Sources
   - 4.2. Subcritical Multiplication
   - 4.3. Excore Sources
5. Reactor Events
   - 5.1. Recriticality
   - 5.2. Loss of Coolant
   - 5.3. Coolant Boiling
   - 5.4. Core Damage
   - 5.5. False SRM Signals Due to Gamma Radiation
6. SRM Chart Interpretation
7. Summary

References

Lesson 9 — RELATIONSHIP OF OCD SOURCE RANGE DETECTORS TO DEGRADED CORE CONDITIONS

1. Source Range Monitor Response After Reactor Trip

Source range monitors (SRMs) are part of the out-of-core detector (OCD) system in B&W reactors and, as such, are located in the reactor cavity (Figure 9-1). Their major function is to measure a reactor's approach to criticality prior to startup. Observations after the TMI-2 accident indicated that the SRM readings could be confusing, but if properly interpreted, they could be useful monitors of core behavior and system conditions during certain accidents. Because of the SRM locations — away from the harsh environment in the core — SRM signals should be a rapid, reliable measure of neutron flux in the reactor cavity which is normally proportional to core flux. However, interpretation of SRM signals is not straightforward during accidents that could lead to a degraded core because the signal responds to changes in both neutron transmission and neutron generation. Thus, supporting data from other installations must be considered along with operator action to define the cause of anomalous SRM readings following accidents.

Neutrons that are born in the core region must pass through several steel and water regions before reaching the detector locations. The shaded area in Figure 9-1 represents the peripheral fuel assemblies in the core where neutrons that are born have a good chance of escaping the core. The unshaded area represents the inner fuel assemblies (later referred to as the inner core), where changes in neutron generation rate will not significantly affect the core escape flux or the SRM response.

The fraction of neutrons transmitted (or its reciprocal, the attenuation factor) is affected by the mass of water between core and cavity. Water mass can vary due to density changes associated with temperature changes and/or voiding due to loss of coolant via some mechanical malfunction or human error. The neutron generation rate is affected by the mass of coolant in the core region

and any other parameters that alter core reactivity, such as coolant void, coolant temperature, boron concentration, control rod location and integrity, and fuel pin integrity. The complexity of the problem becomes apparent when one considers that these phenomena have competing dependent and/or multiple effects on SRM responses.

In this lesson the SRM responses after a typical reactor trip and after the TMI-2 accident trip are discussed. We will consider some of the reactor conditions that perturb SRM responses and then provide suggestions for interpreting SRM charts in accident situations.

1.1. Typical SRM Response (Normal Trip at Oconee 3)

The SRM response to a normal reactor trip is shown in Figure 9-2. Initially, the power (and therefore the SRM signal) drops with an 80-second period, consistent with the 55-second half-life delayed neutron group. During the first 15 to 20 minutes following reactor trip, the delayed neutrons from ^{235}U fission products are the dominant source of neutron flux.

As the delayed neutrons die out (becoming relatively insignificant after about 20 minutes), the photoneutrons from γ,n reactions with D_2O (in the primary coolant) become the dominant source of neturons. This source decays with a variable half-life of 1 to 2 hours (over the time period of about 0.5 to 4 hours after trip) and becomes relatively small by about 10 hours after trip.

After about 10 hours the SRM signal tends to be essentially constant because it is then responding to neutrons from the startup source.

The absolute value of the SRM signal following a normal reactor trip may vary from one reactor to another and even between subsequent trips in the same reactor. However, the general shape should be maintained. Some of the following items could affect the SRM reading:

1. Power history several days prior to trip and, to some extent, burnup — This determines fission product concentration, which is the source of gamma flux for photoneutron production.

2. Coolant temperature variations — This changes the attenuation of neutrons from core to detector.

3. Startup source activity — Depending on the type of startup source, the neutron output may either decrease or increase with time after installation in the reactor.

Lesson 10 — INCORE THERMOCOUPLES AND CORE FLOW BLOCKAGE

Introduction

1. Lecturer —
2. Uniqueness — Incore thermocouples are unique in that they are the only temperature instruments monitoring below a corewide scale.
3. Purpose — To evaluate the capability of incore thermocouple to provide adequate indication for reactor operators to recognize core flow blockage.

Objectives

The following material will be presented during this lesson:

1. Use of incore thermocouples to estimate core flow.
2. Comparing core flow to reactor vessel flow.
3. Estimating core flow blockage.
4. TMI-2 experience.

The key points to be retained are as follows:

1. How to calculate and compare core flow and reactor vessel flow,
2. How to detect and estimate core flow blockage.

Lesson 10 Outline

Lesson 11 — RELEASE OF FISSION PRODUCTS FROM DAMAGED FUEL

Introduction

1. Lecturer —

2. Purpose — To describe the buildup and distribution of fission products in the fuel and to quantify the amount of fission product activity expected to be released after various fuel damage scenarios.

Objectives

Lesson 11 covers the following material:

1. The basic characteristics that control the release of key fission products from the fuel (qualitative).

2. The buildup of key fission products as a function of irradiation time and the distribution of fission product activity (quantitative) between the fuel pellets, the fuel rod gaps, and the reactor coolant.

3. The progressive increase in fission product release associated with the various stages of core damage.

4. Rapid assessment of fuel damage based on the amount of key nuclides in the reactor coolant and the ratio of long- to short-half-life nuclides.

LESSON OUTLINE

1. Behavior of Fission Products
2. Fission Product Buildup and Distribution
3. Assessing Degree of Core Damage
4. Summary

Lesson 12 — FISSION PRODUCT TRANSPORT CHARACTERISTICS AND RELEASE PATHWAYS

Introduction

1. Lecturer —

2. Purpose — To describe the basic chemical characteristics that affect the transport of several groups of key fission product nuclides and to identify potential pathways for the release of these fission products to the containment or to the environment.

Objectives

The following material is covered in Lesson 12:

1. The basic chemical characteristics (qualitative) that affect the transport of key fission product nuclides.

2. Basic facts on how fission products will behave following a reactor accident.

3. Identification of potential pathways for the release of fission products into the containment or into the environment.

4. Rapid identification of fission product release pathways so that they can be expeditiously terminated.

The following key points are to be retained:

1. The noble gases will follow the steam and will accumulate wherever there is a gas phase, i.e., the pressurizer steam space, the makeup tank gas space, or the containment atmosphere.

2. The iodine activity can exist in many forms, but in general if water is readily available, $99.9\%^{+}$ will tend to be in the liquid phase.

3. If for some reason the accident is a dry one, i.e., no water in the reactor and no steam in the containment atmosphere, there is likely to be a very large source of iodine and cesium aerosols airborne in the containment, which should be removed as quickly as possible using the reactor building sprays.

4. The activity release pathways that develop during a reactor accident are not necessarily the intuitively obvious pathways but are more likely to be obscure, forgotten pathways.

LESSON OUTLINE

1. Chemical and Transport Characteristics of Fission Products

    1.1. Noble Gases

    1.2. Iodine

        1.2.1. Elemental Iodine
        1.2.2. Hypoiodous Acid
        1.2.3. Organic Iodide
        1.2.4. Cesium Iodide
        1.2.5. Particulates

    1.3. Cesium

2. Fission Product Release Pathways

    2.1. Pathways From RCS to Containment

    2.2. Pathways Into Auxiliary Building

    2.3. Pathways to the Environment

    2.4. Release Pathways Identified at TMI-2

3. Summary

Lesson 13 — RESPONSE OF GAMMA RADIATION MONITORS

Introduction

1. Lecturer —

2. Purpose — To describe the responses that should be expected from various gamma radiation monitors following a reactor accident and to alert the operators to anomalous readings that may occur due to high background radiation, instrument failure, improper calibration, and source concentration gradients.

Objectives

The following material is covered in Lesson 13:

1. The magnitude of the letdown line monitor response as an indication of the degree of core damage.

2. Anomalous readings that have occurred in process radiation monitors and area radiation monitors following the TMI-2 accident and the CR-3 loss of non-nuclear instrumentation event.

The following key points are to be retained:

1. Expected letdown monitor response.

2. Effects that can cause anomalous readings:
   a. High background radiation.
   b. Scale and calibration errors.
   c. Shielding of detectors.
   d. Mixture of fission product nuclides.
   e. Source concentration gradients.
   f. Electrical power supply failures.
   g. Electronic component failures.

LESSON OUTLINE

1. Letdown Line Monitor
2. Process Monitors
3. Area Monitors
4. Reactor Building Dome Monitor
5. Failure Modes
6. Summary

Lesson 14 — CHEMICAL AND RADIOCHEMICAL PROBLEMS

Introduction

1. Lecturer —
2. Purpose — To describe some of the chemical and radiochemical problems that can occur following a reactor accident that results in degraded core conditions.

Objectives

The following material is presented in Lesson 14:

1. Chemical and radiochemical problems to be expected following reactor accidents that result in degraded core conditions.
2. Interpreting the analytical results, which will appear to be anomalous if compared to normal operating conditions.
3. How to avoid making operational or procedural errors that might lead to plant and equipment contamination as excessive radiation exposure to operating personnel.
4. Avoiding both sampling problems and the use of incorrect analytical methods.